Key Words:
Performance Assessment
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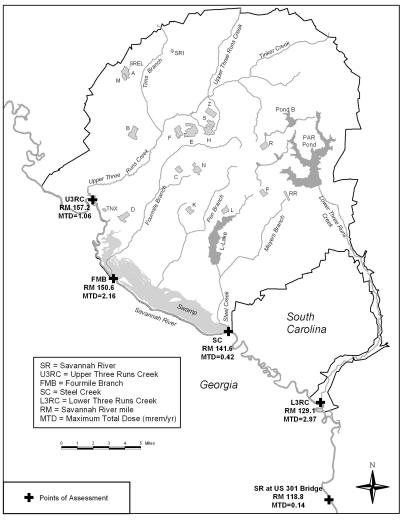
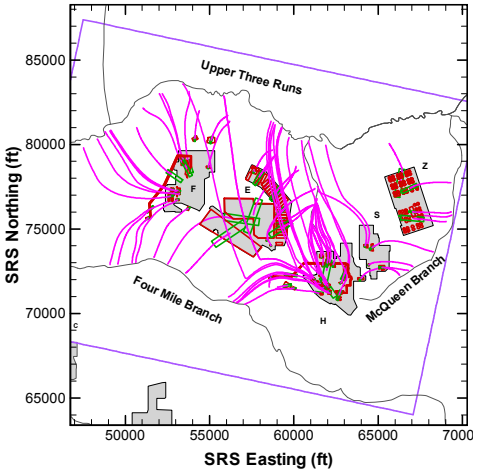
Retention:
Permanent

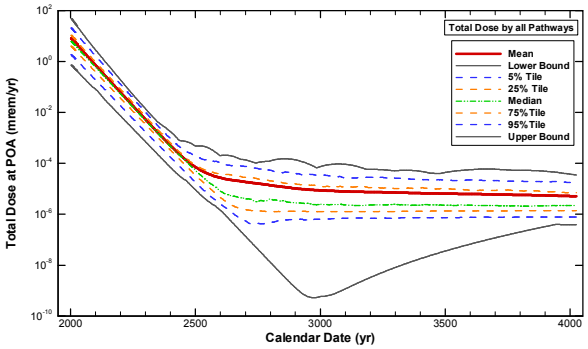
SAVANNAH RIVER SITE DOE 435.1 COMPOSITE ANALYSIS

VOLUME I

JUNE 10, 2010







Savannah River National Laboratory
Savannah River Nuclear Solutions
Aiken, SC 29808

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronyms

AB	Actinide Billet
ACP	Area Closure Project (combination of two predecessor organizations: SGCP and SDD)
AEC	Atomic Energy Commission
ALARA	As low as reasonably achievable
ANL	Argonne National Laboratory
AP	Assessment Period
ARAR	Applicable or Relevant and Appropriate Requirement
ARP	Actinide Removal Process
ARRA	American Recovery and Reinvestment Act
ARU	Acid Recovery Unit
ASA	Auditable Safety Analysis
ASTM	American Society for Testing and Materials International
ATTA	Advanced Tactical Training Area
AWSA	Aqueous Waste Software Application
BCF	Bioaccumulation Factor
BGE	Burial Ground Expansion
BRA	Baseline Risk Assessment

CA	Composite Analysis
CAB	Citizens Advisory Board
CAG	C-Area Groundwater
C&WDA	Closure and Waste Determination
CBRP	C-Area Burning/Rubble Pit
CCS	Central Climatology Site
CDP	cellulose degradation product
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CIF	Consolidated Incineration Facility
CIG	Components-in-Grout
CLAB	Central Laboratory Facility
CLSM	Controlled Low-Strength Material
CLWR	Commercial Light Water Reactor
CMS	Corrective Measures Study
CMScreen	Contaminant Migration Screening
CNSI	Chem-Nuclear Systems, Inc.
COC	constituent of concern
CQF	Cognizant Quality Function
CPT	cone penetrometer
CR	Control Room
CRC	Cesium Removal Column
CRSB	C-Reactor Seepage Basin
CSRA	Central Savannah River Area
CSRACT	The Central Savannah River Area Community Team
CSSX	caustic side solvent extraction
CTS	Concentrate Transfer System

D&D	Deactivation and Decommissioning
DAS	Disposal authorization statement
DB	Diversion Box
DCF	Dose Conversion Factor
DCG	Derived Concentration Guide
DCS	Duke Cogema Stone and Webster
DG	Discharge Gully
DEXOU	D-Area Expanded Operable Unit
DNFSB	Defense Nuclear Facilities Safety Board
DOD	United States Department of Defense
DOE	United States Department of Energy
DOE-HQ	U.S. Department of Energy – Headquarters
DOE-SR	U.S. Department of Energy – Savannah River Operations Office
DP	distributed processing
DPFR	Decommissioning Project Final Report
DRI	Desert Research Institute
DSA	Documented Safety Analysis
DTCS	Demolish to Concrete Slab
DU	Depleted Uranium
DWPF	Defense Waste Processing Facility
EA	Exposure Area
ECP	E-Area Low-Level Waste Facility Completion Project
EDE	Effective Dose Equivalent
EIS	Environmental Impact Statement
ELLWF	E-Area Low Level Waste Facility
EM	Environmental Management
EPA	United States Environmental Protection Agency
ERDA	U.S. Energy Research and Development Administration
ERDMS	Environmental Restoration Data Management System
ET	Engineered Trench
ETF	Effluent Treatment Facility
ETP	Effluent Treatment Plant/Project

FAMS	F-Area Materials Storage (Facility)
FB	F-Area Button
FBSB	Ford Building Seepage Basin
FESV	Failed Equipment Storage Vault
FFA	Federal Facility Agreement for the Savannah River Site
FIPSL	F-Area Inactive Process Sewer Line
FMB	Fourmile Branch
FMF	Fuel Materials Facility
FRB	F-Area Retention Basin
FS	Feasibility Study
FSAR	Final Safety Analysis Report
FSB	F-Area Seepage Basin
FS-HQ	Forest Service Headquarters
FTF	F-Area Tank Farm
GCC	Geosynthetic Closure Cap
GCL	geosynthetic clay liner
GIS	Geographic Information System
GMS	Groundwater Modeling System
GOU	Groundwater Operable Unit
GSA	General Separations Area
GSACU	General Separations Area Consolidation Unit
GTS	Gas Transfer System
GWSB	Glass Waste Storage Building
HAD	High Activity Drain
HANM	H-Area New Manufacturing
HAOM	H-Area Old Manufacturing
HAW	High Activity Waste
HB	H-Area Button
HC	Hazard Category
HCA	high contamination area
HDPE	high density polyethylene
HDU	Highly Depleted Uranium
HEPA	High Efficiency Particulate Air
HEU	highly enriched uranium
HIPSL	H-Area Inactive Process Sewer Line
HLW	High Level Waste
HRB	H-Area Retention Basin
HSB	H-Area Seepage Basin
HTF	H-Area Tank Farm
HVAC	Heating, Ventilation, and Air Conditioning
HWCTR	Heavy Water Components Test Reactor
HWMF	Hazardous Waste Management Facility

ICRP	International Commission on Radiological Protection
ILNT	Intermediate Level - Non Tritium
ILT	Intermediate Level - Tritium
ILV	Intermediate Level Vault
IOU	Integrator Operable Unit
IPSL	Inactive Process Sewer Line
IS	Inner Swamp
ISCM	Integrated Site Conceptual Model
ISD	In-Situ Disposal
ITP	In–Tank Precipitation
KAC	K-Area Complex
KRSB	K-Reactor Seepage Basins
LAC	L-Area Complex
LAD	Low Activity Drain
LANL	Los Alamos National Laboratory
LAOCB	L-Area Oil and Chemical Basin
LASG	L-Area Southern Groundwater
LASL	Los Alamos Scientific Laboratory
LAWV	Low Activity Waste Vault
LDB	Leak Detection Box(es)
LFRG	Low-Level Waste Disposal Facility Federal Review Group
LLNL	Lawrence Livermore National Laboratory
LLRWDF	Low-Level Radioactive Waste Disposal Facility
LLW	Low-Level Waste
LLWF	Low-Level Waste Facility
LPPP	Low Point Pump Pit
LRSB	L-Reactor Seepage Basins
LTR	Lower Three Runs (refers to the creek unless otherwise noted)
LTS	Long-Term Stewardship
LUC	land use control
LUCAP	Land Use Control Assurance Plan
LUCIP	Land Use Control Implementation Plan
LWO	Liquid Waste Operations (taken over by SRR under a new DOE contract)

M&O	Maintenance and Operations
MCL	Maximum Contaminant Level
MCU	Modular Caustic Side Solvent Extraction Unit
MIPS	M-Area Inactive Process Sewer
MIPSL	M-Area Inactive Process Sewer Line
MMI	Modified Mercalli Intensity
MOP	Member of the public
MOX	Mixed Oxide Facility
MOXFF	Mixed Oxide Fuel Facility
MST	MonoSodium Titanate
MTD	Maximum Total Dose
MWMF	Mixed Waste Management Facility
NA	not applicable
NCP	National Contingency Plan
NCRP	National Council on Radiation Protection and Measurements
NDAA	National Defense Authorization Act
NEPA	National Environmental Policy Act
NERP	National Environmental Research Park
NFA	No Further Action
NNSA	National Nuclear Security Administration
NOAA	National Oceanic and Atmospheric Administration
NPDES	National Pollutant Discharge Elimination System
NR	Naval Reactor
NRC	Nuclear Regulatory Commission
NRCDA	Naval Reactor Component Disposal Area
NWS	National Weather Service
OA	Oxalic Acid
OD	Outfall Delta
OFASB	Old F-Area Seepage Basin
OF-F	Outside Facilities F-Area
ORNL	Oak Ridge National Laboratory
ORWBG	Old Radioactive Waste Burial Ground
OS	Outside Section
OST	Old Solvent Tanks
OTSB	Old TNX Seepage Basin
OU	Operable Unit
OWST	Organic Waste Storage Tank

PA	Performance Assessment
PB	Pen Branch
PCE	Tetrachloroethylene
PDCF	Pit Disassembly and Conversion Facility
pdf	probability density function
PEF	Plutonium Experimental Facility
PEIS	Programmatic Environmental Impact Statement
POA	Point of Assessment
PP	Pump Pit
PRA	Probability Risk Assessment
PRG	Preliminary Remediation Goals
PRSB	P-Reactor Seepage Basins
PuFF	Plutonium Fuel Form
PUREX	Plutonium-Uranium Extraction
QA	Quality Assurance
QAP	Quality Assurance Procedure
QC	Quality Control
RA	Remedial Action
R&D	Research and Development
RBOF	Receiving Basin for Offsite Fuels
RCRA	Resource Conservation and Recovery Act
RESRAD	Residual Radioactive Materials Computer Code
RFI	RCRA Facility Investigation
RI	Remedial Investigation
RI&ES	Regulatory Integration and Environmental Services
ROD	Record of Decision
ROI	region of influence
RRF	Resin Regeneration Facility
RRSB	R-Reactor Seepage Basins
RSO	Reservoir Surveillance Operations
RST1	Reactor Screening Software 1

SA	Special Analysis
S&M	Surveillance and Maintenance
SAR	Safety Analysis Report
SATA	Small Arms Training Area
SC	Steel Creek
SCDHEC	South Carolina Department of Health and Environmental Control
SC/PB	Steel Creek/Pen Branch
SDD	Site Deactivation and Decommissioning (subsequently combined with SGCP to become ACP)
SDF	Saltstone Disposal Facility
SDWA	Safe Drinking Water Act
SESOIL	Seasonal Soil Compartment Model
SEUSSN	Southeastern U.S. Seismic Network
SFP	Spent Fuel Project
SGCP	Soil and Groundwater Closure Projects (subsequently combined with SDD to become ACP)
SNL	Sandia National Laboratory
SOF	sum-of-fractions
SQAP	Software Quality Assurance Plan
SPF	Saltstone Production Facility
SPFF	Special Products Fabrication Facility
SR	Savannah River
SREL	Savannah River Ecology Laboratory
S/RIDS	Standards/Requirements Identification Document
SRIP	Site Regulatory Integration and Planning (subsequently replaced by RI&ES during contract split)
SRL	Savannah River Laboratory
SRLSB	Savannah River Laboratory Seepage Basins
SRNL	Savannah River National Laboratory
SRNS	Savannah River Nuclear Solutions
SRP	Savannah River Plant
SRR	Savannah River Remediation (new DOE contractor who took over LWO)
SRS	Savannah River Site
SRTC	Savannah River Technology Center
SS	stainless steel
ST	Slit Trench(es)
SUNY	State University of New York
SWDF	Solid Waste Disposal Facility
SWM	Solid Waste Management (previously designated WMAP)
SWPF	Salt Waste Processing Facility

TBG	TNX Burying Ground
TCE	Trichloroethylene
TEDE	total effective dose equivalent
TEF	Tritium Extraction Facility
TPBARs	Tritium Producing Burnable Absorber Rods
TRL	Three Rivers Landfill
TRU	Transuranic
TSCA	Toxic Substances Control Act
TSR	Technical Safety Requirement
UCNI	Unclassified Controlled Nuclear Information
UDQ	Unreviewed Disposal Question
USDOE	U.S. Department of Energy
USGS	United States Geological Survey
UTM	Universal Transverse Mercator
UTR	Upper Three Runs (refers to the creek unless otherwise noted)
V&V	verification and validation
VBA	Visual Basic for Applications
VEGP	Vogtle Electric Generating Plant
VZ	Vadose Zone
VZCOMML©	Vadose Zone Contaminant Migration Model - Multi-Layered
WAC	Waste Acceptance Criteria
WCS	Waste Characterization System
WHV	Waste Handling Vault
WIPP	Waste Isolation Pilot Plant
WMAP	Waste Management Area Project (subsequently designated SWM)
WQS	water quality standard
WSB	Waste Solidification Building
WSRC	Washington (or Westinghouse) Savannah River Company

Abbreviations

A.A.S.	Associate of Applied Science
B.A.	Bachelor of Arts
Bq	Becquerel
B.S.	Bachelor of Science
˚C	degrees Celsius
cfs	cubic feet per second
Ci	curie
Ci/lb	curies per pound
cm	centimeter
D.A.B.T.	Diplomate of the American Board of Toxicology
˚F	degrees Fahrenheit
ft	foot
ft^3/s	cubic feet per second
g	gram
gpm	gallons per minute
in	inch
k	kilo
K_d	distribution (or sorption or partition) coefficient
Kg	kilogram
K_v	vertical hydraulic conductivity
L	liter
lb	pound
L/Kg	Liter per kilogram
m	meter or milli
m^2	square meters
mi	mile
mrem	millirem
mrem/yr	millirem per year
M.S.	Master of Science
msl	mean sea level
n	nano
p	pico
pcf	pounds per cubic foot
pCi/g	picocuries per gram
pCi/mL	pico curies per milliliter
PhD.	Doctor of Philosophy
ppm	parts per million
s	second
ScD.	Doctor of Science
yr	year

1.0 EXECUTIVE SUMMARY

A Composite Analysis (CA) is required by the United States Department of Energy (DOE) Order 435.1 (DOE 1999a) to provide a reasonable expectation that DOE low-level waste disposal, high-level waste tank closure, and transuranic waste disposal ensure radiological protection of the public. This Order requires an accounting of all sources of DOE man-made radionuclides and DOE enhanced natural radionuclides that are projected to remain on the site after site operations have ceased. A 100 mrem/yr primary dose limit, based upon DOE Order 5400.5 (DOE 1990), has been established as the CA performance measure. However to prevent the potential dose from exceeding a significant fraction of the primary dose limit, a dose constraint (i.e., administrative dose limit) of 30 mrem/yr has also been established by DOE. The results of a CA are an estimated dose to a hypothetical member of the public at points of assessment, which are selected based upon the site's land use plans, over a minimum 1,000 year period after disposal facility and tank closure and/or all DOE site operations have ceased.

This CA report documents the projected cumulative impacts to future members of the public from the disposal of low-level radioactive waste (LLW), closure of radioactive liquid waste storage tanks, and potential disposal of transuranic waste at the Savannah River Site (SRS) and all other sources of residual radioactive material projected to be left at SRS that could interact with the disposal facilities and closure sites to affect the future radiological dose of a member of the public. The impacts were compared with the applicable DOE dose limit and constraint.

This CA satisfies the CA requirements identified in DOE Manual 435.1-1 (DOE 1999b), Sections IV.P. (3) and IV.P. (4)(a).

An SRS site description was compiled that is sufficiently detailed to support the development of the conceptual modeling carried out in the CA, to summarize existing information in support of conceptual model development on SRS facilities for which Performance Assessments (PAs) have, or are, being developed in the General Separations Area (GSA), and to provide an overview of related documentation that influences the CA. Following the identification of relevant points of assessment (POAs) and the specification of the assessment period (AP) of interest, screening analyses were conducted to focus the dose analyses on the significant radionuclides and the significant pathways of transport and exposure.

An intensive effort was undertaken to consult with custodial organizations for SRS facilities, waste tanks and waste sites, including the 515 waste sites listed in the SRS Federal Facilities Agreement (FFA), to identify those having a process history associated with radionuclides in order to establish the radionuclide inventory of all sources of residual radioactive material expected to remain on the SRS at the site end state. All identified sources of radioactive material projected to remain at SRS are included in this CA.

Projected doses to the receptor, a hypothetical future member of the public, from all sources were calculated for a 10,100-year period from 1950 to 12050. The doses from 41 of the sources for which doses had not clearly peaked within the 10,100-yr period were further run to year 102,050 to determine whether maximum doses had been calculated. These analyses included the following three periods:

1. The 1,000-year CA period of assessment required by DOE 435.1-1 (DOE 1999b), beginning at the projected site end-state date of 2025.
2. The 10,025-year period following the projected site end-state date. This period is not mandated by DOE 435.1-1 (DOE 1999b), rather these analyses provide information that is potentially of value for those facilities regulated by Section 3116 of the 2005 National Defense Authorization Act.
3. The post-compliance period (beyond 1,000 years). Analyses were also carried out to year 102,050 to provide perspective on the magnitude of doses beyond the compliance timeframe, as recommended by the DOE PA/CA Format and Content Guide (DOE 1999c).

Because the SRS land use planning foresees perpetual control of the current SRS land, with no residential use, the receptor is assumed to reside near the Savannah River, but not on the Savannah River Site, during all of these time periods. The receptor is assumed to use river water for residential and agricultural uses (e.g., drinking water, vegetables, meat, milk) and to recreate (e.g., boating, fishing) at the mouths of the onsite streams.

This CA focuses on groundwater and surface water transport pathways in calculating potential doses to the receptor. The groundwater transport pathway describes the movement of radionuclides that leach from the source and move vertically down through the vadose zone to the aquifer and transport through the aquifer to the point of outcrop to a surface stream or the river. The model uses stream and river flow rates to calculate concentrations of radionuclides in the streams or river from the flux of radionuclides outcropping from the aquifer and arising from former streambed contamination.

The CA model uses the GoldSim™ code to calculate the transport from each source to its respective POA and the resulting doses at each POA. Table 1-1 and Figure 1-1 and Figure 1-2 summarize the CA results. As shown in Table 1-1, the maximum dose over the 1,000-yr AP occurs at the Lower Three Runs (LTR) POA and is about three mrem/yr, indicating compliance with the 100-mrem primary dose limit and 30-mrem dose constraint. This maximum dose occurs at the projected site end-state date of 2025 and is due primarily to Cs137 contained within the sediment of the LTR streambed. Because Cs137 is the primary contaminant and it has a relatively short half-life of 30 years, the LTR POA dose quickly declines from its 3 mrem/yr high in 2025 to approximately 0.1 mrem/yr in 2150, at which point LTR is no longer the controlling POA. Extending the AP to 10,000 years does not increase the maximum dose.

Table 1-1. Maximum Cumulative Dose at each POA during CA Period of Assessment

Point of Assessment	Maximum Cumulative [1] Dose mrem/yr 2025 to 3025	Maximum Cumulative [1, 5] Dose mrem/yr 3025 to 12050	Major Contributing Source [2]	Major Contributing Radionuclide	Major Exposure Scenario/ Pathway
Upper Three Runs	1.06	0.40	H-Canyon	Np237	Recreational/ Fish Ingestion
Fourmile Branch	2.16	0.14	FMB IOU [3]	Cs137	Recreational/ Fish Ingestion
Steel Creek/Pen Branch	0.42	0.05	SC IOU	Cs137	Recreational/ Fish Ingestion
Lower Three Runs	2.97	0.05	LTR IOU	Cs137	Recreational/ Fish Ingestion
Savannah River	0.17 [4]	0.05 [4]	LTR IOU	Cs137	Residential/ Vegetable Ingestion

1 Sum of doses from the residential and recreational exposure scenarios, using the respective stream flow rate for recreational dose and the Augusta, GA, river flow rate, unless otherwise noted, for residential dose.
2 See Table C-1 for Source Identification corresponding to abbreviations given below.
3 IOU stands for Integrator Operable Unit, which are the stream and river beds.
4 Both residential and recreational doses are cumulative from all sources; the highway 301 bridge flow was used.
5 In all cases, the maximum dose in the 9,000 years beyond the 1,000 year assessment period occurred in year 3025.

Numerous sensitivity analyses were carried out to consider the effects of such factors as release rates, radionuclide inventories, alternative points of assessment, groundwater divides, stream flow variation, and alternative disposal actions on CA model results. A quantitative uncertainty analysis was also performed to assess the uncertainty in dose calculated from direct discharge of radionuclides to each POA. The results of both types of analyses provide great confidence that the dose to a member of the public will not exceed the performance measures (i.e., 100 mrem/year primary dose limit and the 30 mrem/year dose constraint).

In summary, there is a reasonable expectation that the performance measures identified for the CA will not be exceeded. An options analysis was deemed not necessary because the CA dose constraint of 30 mrem/year was not exceeded.

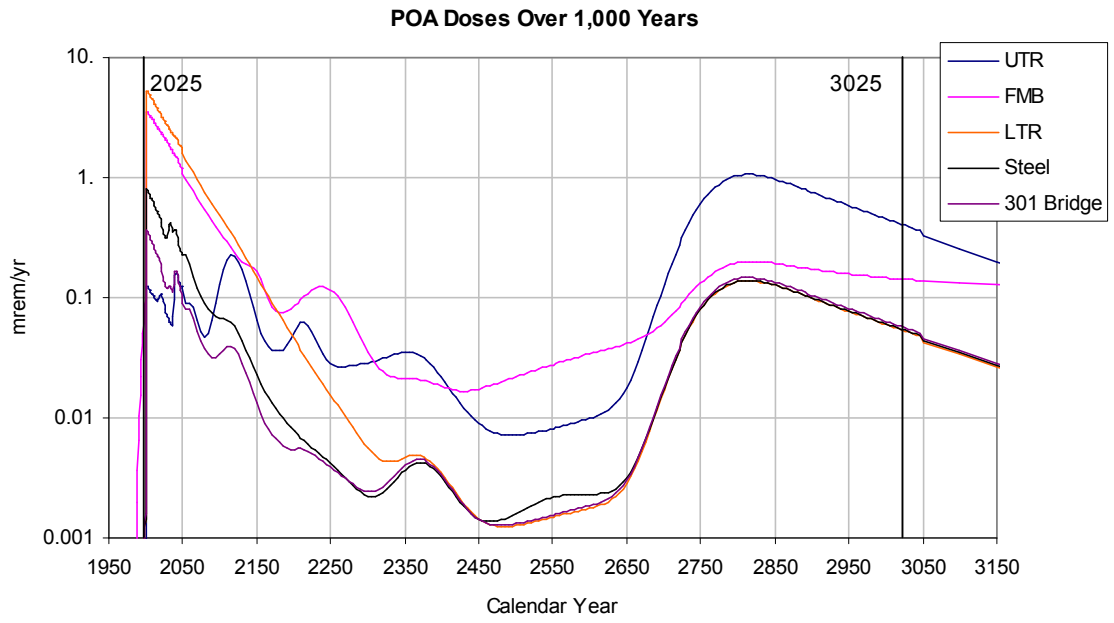

Figure 1-1. Total Cumulative Dose at POAs for 1,000 Year Assessment Period

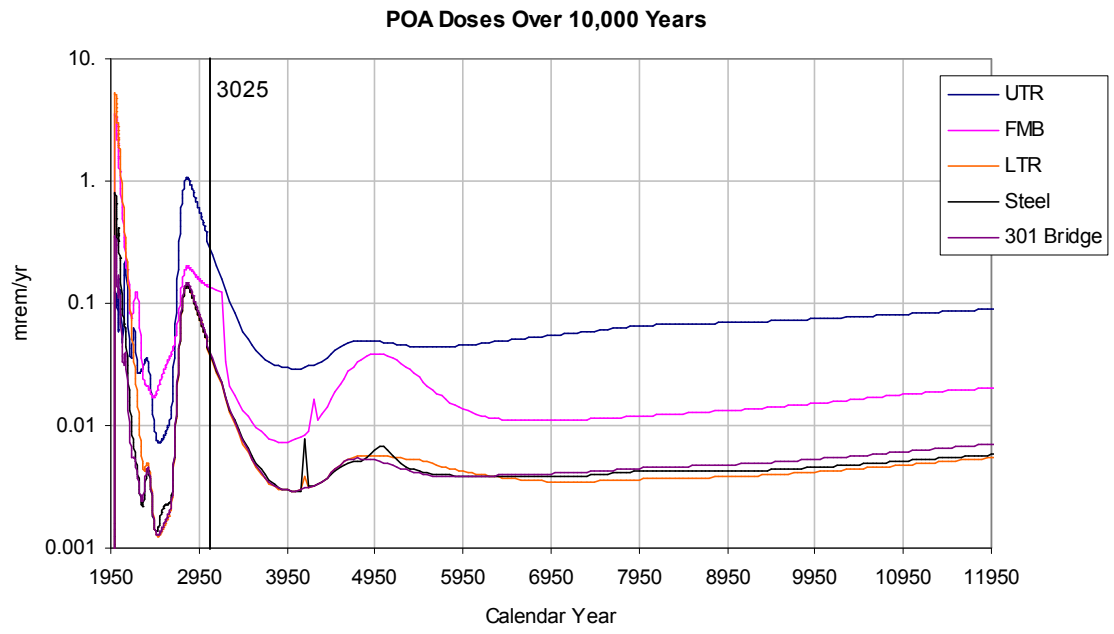

Figure 1-2. Total Cumulative Dose at POAs for Extended 10,000 Year Analysis Period

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2.0 REVISION HISTORY

This Composite Analysis (CA) is the first revision of the original CA that was issued in November 1997 (WSRC 1997). The following discussions in Sections 2.1 and 2.2 were taken directly from Section 1.0 in the SRS Composite Analysis Program Plan (Phifer and Cook 2007). These discussions provide details of the sequential developments that led to the current revision and list the major changes necessitating a CA revision. Finally, Section 2.3 provides a discussion of the major revisions to the original CA (WSRC 1997) made in this CA revision.

2.1 PURPOSE OF COMPOSITE ANALYSIS REVISION

A review of the United States Department of Energy (DOE) Performance Assessment (PA) process by the Defense Nuclear Facilities Safety Board (DNFSB) resulted in a commitment by DOE to "conduct a composite analysis that accounts for other source terms that add to the dose to a hypothetical future member of the public" associated with active low-level waste (LLW) disposal facilities (DOE 1996a). The DOE further committed to perform the required CA for the E-Area Low Level Waste Facility (ELLWF) and the Z-Area Saltstone Disposal Facility (SDF) by September 30, 1997 (DOE 1996a). Guidance for conducting a CA for low level waste disposal facilities was also provided by the DOE in 1996 (DOE 1996b).

Based upon the DOE CA guidance and the commitment made to the DNFSB, the Savannah River Site (SRS) issued a CA performed for the two active low-level radioactive waste disposal facilities to the DOE-Headquarters (DOE-HQ) Low-Level Waste Disposal Facility Federal Review Group (LFRG) for review and approval in November 1997 (WSRC 1997). The 1997 SRS CA analysis calculated potential releases to the environment from all sources of residual radioactive material expected to remain in the General Separations Area (GSA). The GSA is the central part of SRS, and it contains the two low-level radioactive waste disposal facilities along with chemical separations facilities and associated radioactive liquid waste tank farms as well as numerous other sources of radioactive material.

The LFRG CA review team prepared and issued in June, 1998, a report (DOE 1998) that documented their review of the 1997 SRS CA. Based upon the LFRG review, the DOE provided conditional approval of the 1997 SRS CA in January 1999 (Fiore and Frei 1999a). One of the major LFRG concerns expressed in the conditional approval was that sources of residual radioactive material from the entire SRS and not just from the GSA alone should be considered. As a first step toward considering the entire SRS, one of the major LFRG conditions was that SRS should issue an addendum to the CA, which revised the CA to include all source terms within the Upper Three Runs (UTR) watershed (i.e., add sources in A and M Areas to those in the GSA). To address this requirement for approval, SRS issued the CA Addendum, Revision 0 (WSRC 1999) in September 1999 and the CA Addendum, Revision 1 (WSRC 2002) in March 2002. The LFRG stated that the next revision of the SRS CA would need to consider the entire site. The review and approval process for the 1997 SRS CA resulted in a number of other commitments for the next revision. These are documented in the LFRG Review Team report (DOE 1998) and the CA Addenda (WSRC 1999 and WSRC 2002) and have been tracked in the PA/CA Maintenance Plan that is prepared each year (SRNS 2009a; latest version of the plan).

In July 1999 the DOE issued DOE Order 435.1, "Radioactive Waste Management" (DOE 1999a), along with the companion Manual (DOE 1999b) and Guide (DOE 1999c). The Order, Manual, and Guide include the requirement for a CA along with guidance for production of a CA associated with active LLW disposal facilities and the closure of high-level waste (HLW) tanks. Subsequent to issuing the order, the DOE issued the following supplemental documents which provide additional guidance concerning the preparation of CAs:

- "Format and Content Guide for U.S. Department of Energy Low-Level Waste Disposal Facility Performance Assessments and Composite Analyses" (DOE 1999d)
- "Transuranic Waste Disposal Facility Federal Review Group Manual" (DOE 2001)
- "Low-Level Waste Disposal Facility Federal Review Group Manual" (DOE 2008)

DOE 2001 and DOE 2008 make it clear that the DOE Order 435.1 requirement for a CA is also applicable to transuranic (TRU) waste disposals at facilities other than the Waste Isolation Pilot Plant (WIPP). See Section 2.0 of Phifer and Cook 2007 for the specific requirements for a CA outlined by DOE Order 435.1 and its clarifying documents for active LLW disposal facilities, the closure of HLW tanks, and the disposal of TRU waste. Based upon the DOE Order 435.1 requirement for a CA associated with the closure of HLW tanks, the LFRG has indicated that the next revision of the CA would need to address more directly the CA requirements for closure of HLW tanks.

In September 1999, DOE-HQ issued a disposal authorization statement (DAS) for the ELLWF and Z-Area SDF (Fiore and Frei 1999b), in part based upon the 1997 CA. The DAS included conditions for approval, including meeting the commitments made to the LFRG during the 1997 SRS CA review and approval process. The DAS serves as the "license" for DOE LLW disposal as per DOE Order 435.1 (DOE 1999a).

An accurate and up-to-date CA is a DOE Order 435.1 (DOE 1999a, 1999b, 1999c) requirement for continued DOE-HQ authorization to dispose LLW within the ELLWF and SDF, for closure of the F and H-Area radioactive liquid waste storage tanks, for potential disposal of TRU Pad 1 waste in-place, and for potential in-situ disposal (ISD) of TRU material within the F-Area Material Storage Facility (FAMS) at its deactivation and decommissioning (D&D). Due to the significant changes that have occurred since issuance of the initial SRS CA in 1997, many of which are listed in the next section, a revised SRS CA is necessary in order to support the LLW DAS, tank closure, and potential on-site TRU disposal.

2.2 MAJOR CHANGES NECESSITATING A COMPOSITE ANALYSIS REVISION

Many changes necessitating a CA revision have occurred, since issuance of the initial SRS CA in 1997, including but not limited to:

- DOE Order 435.1, requiring and providing guidance for a CA, was issued after the initial SRS CA.
- Commitments were made to the LFRG including consideration of all sources of potential residual radioactive material at SRS and more complete inclusion of the F- and H-Area tank closures (other commitments to the LFRG are listed in DOE 1998, WSRC 1999, WSRC 2002, and SRNS 2009a).
- Substantial operational and design changes associated with the ELLWF, SDF, and F and H-Area Tank Farms (FTF and HTF) have occurred.
- SRS potentially plans the disposal of a relatively small volume of TRU waste currently stored on Pad 1 in the E-Area TRU waste facilities.
- SRS potentially plans the ISD of a relatively small volume of TRU material within FAMS (i.e., after D&D of the facility is complete a small volume of TRU material may potentially be left within FAMS).
- Many Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) and Resource Conservation and Recovery Act (RCRA) remedial actions and D&D of facilities have occurred.
- Modifications to planned CERCLA and RCRA remedial actions and facility D&D have been made.
- New facility construction, such as the Mixed Oxide (MOX) Fuel Fabrication Facility and Salt Waste Processing Facility (SWPF), has begun.

2.3 MAJOR REVISIONS TO 1997 COMPOSITE ANALYSIS

Table 2-1 provides a listing of the major revisions to the 1997 original CA made in this CA revision. The major revisions include an expansion of the facilities supported by the CA, consideration of the entire SRS site, an expansion of the Points of Assessment (POAs) and exposure scenarios considered, updated inventories and input data, use of a unified CA model, and inclusion of uncertainty (probabilistic) analyses.

Table 2-1. Major Revisions to the 1997 CA

1997 Original CA	2009 CA Revision
The original CA was performed to support LLW disposal within the ELLWF and SDF.	The revised CA was performed to support LLW disposal within the ELLWF and SDF, closure of the F and H-Area radioactive liquid waste storage tanks, potential disposal of TRU Pad 1 waste in-place, and for potential ISD of TRU material within the FAMS at its D&D.
Considered GSA portion of SRS	Considered entire SRS
SRS land use planning based upon: • Savannah River Future Use Project Report (DOE 1996c)	SRS land use planning based upon: • Savannah River Site Comprehensive Plan/ Ten Year Site Plan (SRNS 2009b) • Savannah River Site End State Vision (DOE 2005)
POAs included UTR, Fourmile Branch (FMB), and the Savannah River	Expanded the POAs to include UTR, FMB, Steel Creek (SC), Lower Three Runs (LTR), and the Savannah River
Base case exposure scenarios: • Recreation in mouth of UTR and FMB • Recreation plus drinking water in Savannah River at U.S. Highway 301 Bridge	Base case exposure scenarios: • Recreation in mouth of UTR, FMB, SC, and LTR plus residential with Savannah River water • Recreation in and residential with Savannah River at U.S. Highway 301 Bridge
Inventory for projected end state source locations within the GSA portion of SRS (CDM 1996, CDM 1997)	Revised inventory based upon most up-to-date data available for projected end state source locations within the entire SRS (Hiergesell et al. 2008)
Transport of radionuclides to the POA and calculation of dose was performed utilizing the following tools: • The PATHRAE code was used to estimate radionuclide transport from the source and through the vadose zone, and the radionuclide flux to the water table. • The PORFLOW code was used to estimate radionuclide transport through the saturated zone and the flux to the streams. • LADTAP XL code was used to estimate stream concentrations and doses.	Radionuclide transport from the source, through the vadose and saturated zones, radionuclide flux to the streams, radionuclide stream concentrations, and doses were all estimated using a unified CA model developed with the commercial GoldSim™ software Version 9.60 SP4 (Service Pack 4) (GTG 2007)
Base case and sensitivity analyses were performed.	Base case, sensitivity, and uncertainty (probabilistic) analyses were performed.

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3.0 BACKGROUND INFORMATION

This section of the Composite Analysis (CA) addresses the requirements of CA Tasks 4.2.4, 4.2.5, and 4.2.6 in the Comprehensive Savannah River Site (SRS) Composite Analysis Program Plan (Phifer and Cook 2007). The objectives of these tasks are to provide an SRS site description that is sufficiently detailed to support the development of the conceptual modeling carried out in the CA (see Sections 3.1 through 3.3 below); to summarize existing information on SRS facilities for which Performance Assessments (PAs) have, or are, being developed in the General Separations Area (GSA) in support of conceptual model development (see Section 3.4 below); and to provide an overview of related documentation that influences the CA (see Section 3.5 below). Sections 3.1 through 3.4 were taken largely from Phifer et al. (2008).

3.1 SRS FACILITY DESCRIPTIONS AND OPERATIONAL HISTORY

Construction of and subsequent operations at the SRS began in 1951 under the direction of the Atomic Energy Commission (AEC) (DOE 1997a). The primary mission of the SRS was to produce tritium and plutonium for the national nuclear weapons complex, although many diverse missions rapidly developed in the early years and throughout its history (Reed et al. 2002). The primary SRS mission was accomplished through extraction and purification of heavy water, fabrication of fuel and target assemblies, reactor operations, chemical separation processes, tritium extraction and purification, and waste management operations. The SRS produced about 36 metric tons of plutonium from 1953 to 1988 (SRS 2008a). In addition to tritium and plutonium, Co60, Po210, Pu238, Cm244, Cf252, and other isotopes were produced for non-weapons purposes (Reed et al. 2002). Following the end of the Cold War, the mission of the SRS changed to stewardship of the nation's nuclear weapons stockpile, nuclear materials, and the environment (Blake et al. 2005a, Mamatey 2006).

In support of its primary mission, SRS brought five reactors into operation between December 1953 and March 1955 (DOE 1997a), along with support facilities, including two chemical separations plants, a heavy water extraction plant, a nuclear fuel and target fabrication facility, a tritium extraction facility and waste management facilities (SRS 2008a). The Heavy Water Plant began operation in August of 1952 (Reed et al. 2002). In 1954, the first of the two chemical separations areas, both of which allowed recovery of Np237, Pu238, U238, and Pu239, began operation and the first delivery of Pu to the AEC occurred in 1954. As early as 1953, the first burial ground for solid waste was used, while the first radioactive liquid waste storage tank was filled to capacity in June of 1955 (Reed et al. 2002).

Even in the 1950s, the SRS reactors were producing radioisotopes for non-weapons uses. Before the reactors were shut down, over 100 different radioisotopes had been made at the site. Much of the non-military isotope production was carried out between 1964 and 1970, and focused on isotopes for use as heat sources and transplutonium isotopes for use in industry and medicine (Reed et al. 2002).

The SRS has had an extensive environmental monitoring program in place since 1951, covering both on- and off-site locations (Blake et al. 2005a). This early monitoring established an environmental baseline for the site. By early 1953, the natural radiation in a 1500-square-mi area was determined and the requirements for future monitoring were in place. Thus, stewardship of the site began essentially at its inception. Early on, distances between individual facilities, and distances to the plant perimeter were carefully selected to maximize public safety with respect to possible releases of radioactivity to the atmosphere and streams. The Savannah River Ecology Laboratory (SREL) was established as a permanent onsite facility in 1961, with a research program mostly directed toward basic ecological studies. Thermal ecology became a major program of research in the early 1970s, studying the effects of thermal water from the site's reactors on local ecosystems (Reed et al. 2002).

In 1972, the entire SRS was designated as the nation's first National Environmental Research Park by the AEC, which established the site as a place in which ecologists, engineers, and land managers could work collaboratively to understand how activities and energy technologies affect the natural environment. By 1980, a five-year Restoration Program for process facilities on the site was launched. In 1981, chlorocarbon groundwater contamination was discovered at the M-Area Settling Basin, and an M-Area groundwater cleanup program was initiated that resulted in operation of a full-scale groundwater treatment system by 1985 (Reed et al. 2002). In 1991, the Mixed Waste Management Facility (MWMF) was the first SRS facility closed and certified under the Resource Conservation and Recovery Act (RCRA) (Reed et al. 2002). Since then, extensive remediation has been carried out on the site (Mamatey 2007), with a scheduled completion date for all SRS cleanup of 2031 (SRS 2006).

With the declining need for a large nuclear weapons stockpile, many SRS facilities no longer produce or process nuclear materials. All reactors were shut down by 1988. However, the SRS continues as the Department of Energy's (DOE) center for the supply of tritium to the nuclear weapons stockpile with the startup of the 2006 Tritium Extraction Facility (TEF), which extracts tritium from rods irradiated in commercial reactors. Operations at the SRS K-Area currently provide interim safe storage for much of DOE's excess Pu, in a building which formerly housed K Reactor. New Pu facilities are presently being built at the SRS (SRS 2008a).

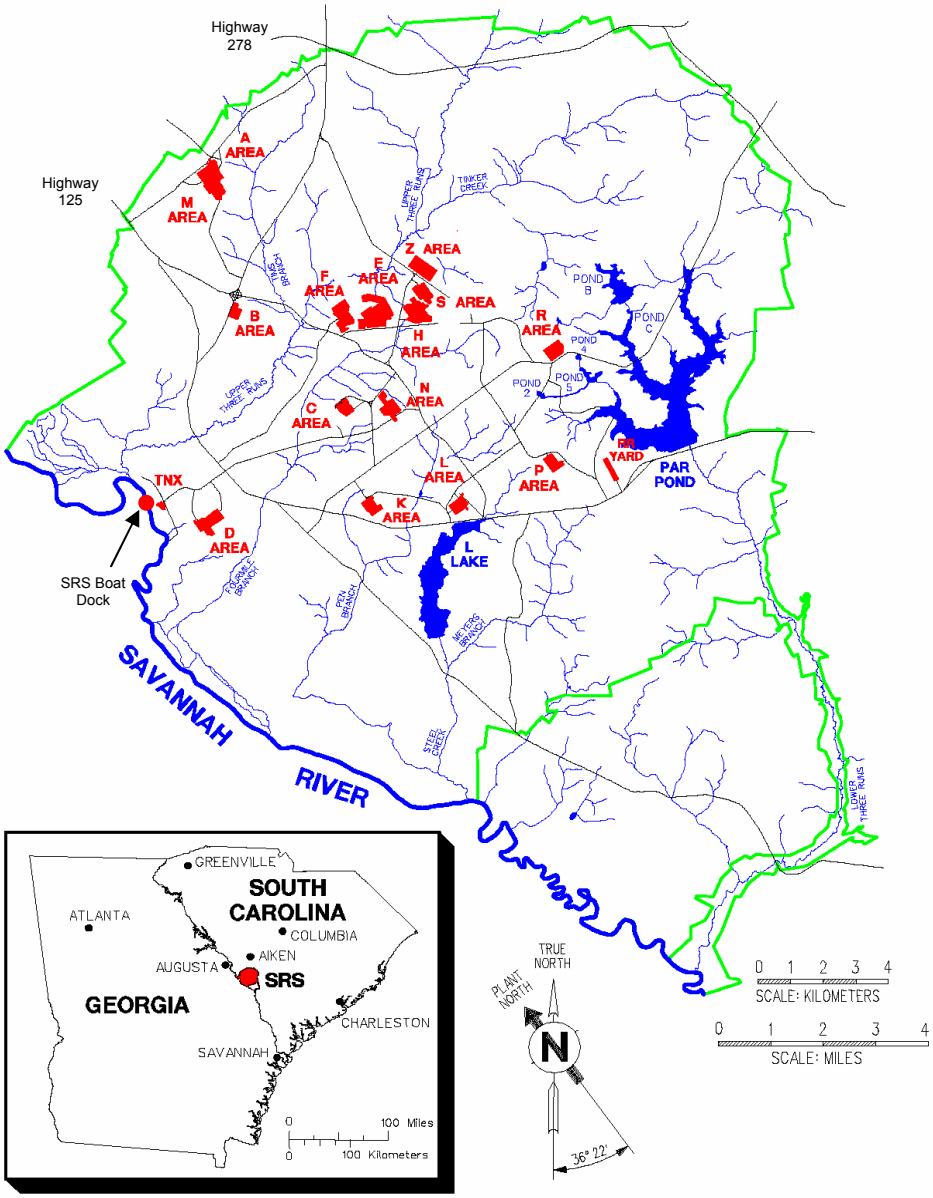
As the SRS mission has changed, the many surplus facilities must be dispositioned safely and economically. In 2002, SRS began extensive decommissioning activities in D-Area, M-Area, and T-Area (also known as CMX/TNX or TNX). Site Deactivation and Decommissioning (D&D) continued extensive operations through 2006. A total of 247 facilities were removed by the end of 2006, representing a footprint reduction of 2.5 million square feet. TNX completion was achieved in 2006, while M-Area completion activities continued and D-Area completion began (Mamatey 2007).

Activities associated with the SRS missions have been, and are, located in several developed areas around the site (Blake et al. 2005a). At present, the SRS consists of 16 areas (Figure 3-1), which can be categorized into the following nine major categories of past, current, and planned future operations:

- Heavy water production and reprocessing area (D-Area)
- Reactor materials area (M-Area)
- Reactor areas (C-, K-, L-, P-, R-Areas, and U-Area)
- Separations and tritium areas (F- and H-Areas)
- Waste management areas (E-, F-, H-, N-, S-, and Z-Areas as well as seepage basins in most of the other major areas)
- Savannah River National Laboratory (SRNL) and CMX/TNX (A-Area and T-Area)
- Administration and support areas (A-, B-, and N-Areas [Central Shops])
- Other radioactive missions areas (A-, C-, F-, H-, and K-Areas)
- Future missions/facilities (F- and S-Area)

U-Area is not shown in Figure 3-1 but was contiguous with B-Area, and was the location of the Heavy Water Components Test Reactor (HWCTR) which operated between 1961 and 1964 (DOE 1997a).

The following discussion summarizes important historical and geographical aspects of each of these operational areas, beginning with heavy water production, which began in D-Area in 1952, in anticipation of the upcoming reactor operations. The focus is on operations which involve the use, production or disposal of radionuclides, consistent with the focus of the CA. Thus, there are some non-radiological activities or related land uses on the SRS that are not discussed below, including a number of sanitary and construction and debris landfills that are present on the site.



Note: During design of the SRS facilities in the 1950's, an SRS site grid system was established for facility location, based upon a Plant North which was rotated 36°22′ west from True North. This connotation applies to all figures contained within this CA that are based upon Plant North.

Figure 3-1. SRS Operational Area Location Map

3.1.1 Heavy Water Production and Reprocessing Area (D-Area)

The heavy water production area, D-Area (Figure 3-1), operated from 1952 until 1982 (SRS 2006; Reed and Swanson 2008), and occupies 445 acres near the west boundary of the SRS, near the Savannah River. The nearest site boundary, about 1 mi west of D-Area, is the Savannah River (WSRC 2007a). An operating coal-fired power plant is also located at D-Area.

D-Area originally consisted of a heavy water production plant, a moderator rework facility and an analytical laboratory (WSRC 2007a). In the production plant, a hydrogen sulfide-water exchange process was used to partially enrich heavy water, followed by fractional distillation and electrolysis to further concentrate deuterium oxide (DOE 1997a). The moderator rework facility used fractional distillation to re-enrich reactor moderator (DOE 1997a). The facility began operations in 1952 and was shut down in 1982 because of a sufficient supply of heavy water (Reed et al. 2002, WSRC 2007a). Both processing/reprocessing of heavy water and operations of the coal-fired power plant have resulted in contaminated soil, surface water, sediment, and groundwater in D-Area (WSRC 2003a). The SRS began extensive decommissioning of D-Area in 2002, and the planned D&D was completed in 2006 (Mamatey 2007).

3.1.2 Reactor Materials Area (M-Area)

The Reactor Materials area (also known as the Raw Materials area), or M-Area (Figure 3-1), was used to provide support to the reactor facilities, heavy water facilities, and the fuel fabrication facilities. M-Area is located near the north boundary of the SRS, immediately adjacent to the SRNL, and 0.8 mi southeast of the nearest site boundary. Operations at M-Area have ceased, and the area is undergoing decontamination and decommissioning, with most buildings having already been demolished and removed (WSRC 2007a).

M-Area went into production in 1952 (Reed et al. 2002; Reed and Swanson 2006a), manufacturing the fuel and target assemblies used in the five SRS production reactors. The area included three large buildings in which fuel elements were fabricated, two laboratories, a wastewater treatment plant, a waste vitrification facility, and several support facilities including warehouses and offices (SRS 2006). Prior to 1958, process effluent was routed to a tributary of Tims Branch (Lewis and Aadland 1994). The M-Area Hazardous Waste Management Facility (HWMF) was constructed in 1958, operated from 1958 to 1985, and includes the M-Area Settling Basin and associated areas (Lewis and Aadland 1994). A network of process sewer lines consisting of vitrified clay pipes, now referred to as M-Area Inactive Process Sewer Lines (MIPSL) transported M-Area facility effluents to the Settling Basin and an outfall which flowed to a tributary of Tims Branch (WSRC 2006a). All operations were shut down in M-Area by the late 1980s. By the end of 2006, D&D was complete for all M-Area buildings (Mamatey 2007).

3.1.3 Reactor Areas (C-, K-, L-, P-, and R- Areas and Heavy Water Components Test Reactor)

As noted, the five production reactors (C, K, L, P, and R reactors) were built and began operating between December of 1953 and March of 1955 (DOE 1997a). The reactor areas (area names correspond to names of reactors) are each approximately 5 to 6 mi from the nearest plant boundary (Figure 3-1, WSRC 2007a). These reactors were cooled and moderated with heavy water flowing in a closed loop system at low temperatures and pressures. Cooling water drawn from the Savannah River removed the heat generated in the reactors before being returned to the river via surface streams. Two large artificial lakes at the site, PAR Pond (created in 1958) and L Lake (created in 1985), acted as coolant reservoirs (Wike et al. 2006). None of the reactors is currently operating, having been shut down in late 1988 (with the exception of R reactor, which was shut down in 1964) (DOE 1997a). Significant amounts of moderator and radioactive materials are currently stored in some of the reactor facilities (WSRC 2007a).

The SRS production reactors were designed and constructed by DuPont. The reactors initially used a combined fuel and target assembly, consisting of aluminum-clad natural uranium. In 1968 the reactors converted to highly-enriched uranium fuel assemblies and separate depleted uranium target assemblies (DOE 1997a). The fuel and target assemblies were loaded and unloaded into the reactor vessel remotely. Each production reactor building housed an assembly area for storage and assembly of fresh fuel and targets; a disassembly area, consisting of a large pool of water for storage of irradiated fuel and targets and disassembly for transport to the separations plants, and preparation for transfer; and a purification area, for heavy water treatment and purification. These buildings were equipped with filtered ventilation systems to mitigate potential airborne radioactivity releases (DOE 1997a).

Releases of radioactive coolant were not routine because of the closed-loop cooling systems used in the SRS reactors. However, unplanned leaks or discharges allowed radioactive water to escape. During the first years of operation, reactor cooling water and disassembly basin effluents were released directly to Steel Creek, Lower Three Runs (LTR), and the Pen Branch stream. After 1958, discharges to PAR Pond (and later to L Lake) allowed reactor effluent to cool before leaving the site. Sediments in PAR Pond are contaminated with Cs137 and transuranics (TRUs) as a result of reactor discharges in the late 1950s and early 1960s (DOE 1997a).

In addition to reactor coolant, reactor operations generated many liquid wastes, including contaminated discharge and disassembly basin water and liquids used for equipment decontamination. In 1957, seepage basins in the Reactor Areas began to receive low-level radioactive water from the purge water in the fuel disassembly basins (DOE 1997a).

The SRS stored thousands of irradiated targets in the K, L, and P Reactor disassembly basins for several years after shutdown. To stabilize the corroding material, the targets were dissolved and processed in 1996 (DOE 1997a). Spent nuclear fuel from the production reactors, and from domestic and foreign research reactor programs, is currently stored in L-Area's disassembly basin, awaiting final disposition (SRS 2008a).

The HWCTR was located in an area formerly called U-Area, which is contiguous to B-Area (Figure 3-1). It was used to test various fuel and target assemblies between 1961 and 1964 (DOE 1997a). The support buildings and structures have been demolished, and the only structure remaining is the reactor building, a steel containment structure that has been deactivated and welded shut, placing the facility into safe storage pending demolition (SRS 2006). Some of the welds were cut in 2009 to open the facility in preparation for demolition.

3.1.4 Separations Areas and Tritium (F- and H-Areas)

The Separations Areas (F- and H-Areas) are located near the geographic center of the SRS, and occupy 364 and 395 acres, respectively (Figure 3-1). Nuclear materials were processed and purified in these two areas after leaving the reactors. The chemical separations plants were initially built to recover plutonium from the five production reactors and to produce tritium. To accommodate the changing missions of the SRS, the Separations Areas also processed nuclear materials for the heat source and transplutonium programs, requiring the development of new techniques and facilities. Facilities were also built in H-Area to receive, store, and process offsite reactor elements (Reed et al. 2002).

3.1.4.1 Liquid Separations Processes

Liquid separations involved liquid extractions to purify plutonium, uranium and other elements (Reed et al. 2002). F Canyon and H Canyon are the two areas largely devoted to wet separation processes. Wet processes involved dissolving reactor fuel and target assemblies, and chemically treating the resulting solution to separate unwanted from wanted material. A chemical process developed for use at the SRS, called PUREX, has become the standard for Pu and U separation. A variation on the PUREX process, called HM, was used in processing enriched uranium (Reed et al. 2002). The process buildings were designed to allow work to be performed remotely. Waste from the wet separations process was transferred to the radioactive liquid waste storage tanks at both F- and H-Area.

F Canyon was constructed in the early 1950s, and began operation in 1954 (SRS 2008c). The building that housed F Canyon also housed FB Line, and together, these were the two main processing facilities at F-Area (WSRC 2007a). F Canyon and FB Line were used primarily to recover Pu239 from reactor target assemblies and depleted U238 was recovered as a by-product. The production mission of FB Line and F Canyon was completed in March 2002, although stabilization and packaging of legacy nuclear materials for long-term storage was carried out until 2005 (SRS 2008c, 2008d). Currently, F-Area is undergoing decommissioning. F-Area support facilities outside the F Canyon Building include the U Processing Facility (FA line), the F-Area Materials Storage (FAMS) facility (See Section 3.4.6 for additional FAMS information), a Naval Fuels Fabrication Facility, Central Laboratories, a Mock-up/Fabrication Facility, and the F-Area Tank Farm (FTF) (SRS 2006).

H Canyon came online in July of 1955 (Reed et al. 2002), and similar to F Canyon, housed HB Line in the same building. Historically H Canyon and HB Line were used primarily to recover U235, Np237, and Pu238 from reactor assemblies (Reed et al. 2002; SRS 2008a). More recently H Canyon has been used to convert weapons-grade highly enriched uranium (HEU) to low-enriched material suitable as fuel in commercial power reactors; disposition research reactor spend nuclear fuel; and process and stabilize scrap Pu (WSRC 2007a; SRS 2008e). HB Line has three current process lines: a line to recycle legacy Pu scrap for purification and concentration to a solid form; a line to produce solid oxide material from Np237 or Pu239 nitrate solutions; and a line to produce Pu238 oxide from nitrate solutions. H Canyon and HB Line remain in operation at the SRS (WSRC 2007a). H-Area facilities outside the H Canyon Building include the Tritium Facilities, the H-Area Tank Farm (HTF), the Receiving Basin for Offsite Fuels (RBOF), the Effluent Treatment Project (ETP) facilities, the Consolidated Incineration Facility (CIF), and various other chemical, industrial, administrative, laboratory, and storage facilities (SRS 2006, WSRC 2007a).

F-Area and H-Area Inactive Process Sewer Lines (FIPSL and HIPSL) extend from Building 221-F and H-Canyon to their respective seepage basins in F- and H-Area. The FIPSL and HIPSL were vitrified clay lines utilized from 1955 to 1982 to transport hazardous and low-level radioactive wastes from Separation, Tritium, and Waste Management facilities in F- and H-Area. Due to deterioration and leakage, the vitrified clay lines were abandoned in place and replaced with a set of high-density polyethylene (HDPE) lines (WSRC 1993a).

3.1.4.2 Tritium Processing

The original SRS tritium processing facility, involving a gas separations process for tritium purification, was located in F-Area and began operation in 1955 (Reed et al. 2002). Irradiated lithium-aluminum assemblies were shipped in shielded casks from the onsite reactors to F-Area. The tritium extraction process included gas extraction from the assemblies, primary separation of hydrogen from the gas, stripping of remaining hydrogen from the waste gases, isotope separation, and packaging in containers for shipment offsite. In 1958, all tritium production was shifted to the larger facility in H-Area, which began operation in 1956. The H-Area tritium facility's packaging facility allowed tritium to be loaded in the actual weapons components, rather than shipped to another site for that procedure (Reed et al. 2002). The F-Area tritium facility sat idle until being decommissioned between 1994 and 1996 (DOE 1997a). Since shut down of the reactors, the SRS did not produce new tritium until 2007 (SRS 2008a).

The present day SRS Tritium Facilities, located in H-Area, consist of three main active process buildings that house operations to reclaim previously used tritium reservoirs; to receive, package and ship such reservoirs; and to recycle, extract and enrich tritium gas. One building was built in the late 1960s; the second, called the H Area New Manufacturing Facility (HANM), began operations in 1994; and the third is the TEF, which became fully operational in early 2007 (SRS 2008e). The HANM is the reservoir loading and unloading facility. In the TEF, tritium is extracted from irradiated Tritium Producing Burnable Absorber Rods (TPBARs) received from commercial reactors (SRS 2008a) and transferred via underground piping to the Tritium Loading Facility.

3.1.5 Waste Management (E-, F-, H-, N-, S-, and Z-Area)

The types of waste generated and managed at the SRS have included both liquid and solid wastes, which can be broadly classified as high-level, low-level, TRU wastes, and mixed waste. Both by volume and radioactivity, the majority of waste generated is liquid, and is stored in large tanks in the FTF and HTF of the Separations Areas. Approximately 36 million gallons of radioactive liquid waste are stored in the F-Area and H-Area underground tanks (SRS 2008a). Solid low-level, mixed, and TRU wastes have been disposed and stored in the site burial grounds in E-Area, with some mixed waste storage in H- and N-Areas, and are generally characterized by long-lived radionuclides. Very low-level liquid waste was sent to seepage basins in the early years of operations until the early 1970s, when some seepage basins were replaced by lined retention basins (Reed et al. 2002).

The management of wastes has evolved throughout the history of the SRS, with the level of isolation, segregation, and treatment sophistication increasing through the years. Provided below is a summary of the facility and operational history of the E-Area Low-level Waste (LLW) Burial Grounds; the mixed waste storage facilities in E-, H-, and N-Area; the TRU waste pads in E-area; the FTF and HTF and LLW processing facilities; the S-Area Defense Waste Processing Facility (DWPF); the Z-Area Saltstone Facilities; and the site seepage basins.

3.1.5.1 Low-level Waste Burial Grounds (E-Area)

The E-Area LLW Burial Grounds is comprised of the formerly-utilized burial grounds, and the E-Area Low Level Waste Facility (ELLWF) (SRS 2006). The formerly-utilized burial grounds consist of two sections: the south 76-acre section, termed the Old Radioactive Waste Burial Grounds (ORWBG), or 643-E; and the north 119-acre section, called the Burial Ground Expansion (BGE), or 643-7E (ERDA 1977, WSRC 1999a).

The E-Area LLW Burial Grounds is located between the F- and H-Areas, and near the approximate center of the SRS (Figure 3-1). It is approximately 6.5 mi east of the nearest site boundary. The E-Area LLW Burial Grounds is used for disposal and storage of radioactive, hazardous, and mixed solid waste generated at the SRS, as well as occasional special shipments from offsite. It also provides assaying, repackaging, and storage of TRU waste (WSRC 2007a). In 2001, the SRS began shipping its TRU waste, about 28,000 legacy drums, to the Waste Isolation Pilot Plant (WIPP) in New Mexico. At the end of 2007, over 24,000 drums had been shipped, and the current projections have the site scheduled to ship all of the SRS TRU drummed waste to WIPP by 2010 (SRS 2008a).

The ORWBG received radioactive waste from 1952 to 1972, with a small additional quantity of waste being disposed in 1974. The ORWBG contains over 600,000 curies of radioactive materials and over 93 tons of hazardous waste. The area also contains 22 underground tanks, formerly used to store radioactive solvents. More than seven million cubic feet of radioactive wastes were buried at the ORWBG. As part of an Interim Action RCRA/CERCLA ROD, SRS installed a soil cover over a majority of the ORWBG to reduce surface radiation levels in 1998. As part of a Final RCRA/CERCLA ROD, SRS installed a final geosynthetic closure cap over the ORWBG to reduce storm water infiltration to the waste layer and mitigate contaminant migration to the groundwater in November 2007 (SRS 2007a).

A 58-acre portion of the BGE (the north 119-acre section of the formerly-utilized burial grounds), is termed the MWMF, or 643-28E. The MWMF was carved out of 643-7E, and was closed in 1990 after receiving mixed waste from the early 1970's to 1986. Some of the wastes disposed in the MWMF were uncontained, some were encapsulated in concrete, and some were contained in concrete or steel boxes. Closure consisted of dynamically compacting the wastes, and placing a low-permeability cap to minimize rainwater infiltration, thus minimizing contaminant migration into the subsurface (Bullington and Frye-O'Bryant 1993). An additional 18 acres of the BGE (643-7E), designated as the Low Level Radioactive Waste Disposal Facility (LLRWDF), received waste from 1986 to 1995. Thirteen of the 18 acres were also determined to have received mixed waste from 1986 to 1990. These thirteen acres were carved out of the LLRWDF, added to the MWMF (643-28E), and closed in 1998. The remaining five acres of the LLRWDF continued to receive LLW for disposal until 1995. The 1998 closure also covered these five acres, which received LLW for disposal between 1990 and 1995. The remaining 43 acres of the BGE (643-7E) is used to store TRU waste and mixed waste prior to shipment to off-site disposal facilities (Bullington and Frye-O'Bryant 1993, ERDA 1977, WSRC 1991, WSRC 1999a, Phifer and Serrato 2000, WSRC 2003b).

Radiological operations at the ELLWF began in 1994. The ELLWF comprises 200 acres (100 acres currently allocated for waste disposal and an adjacent 100 acres potentially available for waste disposal) and a surrounding buffer zone that extends out to the 100-m point of compliance. The current ELLWF area developed for disposal consists of approximately 100 acres. It is an elbow-shaped, cleared area, which curves to the northwest, situated immediately north of the MWMF and LLRWDF. Disposal units within the footprint of the ELLWF include the Slit Trenches (ST), Engineered Trenches (ET), Components-in-Grout (CIG) Trenches, the Low-Activity Waste Vault (LAWV), the Intermediate-Level Vault (ILV), and the Naval Reactor Components Disposal Areas (NRCDAs). The current ELLWF also includes a buffer zone surrounding the 100-acre disposal area, which extends out to the 100-m point of compliance for all disposal units (WSRC 2008a).

The ST, ET and CIG Trenches are below-grade earthen disposal units, accepting LLW consisting of soil, debris, rubble, wood, concrete, equipment, and job control waste. The CIG Trenches are designed to grout-encapsulate large radioactively contaminated equipment and other smaller wasteforms such as B-25 boxes to fill in the space around and above the large equipment (WSRC 2008a).

The LAWV is an above grade, reinforced concrete vault, which is designed to contain predominately B-25 boxes, B-12 boxes, drums and/or concrete containers. The ILV is a below-grade, reinforced concrete vault built to accommodate intermediate-activity waste including tritium crucibles, job control waste, scrap hardware, and contaminated soil and rubble. The NRCDAs are above-grade gravel pads for the disposal of Naval Reactor Waste Shipping/Disposal Casks containing NR components (WSRC 2008a).

Final closure of the ST, ET, CIG Trenches, LAWV, ILV and NRCDAs will occur during final closure of the entire ELLWF at the end of the 100-year institutional control period, following the end of the ELLWF operations. Final closure of the ELLWF will consist of the installation of an integrated closure system designed to minimize moisture contact with the waste and to provide an intruder deterrent. The integrated closure system will consist of one or more closure caps installed over all the disposal units and a drainage system (WSRC 2008a). See Section 3.4.1 for additional information concerning the ELLWF.

3.1.5.2 Mixed Waste Storage (E-, H-, and N-Areas)

Mixed LLW (hereafter referred to as “mixed waste”) is LLW that also contains a hazardous component subject to the RCRA or the Toxic Substances Control Act (TSCA). Therefore, mixed waste is managed at the SRS in accordance with the requirements of RCRA, TSCA and DOE Order 435.1-1 (DOE 1999a). Currently, mixed waste is only stored, not disposed, at the SRS. The final disposition of all mixed waste is that it will be permanently disposed offsite at a commercial facility (DOE 2005).

Early site practices dispositioned some mixed waste in the MWMF. This facility, located in E Area, was closed in 1990 under RCRA requirements and is now under postclosure care. Presently, new mixed waste is stored onsite for less than one year per RCRA before being shipped offsite for disposal via commercial vendors. Mixed waste is primarily containerized in 55-gallon drums or B-25 boxes and stored in RCRA-permitted facilities in E-Area and the N-Area Hazardous/Mixed Waste Facilities (DOE 2005, WSRC 1994a). In addition to these facilities, underground solvent storage tanks in H-Area periodically receive radiologically-contaminated organic solvent waste. Historically, building 710-B, in B-Area, stored both hazardous and mixed waste, but this mixed waste storage is currently precluded in this building (WSRC 1994a).

The Mixed Waste Storage Buildings, 643-29E and 643-43E, are at-grade structures located in E-Area. Building 643-29E became operational during the late 1980s and Building 643-43E in 1995. Both buildings are designed to store waste that contains free liquids and containerized solid waste, and have been used to provide interim storage of mixed waste in containers as specified in the current RCRA Permit (WSRC 2007b). Building 643-29E continues to be used for storing mixed waste, but Building 643-43E is no longer used for this purpose.

Three at-grade Hazardous Waste/Mixed Waste Storage Buildings (645-N, 645-2N, and 645-4N) and three associated at-grade asphalt waste storage pads are located within N-Area. The vented metal buildings are segregated into one or more cells (or bays), providing interim storage of containerized waste as specified in the current RCRA Permit. Concrete curbs, dikes, and collection sumps are utilized to provide liquid containment (WSRC 2007b). Only solid containerized hazardous waste and mixed waste (free liquids are prohibited by the RCRA permit) are stored on the outside storage pads. These storage pads, which are partially protected by weather shelters, provide interim storage of the containerized waste (WSRC 2007b).

The four H-Area solvent storage tanks (607-33H through 607-36H) periodically receive mixed waste from the separations facilities. The four tanks are underground, double-walled, carbon steel tanks, approximately 42 ft in length with a 12-ft diameter. The nominal storage capacity of each tank is 30,000 gal, and each vented tank has secondary liquid containment, leak detection and leak collection sumps, and overfill protection (WSRC 2007b).

3.1.5.3 Transuranic Waste Pads (E-Area)

Since the 1970s, TRU waste has been retrievably stored at the SRS for eventual disposal at the WIPP in Carlsbad, New Mexico. The waste is contained in steel boxes, steel drums, and concrete containers. Containers are either stored directly on a TRU pad or inside an overpack container which is stored on the pad. The primary overpack container is the concrete culvert, which can store up to fourteen 55-gallon drums. Drums that are not stored in concrete culverts were stored on weather-protected TRU pads (WSRC 2007b).

Pads 1-6 were constructed and began receiving waste during the early 1970's. As Pads 1-5 were filled, they were covered with a light sandy soil, a weatherproof tarp, overlain by a heavier loamy soil. This practice of mounding the pads ceased in 1985 prior to completing Pad 6 (Stone and Milner 2006).

Subsequent pads were not placed under earthen mounds and the 55-gallon TRU drums were fitted with filter vents. Pads 7 – 13 were open and pads 14 -19 were covered by weatherproof enclosure buildings. The uncovered drums on pads 7 through 13 accumulated water from rainfall, which penetrated the filter and accumulated in the drums. These drums were radiographed, dewatered and moved to covered storage in the early 1990s (Stone and Milner 2006).

TRU Pads 1 through 19 and 26 were at-grade concrete pads, and TRU Pads 23-25 were at-grade gravel pads. Concrete TRU pads stored radioactive and hazardous wastes; gravel TRU pads were used to store radioactive waste only. TRU Pads 1 through 19 were located in the BGE (643-7E). TRU Pads 23-26 were located in the ELLWF[1].

Prior to the first TRU waste shipment to WIPP, in March 2001, there were over 30,000 containers, comprising over 11,000 m^3, of TRU waste in storage at SRS. At the end of 2007, over 24,000 55-gal drums, or 5,000 m^3, of the original TRU waste inventory had been shipped. The remaining drummed waste is projected to be shipped to WIPP by 2010. Non-drummed waste, accounting for about 3,600 m^3, is projected to be repackaged and shipped by 2014 (SRS 2007b).

In preparing shipments to WIPP, the safe retrieval of almost 9,000 buried drums was completed in 1999. One important step in preparing the waste for shipment involves venting and purging the drums of hydrogen and other gases. All of the TRU waste storage containers with low-activity waste were vented by 1999. The project to vent the high-activity TRU waste drums was completed in 2008. Assaying the containers' contents is required prior to shipment, and repackaging is required for many drums. Finally, certified drums are placed in NRC-licensed transportation casks designed specifically for TRU waste transportation (SRS 2007b).

TRU Pad 1 contains TRU waste transferred from the Mound Facility and Los Alamos Scientific Laboratory which has a relatively high radioactivity content from the Pu238 present (Towler 1979). This waste consists of the residual material used in the production of power sources for the US space program. The TRU Pad 1 waste contains concentrations of Pu238 that exceed the regulatory definition of TRU waste (100 nCi/g), which means that it should be sent to WIPP for disposal (Cook 2005). However, the waste must be characterized and repackaged to meet transportation requirements, but there are worker safety concerns that need to be addressed in order to accomplish this. Although the plan remains to ship this waste after addressing the technical and safety issues, technical analysis in the form of a PA is currently underway to explore the possibility of disposing of this waste on the SRS, in accordance with United States Environmental Protection Agency (EPA) regulation 40 CFR Part 191 and DOE Order 435.1-1 (DOE 2005, DOE 1999a, Cook 2005). See Section 3.4.5 for additional information concerning TRU Pad 1.

[1] TRU Pad 24 was removed in 2009. All of the TRU waste on this pad was either shipped to WIPP or relocated to other pads.

3.1.5.4 Tank Farms and Low-level Waste Processing (F- and H-Areas)

In addition to the nuclear materials processing which took place in F- and H-Areas (the Separation Areas), waste management operations were performed in these two areas (SRS 2006). The two areas contain fifty-one underground carbon steel tanks, encased in concrete vaults, which store radioactive liquid waste arising from the separations processes. The waste present in the tank farms is both solid and liquid in form, in that insoluble solid chemicals are present along with water-soluble salts. The insoluble solids settle and accumulate on the bottom of the tanks as sludge. Liquid above the sludge is concentrated by evaporation to reduce its volume, resulting in a solid salt phase in some tanks (SRS 2007c).

The FTF is an active radioactive waste storage facility consisting of 22 carbon steel waste tanks and ancillary equipment such as transfer lines, evaporators and pump tanks (SRS 2007c). The FTF stores and processes liquid radioactive waste generated primarily from the PUREX process, and began radioactive operations in 1954. Three different types of tanks were designed and constructed for F-Area (Reed et al. 2002), with two types having secondary containment features. An evaporator system, which began operating in 1980, has reduced the supernate to about 30 percent of its original volume (SRS 2007c). Two of the 22 tanks (Tanks 17 and 20) were closed in 1997 by filling with grout after most of the contaminated waste and sludge were removed (WSRC 2008b). See Section 3.4.2 for additional information concerning the FTF.

The HTF consists of 29 large underground storage tanks, of four different designs, for radioactive liquid waste and evaporated saltcake. Three of the H-Area tank designs have secondary containment features. The one type of tank in H-Area that was not constructed for use in F-Area had leakage incidents. Two evaporators, one of which began operating in 1982, the other in 2000, are also part of the HTF operations (SRS 2007c). See Section 3.4.3 for additional information concerning the HTF.

The ETP at H-Area began treating low-level radioactive wastewater from the F- and H-Area separation facilities in 1988. The ETP processes about 20 million gallons of wastewater per year, and removes radioactive and nonradioactive contaminants, except tritium, from process effluents and allows the water to discharge to Upper Three Runs (UTR) (WSRC 2007a). The CIF at H-Area was constructed in 1993, incinerated SRS hazardous, mixed, and LLW, both solid and liquid, and is now shut down (WSRC 2007a).

3.1.5.5 Defense Waste Processing Facility (S-Area)

S-Area is located north of H-Area (Figure 3-1) and is the site of the DWPF Vitrification Plant and support facilities including two Glass Waste Storage Buildings (GWSBs) (SRS 2006). S-Area is approximately 7 mi south of the nearest site boundary (WSRC 2007a). The DWPF immobilizes sludge from the tank farms using a vitrification process and will also receive and vitrify the higher activity salt waste fraction from the Salt Waste Processing Facility (SWPF) and interim salt processing units (SRS 2007c).

Construction of the DWPF began in late 1983, and radioactive operations began in March 1996. The facility is projected to produce more than 5,000 canisters by the year 2019, in order to complete its waste vitrification mission (SRS 2007d).

The tank farm sludge feed to the DWPF comprises only about 10% of the volume in the radioactive liquid waste storage tanks, but about half of the radioactivity. The tank salt waste readily dissolves in water and comprises about 90% of the volume and the balance of the radioactivity in the tanks. Treatment of the tank salt waste began in 2008 through the interim salt processing units (Modular Caustic Side Solvent Extraction Unit (MCU) and Actinide Removal Process (ARP)) to remove fission products and actinides. The fission products and actinides, which are removed, are processed with the sludge in DWPF. The treated salt waste is sent to the Saltstone Processing Facility (SDF). It is anticipated that the SWPF will become operational in FY 2013, replacing the interim salt processing units (SRS 2007d; SRNS 2009a).

In the DWPF processing, molten glass-waste mixture is poured into 10-ft tall (2-ft wide) canisters, which are stored in the GWSBs. The GWSBs provide for underground storage in concrete, seismically-qualified vaults. The storage capacity in the two GWSBs is 4,400 canisters (SRS 2007d).

3.1.5.6 Saltstone Facility (Z-Area)

Z-Area is located northeast of S-Area (Figure 3-1), approximately 6.2 mi south of the nearest site boundary (WSRC 2007a). The Saltstone Facility, which includes the SPF and the Saltstone Disposal Facility (SDF) (SRS 2006, 2007d), treats evaporator waste from the ETP in H-Area, as well as LLW arising from the tank farm waste. The SPF receives the low-activity ETP waste or treated salt solution from the interim salt processing units (MCU/ARP) and stabilizes it to produce a grout slurry. The slurry is pumped into the engineered disposal vaults that make up the SDF, and when solidified, makes up the non-hazardous low activity saltstone (SRS 2007e).

Construction of the SPF and the first two rectangular vaults (Vaults 1 and 4) of the SDF were completed in 1988. The SPF has operated intermittently since 1990 (SRS 2007e). Both existing rectangular vaults are partially full. It is projected that 64 cylindrical, reinforced concrete disposal cells will be required beyond the existing two rectangular vaults. There is currently one pair of cylindrical, reinforced concrete disposal cells (Disposal Cell 2A and 2B) under construction within the SDF. Future disposal cells will be constructed on a “just-in-time” basis in coordination with salt processing production rates (LWO 2009). After filling, each vault will be capped with clean concrete, and final closure will consist of covering the vaults with engineered closure caps and backfilling (SRS 2007e). See Section 3.4.4 for additional information concerning the SDF.

3.1.5.7 Seepage Basins

Many of the operational areas at the SRS utilized seepage basins to receive LLW. The seepage basins were located as far as practical from streams, with consideration of the depth of the natural water table and soil permeability to allow as much radioactive decay as possible before reaching the local streams (Reed et al. 2002). The more notable seepage basins are those in the Separations Areas, the M-Area Settling Basin, the SRNL Seepage Basins in A-Area, the reactor seepage basins (C-, K-, L-, P-, and R-Areas), and the TNX seepage basins (T-Area).

In the Separations Areas, three unlined F-Area seepage basins (904-41G, 904-42G, and 904-43G) cover approximately 6.5 acres. These basins received approximately 1.8 billion gallons of LLW solutions originating from the processing of uranium slugs and irradiated fuel from 1954 through 1988. The low activity effluents contained a wide variety of radionuclides and dissolved metals. The four unlined seepage basins in H-Area (904-44G, 904-45G, 904-46G, and 904-56G) cover approximately 15.5 acres, and received about 1.6 billion gallons of waste solutions similar to that in the F-Area basins. In addition to those waste solutions, they received effluent from tritium facilities and the RBOF. The seepage basins essentially functioned as designed, but the acidic nature of the basin influent caused mobilization of metals and radionuclides that resulted in groundwater contaminant plumes.

Closures of these seven Separations Areas basins were completed in 1991, which involved partial dewatering, physically and chemically stabilizing the remaining sludge, and covering them with a multilayer infiltration reduction system (SRS 2003a). An additional seepage basin in F-Area, called the Old F-Area Seepage Basin, was used for approximately six months in early 1955, and for three months in 1969, and received a variety of effluents from F Canyon. The resulting groundwater contaminants of concern for this older basin are tritium, I129, and uranium. Stabilization of contaminated soils with grout and placement of a low-permeability cover over the older F-Area basin commenced in 1998 (SRS 2003b).

The M-Area Settling Basin in the M-Area HWMF (see Section 3.1.2) was constructed to receive wastewater from the processes carried out at M-Area, and between 1958 and 1985, approximately two million pounds of volatile organic solvents were routed to the M-Area Settling Basin. The basin was an unlined, man-made depression designed with a liquid capacity of 8 million gallons. The basin overflowed at times, spreading the contamination into an adjacent natural seepage area and Lost Lake. In 1981, a groundwater cleanup operation began at M-Area, under the RCRA, and the basin was certified closed in 1991 (Reed et al. 2002).

The SRNL seepage basins are located in the southern part of A-Area. The four rectangular basins received very low activity waste liquids from two SRNL research facilities. Discharges to the first two basins began in 1954, and all discharges ceased in late 1982. In 1983, characterization of the area showed that the bulk of the radionuclides were contained within the upper few feet of the bottoms of the basins. Tritium was the predominant radionuclide discharged, but other radionuclides included Ac228, Am241, Cs137, Co60, Cm243/244, Pb212, Ra228, Sr90, and isotopes of Pu, Th, and U (SRS 2003c).

The reactor seepage basins received radioactively contaminated purge water from the disassembly basins of the reactors in C-, K-, L-, P-, and R-Areas during the time of reactor operations. For R Reactor, a total of six unlined earthen basins were constructed and used between 1957 and 1964. The basins were backfilled and capped as they were sequentially retired between 1958 and 1964 (SRS 2003d). Three seepage basins associated with C Reactor were used from 1957 to 1970, and again from 1978 to 1986. Remediation of the C Reactor seepage basins was completed in 2002, and involved grouting of one basin and placement of a low-permeability soil cover system over all three basins (SRS 2004). K and L Reactor each had one associated seepage basin. The K-Area basin was used from 1957 to 1969, and the L-Area basin was used from 1958 to 1968 and from 1985 to 1988. Remediation of both basins was completed in 2003, again involving grouting and the installation of low-permeability soil cover systems (SRS 2003e, 2003f). Three unlined seepage basins were used from 1957 to 1970 and from 1978 to 1991 in P-Area. Remediation of the P Reactor seepage basins by grouting and a cover system was completed in 2005 (SRS 2003g, Mamatey 2006). The primary contaminants associated with the reactor basins included tritium, Am241, C14, Co60, Cs137, Ni63, Pu239/240, and Sr90 (SRS 2003d, 2003e, 2003f, 2003g, 2004).

In T-Area, wastewater from pilot-scale tests for the DWPF and the General Separations Area (GSA) was discharged through a process sewer line to the Old TNX Seepage Basin from the mid-1950s until 1980. The old seepage basin received waste containing mercury, other heavy metals, depleted uranium and other radionuclides at low levels. The old basin was closed and backfilled with clean soil in 1981. In 1980, a new basin became operational and discharges to the new basin were carefully regulated to exclude hazardous and/or radioactive wastes (WSRC 1999b). In 1988 the non-hazardous and non-radioactive discharges to the new basin were rerouted to the TNX Effluent Treatment Plant (SRS 2003h).

3.1.6 Savannah River National Laboratory and CMX/TNX (A- and T-Areas)

The SRNL is located in A-Area (Figure 3-1). A-Area occupies 348 acres in the northwest portion of the SRS, approximately 0.4 mi southeast of the nearest site boundary (WSRC 2007a).

The SRNL began as Savannah River Laboratory (SRL) in 1951, to provide research and development support for the startup and operation of the SRS, in its mission of producing nuclear materials for national defense. It initially included a main laboratory and associated waste disposal in A-Area (Reed and Swanson 2007), the Pile Physics Laboratory in M-Area, and two pilot scale test facilities, CMX and TNX, located in T-Area (Reed et al. 2002 and Reed and Swanson 2006b). In 1992, it was renamed the Savannah River Technology Center, and in 2004 was designated as the country's 12th National Laboratory (i.e., SRNL). The laboratory expertise extends from tritium supply and nuclear materials technology to energy technology research and development and environmental and chemical process technology (SRS 2007f).

The first priority of the Laboratory was to gather data needed to support the initial process development at the SRS, which by 1956 led to focusing on improving processes required to fulfill the main missions of the SRS. Because of the operational focus of the SRNL's activities, the Laboratory's history parallels that of the SRS plant operational history. From 1950 to 1964, much of the work conducted at SRNL was in support of reactor operations at the SRS. This included research and development related to control rod configurations, fuel and target improvements, and U233 recovery from irradiated thorium. In the 1960s, SRNL was involved in most phases of the experimental HWCTR design and testing, as well as in the transplutonium programs, particularly the associated separations processes (Reed et al. 2002). The SRNL took part in many projects aimed at improvements to the SRS main mission involving isotope production and separation in the 1970s and 1980s, as well as in projects supporting offsite nuclear reactor programs. This time period for the SRNL was also associated with a greatly expanded role in environmental stewardship and waste management (Reed et al. 2002).

The ending of the Cold War, and shutdown of the SRS reactors, resulted in a shift in the mission of the site as well as the Laboratory in the 1990s to the present. Over the years of operation, a considerable body of knowledge was accumulated by SRNL to support the new missions associated with stewardship of the nuclear weapons stockpile, nuclear materials, and the environment (Reed et al. 2002). The Laboratory continues to develop and employ technologies relevant to these missions.

Activities at the Laboratory have been associated with the groundwater contaminant plume underlying both the A- and M-Areas. In particular, the four seepage basins discussed in the previous section received low-level radioactive waste liquids between 1954 and 1982 (SRS 2003c). Due to the close proximity of the A- and M-Areas to each other, remediation of the groundwater was undertaken as a comprehensive Corrective Action Program for A/M Area (Lewis and Aadland 1994).

Operations in T-Area (Figure 3-1), formerly called CMX/TNX, began in 1951 and included testing of processing methods and equipment before installation in SRS production facilities. T-Area contained many industrial and administrative buildings, warehouses, underground storage tanks, a burial ground, and a seepage basin.

The CMX facility was used to determine the minimum treatment needed for cooling water from the Savannah River that would circulate through the reactor heat exchangers. Hydrodynamic properties of reactor assemblies and components were also studied with respect to failure potential. The CMX facility was closed in 1983, when the remaining functions were moved to the main SRNL laboratory (Reed et al. 2002).

The TNX Area was an industrial facility where pilot-scale testing and chemical process evaluation for fuel and target manufacturing, the Separations Areas, and the DWPF took place. Throughout its operation, wastewater from the area was discharged through a process sewer line to two unlined seepage basins before 1988 and to the TNX Effluent Treatment Plant after 1987 (SRS 2003h).

Process-related debris from an explosion in an experimental evaporator containing uranyl nitrate was also buried in the TNX Burying Ground at the unit in 1953 in four unlined trenches at depths of 6-10 feet below ground surface. Between 1982 and 1984, during an expansion of the TNX facilities, the majority of the buried waste was excavated and moved to the SRS Radioactive Waste Burial Ground. Five small areas were not excavated because they were located under buildings, transformers, and other above ground structures (SRS 2003h).

All structures in T-Area except a pump test facility and a telephone exchange building have been demolished, and contaminated soil and groundwater are being remediated (SRS 2006). T-Area is the first area at the SRS to have achieved closure (SRS 2008a).

3.1.7 Administrative and Support Services (A-, B-, and N-Area)

General site administrative functions are located in B-Area (Figure 3-1). A- and B-Areas are also the locations of many site support services (e.g., radiological and engineering) (WSRC 2007a). The Health Physics Laboratory was originally located in A-Area (Reed et al. 2002) but was moved to B-Area in about 2001.

B-Area consists primarily of administrative, laboratory, and storage facilities. Some B-Area facilities were constructed in the early 1950s, but most are more recent. Contiguous to B-Area is an area formerly called U-Area in which the HWCTR was located (see Reactor Areas discussion). A 70-acre sanitary landfill also is located near B-Area, which began receiving waste in 1974 and has subsequently been closed. The landfill received nonradioactive sanitary waste from SRS administrative areas, cafeterias, and industrial activities (SRS 2002).

N-Area (Figure 3-1), also called Central Shops, contains industrial, administrative offices, health and safety facilities, and warehouses (SRS 2006). Building materials and construction equipment are stored in N-Area, as is hazardous waste, pending shipment to offsite commercial vendors for treatment and disposal.

3.1.8 Other Radioactive Missions

Some of the first non-weapons isotope products made in SRS reactors and extracted in the Separations Areas were heat sources for Arctic stations, as well as thermoelectric sources for use in NASA's space programs. These programs began in the 1950s, but expanded during the late 1960s. Cobalt-60 was one of the first isotopic heat sources produced, but Pu238 proved to be the most effective source for electrical power, and became the standard fuel for radioisotope thermoelectric generators during the 1960s and 1970s. Production of Pu238 began at the SRS in the late 1950s and continued until 1986, and was critical to the space exploration programs (Reed et al. 2002).

The transplutonium work escalated after the shutdown of R Reactor in 1964, beginning with programs to produce Cm244. Although Cm244 was sought as a possible heat source for electricity generation, it was also desired as a target material for production of californium, which was the ultimate goal of the transplutonium programs. Californium was sought after based on its potential in medical applications, oil and mineral exploration and neutron radiography (Reed et al. 2002).

The curium production programs, carried out in C and K Reactors, not only provided source material for the creation of californium, but also provided valuable information on the operation of reactors at high flux. The program to generate californium began in 1969, and concluded in November 1980, in K Reactor, with 2.1 g of californium produced. In the early 1970s, SRL prepared hundreds of californium sources for hospitals and medical research facilities (Reed et al. 2002).

The RBOF, in H-Area, was used as a holding facility for spent nuclear fuels between 1964 and 1992. The original purpose of the RBOF was to provide underwater receipt and storage for used fuel elements from experimental power and research reactors at universities and AEC laboratories. When the official policy governing this receipt expired in 1988, receipt of used uranium fuels was curtailed, and discontinued by 1992. The RBOF is slightly larger than a baseball infield, and encompasses an unloading basin, two storage basins, a repackaging basin, a disassembly basin and an inspection basin. The total volume of all the basins and their linking transfer canals is 500,000 gallons of water (Reed et al. 2002).

In 1996, a policy governing receipt of both domestic and foreign shipments of fuel from research reactors was reinstated under an agreement with the U.S. Department of State and the International Atomic Energy Agency (Reed et al. 2002). In that year, L Basin (in L-Area) was reconfigured to safely handle and store spent nuclear fuel from off-site research reactors and in 1997, the first off-site fuel was received and stored in L Basin. In 1998, SRS decided to consolidate all of the stored spent fuel at the SRS into the much larger, recently refurbished L Basin. Spent nuclear fuel from the SRS production reactors had been stored in L and K Reactors' disassembly basins at that point. By October 2003, all of the fuel previously stored in K Basin and the RBOF had been moved to either chemical separations facilities for processing or L Basin for storage (SRS 2008f). The SRS completed deactivation of the RBOF facility in 2004 (SRS 2006).

In the early 1980s, the SRS agreed to build the Fuel Materials Facility (FMF) to produce fuel for the nuclear navy. The FMF was located in F-Area. In 1982, the SRNL built and began operating a pilot plant related to the FMF project, and by the end of the year, test quantities of fuel material had been produced. Although the FMF, which consisted of five buildings encompassing more than 124,000 sq ft, was completed in 1986, it never went into full operation, and the facility was closed by the Department of Defense in 1989 due to a reduced need for naval fuels (Mamatey 2007, Reed et al. 2002). Demolition of the FMF was completed in March of 2006 (Mamatey 2007).

3.1.9 Future Facilities

3.1.9.1 Salt Waste Processing Facility

In February 1998, work was suspended on the In-Tank Precipitation (ITP) Facility. ITP was expected to process the "salt cake" and supernate (the result of the evaporation process) for the radioactive liquid waste tank farm, to remove cesium, strontium, and actinides. Startup testing of the ITP process had indicated an unacceptable level of benzene production in the process (DOE 2001). In September 2001, a National Environmental Policy Act (NEPA) Record of Decision (ROD) was issued that selected the caustic side solvent extraction (CSSX) technology to form the basis for the design and construction of an SWPF, to replace the ITP process (SRS 2008a).

The SWPF will use MonoSodiumTitanate (MST) to remove the strontium and actinides from the waste and the CSSX process to remove the cesium. The actinide and strontium laden MST slurry and the cesium solution will be transferred to the DWPF. Decontaminated salt waste will be transferred to the SPF for final treatment and disposal in above ground vaults. The SWPF construction phase began in 2008 with the start of radioactive operations planned for FY 2013.

3.1.9.2 Mixed Oxide Fuel Fabrication and Pit Disassembly and Conversion Facilities

Plutonium and nuclear material management missions now being conducted at SRS will be expanded to include materials from dismantled weapons and surpluses from other DOE sites. These missions accomplish the need to reduce stockpiles of weapons-grade plutonium. The SRS was selected to be the location for the DOE's plutonium pit disassembly and conversion and mixed oxide fuel fabrication facilities. In these facilities, excess weapons-usable plutonium is converted to a form that can be used in commercial power reactors (SRS 2008a).

In August of 2007, construction began on the Mixed Oxide (MOX) Fuel Fabrication Facility, which is being built in F-Area. The Pit Disassembly and Conversion Facility will disassemble pits and convert the plutonium inside into MOX fuel. A support facility for these two plants, called the Waste Solidification Building, is also in the design phase (SRS 2008a).

3.2 SAVANNAH RIVER SITE AND FACILITY CHARACTERISTICS

3.2.1 Regional Geography

The SRS occupies approximately 310 mi^2 in South Carolina and is bounded on the southwest by the Savannah River for about 17 mi (WSRC 2007a). The site encompasses 198,344 acres in Aiken, Allendale, and Barnwell counties of South Carolina and is approximately 12 mi south of Aiken, South Carolina, and 15 mi southeast of Augusta, Georgia (Mamatey 2007). The SRS is about 100 mi from the Atlantic Ocean. Figure 3-2 shows the location of the SRS relative to the Savannah River and the surrounding counties.

The SRS is located on the Upper Atlantic Coastal Plain in southwestern South Carolina. The Fall Line, which separates the Atlantic Coastal Plain physiographic province from the Piedmont physiographic province, is approximately 25 mi northwest of the central SRS (Figure 3-3).

Prominent geographic features within 30 mi of the SRS include the Savannah River and Clarks Hill Lake (also known as Thurmond Lake) (Figure 3-4). Clarks Hill Lake is the largest nearby public recreational area. This reservoir lies on the Savannah River approximately 40 mi upstream of the center of the SRS.

Within the SRS boundary, prominent water features include PAR Pond and L Lake (Figure 3-1). PAR Pond is a 4-mi^2 former reactor cooling water impoundment that lies in the eastern sector of the SRS. L Lake is a 1.5 mi^2 former reactor cooling water impoundment that lies in the southern sector of the SRS.

(Figure 3-1) also shows the major operational areas at the SRS (see Section 3.1). Prominent operational areas, both past and present, include (WSRC 2007a):

- Reactor Areas (C-, K-, L-, P-, and R-Areas)
- Separations and Tank Farm areas (F- and H-Areas)
- Waste Management Operations (E-, S-, and Z-Areas)
- Heavy water reprocessing area (D-Area)
- Reactor materials area (M-Area)

Administration and support areas are located in A- and B-Areas. The Central Shops area is in N-Area. SRNL and SREL are located in A-Area. D-Area, M-Area and TNX have undergone D&D (Mamatey 2007).

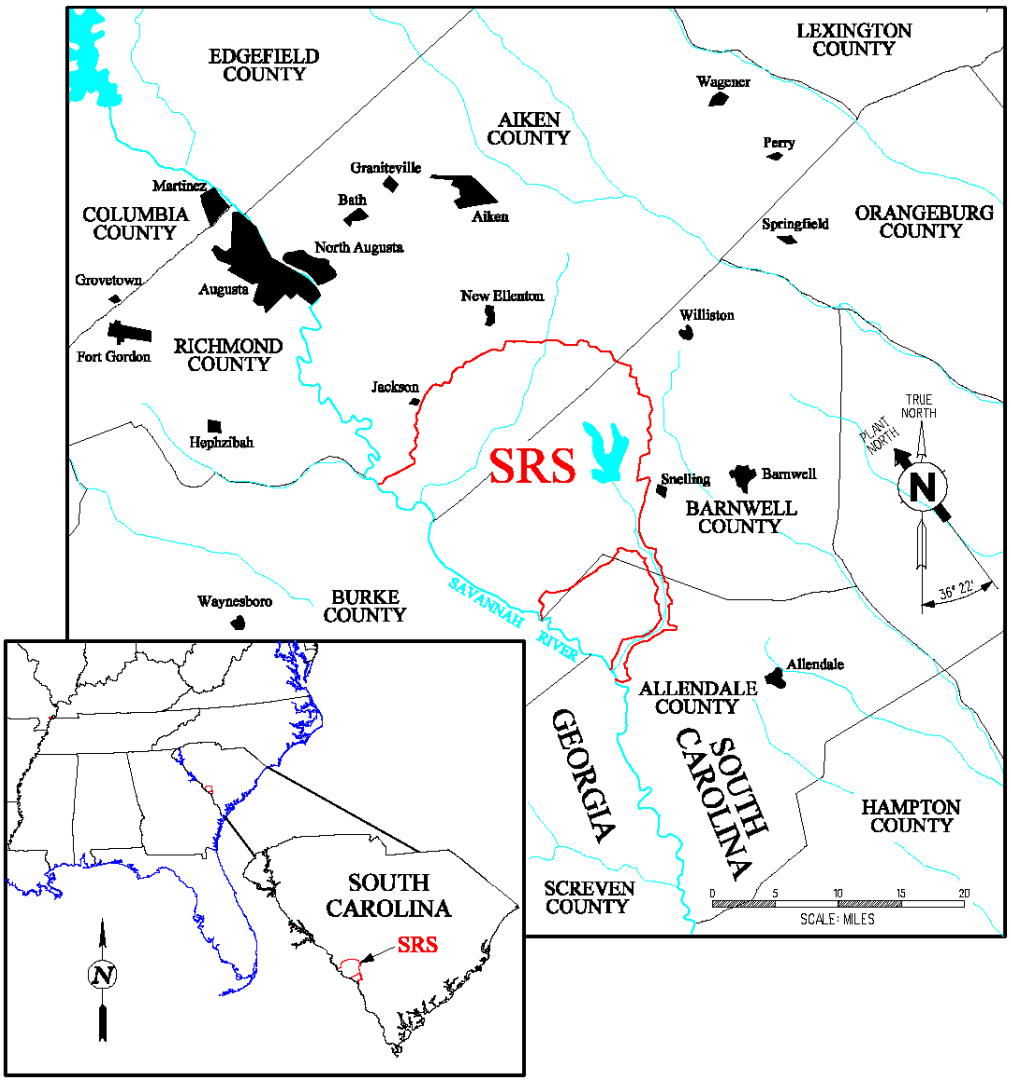


Figure 3-2. Physical Location of Savannah River Site

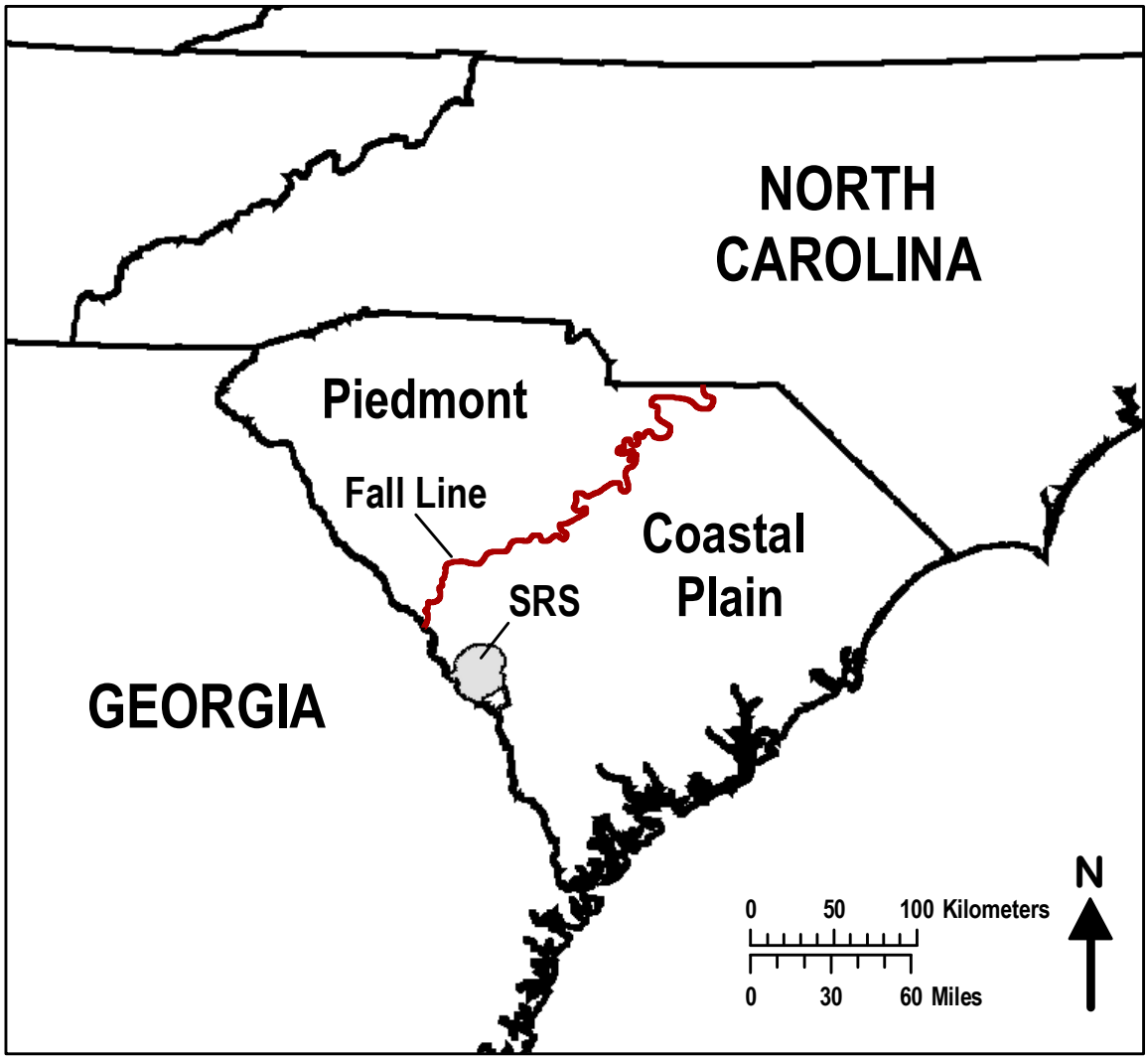


Figure 3-3. Physiographic Location of Savannah River Site

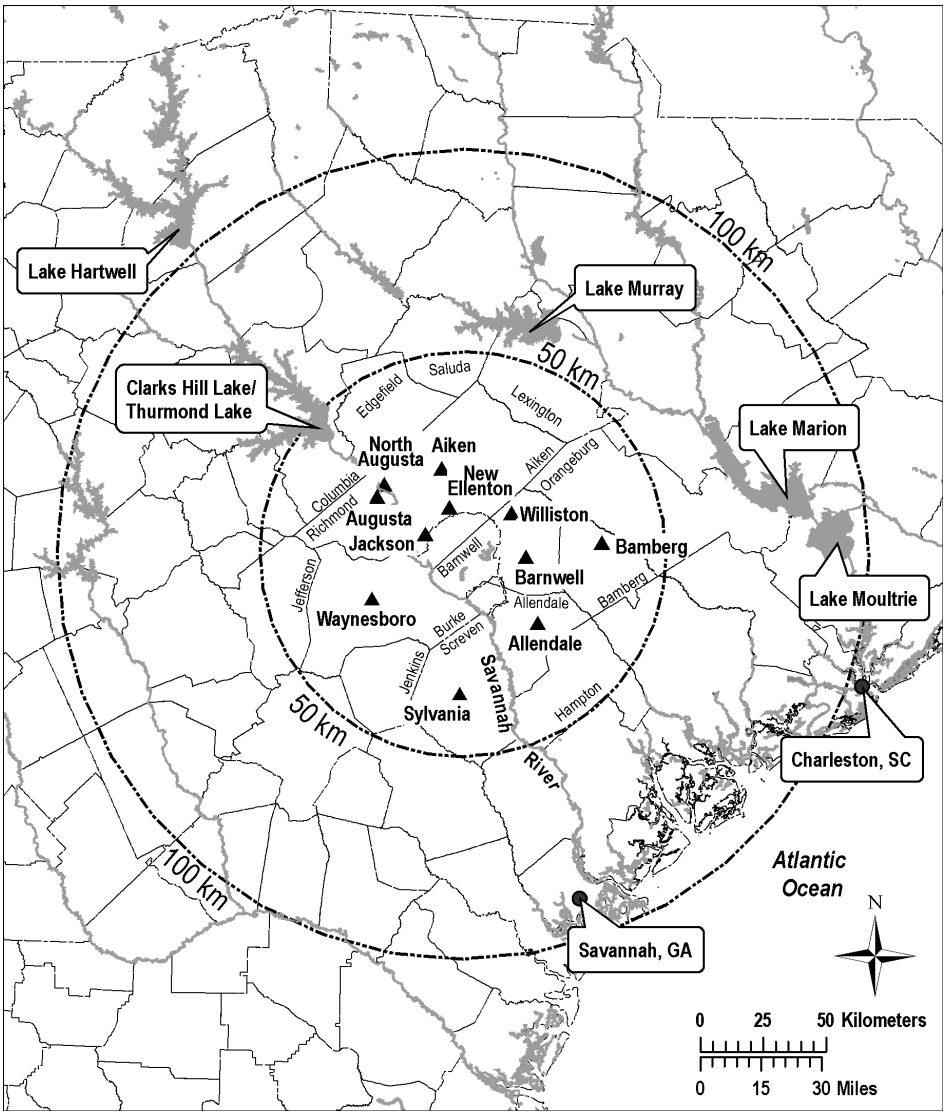


Figure 3-4. Location of Savannah River Site Relative to Geographic Features and Local Communities

3.2.2 Demography

According to the High-Level Waste Tank Closure Final Environmental Impact Statement (DOE 2002), the socioeconomic region of influence for that facility is a six-county area around the SRS where the majority of site workers reside. However, the more recently documented FTF PA (WSRC 2008b) considers eight counties in the vicinity of the SRS, adding two counties in Georgia (Burke and Screven Counties). The CA includes the two additional counties in considering demographics of interest in the SRS vicinity.

Based on U.S. Census Bureau data (from its website: [http://factfinder.census.gov/home/saff/main.html?_lang=en]), the estimated 2007 population in the eight-county region of influence (ROI) was 546,472. Four of the counties lie in South Carolina and include: Aiken, Allendale, Bamberg, and Barnwell. The other four counties lie in Georgia and include Burke, Columbia, Richmond and Screven. Approximately 84% of the ROI population lives in the following three counties: Aiken (27.9%), Richmond (36.1%) and Columbia (20.0%). Only about 16% of the ROI population lives in the remaining counties as shown in Table 3-1.

From 2000 to 2007 the population in the eight-county region grew an estimated 4.8%. Columbia County had the highest estimated growth at approximately 22% followed by Aiken County with an estimated growth of approximately 7% and Burke County with an estimated growth of 2%. Richmond, Barnwell, Allendale, and Bamberg Counties experienced a net population loss. Calculations are based on information obtained from the U.S. Census Bureau website, cited above.

Population projections indicate that the overall population in the region should continue to grow through 2030 in the eight counties (Table 3-2). Allendale and Bamberg County, SC, and Screven County, GA are expected to have little growth or a population decline; Richmond and Columbia County, GA and Aiken County, SC are predicted to have the greatest growth.

Table 3-1. Population Distribution and Percent of ROI for Counties and Selected Communities

Jurisdiction	2007 Population Estimate[a]	2007 % ROI
SOUTH CAROLINA		
Aiken County	152,307	27.9
Allendale County	10,475	1.9
Bamberg County	15,452	2.8
Barnwell County	23,975	4.3
GEORGIA		
Burke County	22,754	4.2
Columbia County	109,100	20.0
Richmond County	197,372	36.1
Screven County	15,037	2.8
Eight-County Total	546,472	100

a. 2007 Population estimates based on 2000 population census and are provided by the U.S. Census Bureau Population Estimates Program [http://factfinder.census.gov/home/saff/main.html?_lang=en]; data for births, deaths, and domestic and international migration were used by the U.S. Bureau of the Census to update the 2000 base counts.

Table 3-2. Population Projections and Percent of ROI

Jurisdiction	April 1, 2000[a] (Estimate Base)	ROI (%)	Projection July 1, 2010	ROI (%)	Projection July 1, 2020	ROI (%)	Projection July 1, 2030	ROI (%)
SOUTH CAROLINA[b]	4,011,816	NA	4,482,260	NA	4,998,110	NA	5,564,460	NA
Aiken County	142,552	27.4	159,540	28.3	179,130	29.2	200,490	29.9
Allendale County	11,211	2.1	11,300	2.0	11,590	1.9	11,800	1.8
Bamberg County	16,658	3.2	15,450	2.8	14,680	2.4	13,760	2.1
Barnwell County	23,478	4.5	24,720	4.4	26,770	4.3	28,540	4.2
GEORGIA	8,186,816	NA	9,517,760	NA	10,788,860	NA	12,172,150	NA
Burke County[d,e]	22,243	4.3	24,733	4.4	27,591	4.5	30,729	4.6
Columbia County[c]	89,288	17.1	111,170	19.7	131,530	21.4	153,280	22.9
Augusta-Richmond County[c]	199,775	38.4	202,410	35.9	210,250	34.2	220,070	32.8
Screven County[d,e]	15,374	3.0	14,020	2.5	12,785	2.1	11,659	1.7
EIGHT-COUNTY TOTAL	520,579	100.0	563,343	100.0	614,326	100.0	670,328	100.0

a. U.S. Bureau of the Census data [http://factfinder.census.gov/home/saff/main.html?_lang=en
b. Projections for Aiken, Allendale, Bamberg and Barnwell Counties are from the SC Office of Research and Statistics, Health and Demographics Division. They are based on 2003 Census population estimates [http://www.ors2.state.sc.us/population/proj2030.asp].
c. Projections are from the 2005 City of North Augusta Comprehensive Plan [http://www.northaugusta.net/Dept_Serv/EconComDev/CompPlan/2%20-%20Population.pdf]
d. Projections for 2010 are based on calculations by the Georgia Governor's Office of Planning and Budget, revised August 2002 [http://www.gadata.org/information_services/Census_Info/NewCensus2000proj.htm]
e. Projections for 2020 and 2030 are estimated in the CA based on the assumption that the trend between 2000 and 2010 continues similarly for the next 20 years.

3.2.3 Land Use Patterns

3.2.3.1 Savannah River Site Land Use

Existing land use on the SRS site can be characterized in three main categories: (1) undeveloped/forest, (2) wetlands/water, and (3) developed. Approximately 73% of the SRS is undeveloped; 22% consists of wetlands, streams, and lakes; and 5% is developed (e.g., facilities, roads, and utility corridors) (DCS 2002).

The forested areas are managed for timber production. The U.S. Forest Service, under an interagency agreement with DOE, harvests approximately 0.8 to 1.3 mi^2 out of a potential 2.8 mi^2 of timber from the SRS each year (Blake 2005). Blake (2005), Barton et al. (2005) and Wike et al. (2006) provide further information regarding forest management practices at the SRS.

Prime farmland soils exist at the SRS, but areas of prime farmland are not identified within the SRS because the land is not available for agricultural activities. A portion of the SRS, designated as the Crackerneck Wildlife Management Area (WMA) is open for fishing. Limited hunting is allowed at the SRS to control the deer population and feral hog populations (Johns and Kilgo 2005, Mayer 2005). Three Rivers Landfill, operating under a 50-year lease agreement, is a municipal solid waste landfill located within the SRS boundary. The Three Rivers Solid Waste Authority provides waste management services to local governments in Aiken, Allendale, Bamberg, Barnwell, Calhoun, Edgefield, McCormick, Orangeburg, and Saluda counties (DOE 2005).

The SRS has been designated a National Environmental Research Park by DOE. The scientific community can use the site to study past impacts of human activity on local ecosystems. Approximately 22 mi^2 of land has been set aside at the SRS for nondestructive environmental research and monitoring. The set-aside areas encompass a wide range of ecological conditions including Carolina bays, major stream systems, fields, and old experimental sites (Davis and Janecek 1997, Blake et al. 2005a, Wike et al. 2006).

As SRS transitioned from the Cold War to the post-Cold War era, the Site's missions have changed from primarily a defense mission to one that includes environmental stewardship and future operational missions not yet defined. Current activities include the development of future infrastructure and facilities in addition to decommissioning obsolete facilities.

Three documents address SRS land use plans for the future. The SRS Comprehensive Plan/Ten Year Site Plan (SRNS 2009b) has been developed with guidance and feedback from the State of South Carolina and community groups such as the SRS Citizens Advisory Board, the Central Savannah River Area (CSRA) Planners Group, the Savannah River Regional Diversification Initiative, Citizens for Nuclear Technology Awareness, local chambers of commerce, and local economic development organizations. The plan defines a vision for the evolution of SRS over the next 50 years and for the long-term stewardship of the site.

The SRS End State Vision (DOE 2005) describes current SRS conditions and the planned end states for disposing all nuclear material and waste hazards permanently, decommissioning all facilities, and remediating all inactive waste sites. The SRS End State Vision states that the entire site will be owned, controlled, and maintained by the federal government, most likely DOE, in perpetuity, as established by Congress. Site boundaries will remain unchanged and the site will be used for industrial purposes for future DOE and non-DOE missions. Residential use will not be allowed onsite.

The U.S. Department of Energy Savannah River Site Strategic Plan (DOE 2009) lays out the plan for the future of SRS. One of the purposes of the strategic plan is to establish the connection between the DOE Strategic Plan and the SRS vision. The strategic plan has a number of planning bases, two of which are "SRS is a national asset and will endure beyond current EM and NNSA missions" and "SRS will maintain its current physical boundary of 310 square miles." The plan also has several planning assumptions, one of which is "The SRS Federal landlord responsibilities will remain with EM until its missions are complete."

3.2.3.2 Regional Land Use

Regional land uses in the vicinity of the SRS, but offsite, can predominantly be classified as urban, residential, industrial, agricultural, or recreational. In the area adjacent to the SRS, less than 8 percent of the existing land is devoted to urban and built-up uses. Most such uses are in and around the cities of Augusta and Aiken. Agriculture accounts for about 21 percent of total land use; forests, wetlands, water bodies, and unclassified, predominantly rural, lands account for about 70 percent.

Forest and agricultural land predominantly border the SRS, with only limited urban and residential development. The nearest residences are located to the west, north, and northeast, some within 200 ft of the SRS boundary. Farming is diversified throughout the region and includes such crops as peaches, watermelon, cotton, soybeans, corn, and small grains. Incorporated and industrial areas are also present near the site, including textile mills, polystyrene foam and paper plants, chemical processing plants, a commercial nuclear power plant, and a commercial LLW disposal facility (WSRC 2007a).

Open water and nonforested wetlands occur along the Savannah River Valley. The Crackerneck WMA, which includes a portion of the SRS along the Savannah River, is open to the public for fishing. It encompasses about 17 mi^2 and consists of pine, bottomland hardwood, and cypress-tupelo swamp habitats (Blake et al. 2005a). Other recreational areas within 50 mi of the SRS include Sumter National Forest, Santee National Wildlife Refuge, and Clarks Hill/Thurmond Lake. State, county, and local parks include Redcliffe Plantation, Rivers Bridge, Barnwell and Aiken County State Parks in South Carolina, and Mistletoe State Park in Georgia (DOE 1999).

Non-SRS nuclear facilities within 50 mi of the SRS include Chem-Nuclear Systems, Inc. (CNSI) Low-Level Radioactive Waste Disposal Facility in Barnwell, SC, and the Vogtle Electric Generating Plant (VEGP), a nuclear power facility across the Savannah River from the SRS[1]. Both of these non-DOE nuclear facilities are radionuclide sources to the Savannah River. CNSI, a 300-acre site located in Barnwell County, South Carolina, near the eastern SRS boundary, is a commercial facility for the disposal of LLW and hazardous chemicals. CNSI facilities include a burial site, transportation, and maintenance units, and facilities for waste solidification and decontamination. It was licensed for operation in 1971 (WSRC 2007a).

The VEGP is a two-unit nuclear power plant licensed by NRC and located across the Savannah River from the SRS. It is located in Burke County, Georgia, and is about 4.5 mi south-southeast of 400-D Area (see Figure 3-1). Unit 1 was licensed for full-power operation in May 1987. Unit 2 began operation in May 1989 (WSRC 2007a). The VEGP withdraws an average of 92 cubic feet per second (cfs) from the Savannah River for cooling purposes, and returns an average of 25 cfs (CSRACT 2007). An analysis of effects from offsite facilities conducted in support of a 1990 Final Environmental Impact Statement (DOE 1990) indicated that radiological impacts from the VEGP were very small (WSRC 2007a). In August, 2009, the VEGP was issued an early site permit for two reactors. Despite the addition of the new reactors, VEGP's radiological impacts are expected to continue to be very small.

3.2.4 Ecology

Comprehensive descriptions of the SRS ecological resources and wildlife can be found in *Ecology and Management of a Forested Landscape: Fifty Years on the Savannah River Site* (Kilgo and Blake 2005) and *SRS Ecology: Environmental Information Document* (Wike et al. 2006). The discussion of SRS ecology provided in this section is largely summarized from these two documents.

With the majority of the SRS undeveloped, the site sustains a variety of ecosystems. Within these ecosystems, habitat types include upland pine forests, mixed hardwood forests, bottomland hardwood forests, swamp forests, Carolina bays, streams and lakes. Since the early 1950s, the site has changed from 60 percent forest/aquatic habitats and 40 percent agriculture to 95 percent forest/aquatic habitats, with the remainder in developed (facility) areas. The wildlife correspondingly shifted from forest farm edge species to a predominance of forest-dwelling species. Currently, 255 species of birds, 59 species of reptiles, 44 species of amphibians, 80 species of freshwater fish, and 55 species of mammals exist on the SRS (Mamatey 2006, Wike et al. 2006). These populations include several commercially and recreationally important species, and seven species of federally threatened or endangered plants and animals. The SRS has designated thirty-two species of animals as sensitive (Kilgo and Blake 2005, Wike et al. 2006).

[1] Although these non-SRS commercial nuclear operations may release radionuclides into the Savannah River and may leave residual radioactive material when operations cease, they are not considered in this SRS CA because the maximum dose projected in the CA is less than 30 mrem in a year (see Section 10.3).

SRS has extensive, widely distributed wetlands, most of which are associated with floodplains, creeks, or impoundments. In addition, approximately 343 Carolina bays and baylike depressions occur on SRS (Kolka et al. 2005). Carolina bays are unique wetland features of the southeastern United States. They are isolated wetland habitats dispersed throughout the uplands of SRS. The SRS Carolina bays exhibit extremely variable hydrogeology and a range of plant communities from herbaceous marsh to forested wetland (DOE 2002). The source of recharge water for most of the SRS Carolina bays appears to be rainwater, though periodic inflow from groundwater may be important for a few bays (Kolka et al. 2005). Thus, bays tend to have fluctuations in water levels, and some are more sensitive to rainfall and evapotranspiration than others.

3.2.4.1 Vegetation

There are 1,322 vegetation species, representing 151 separate taxonomic families, listed for the SRS. These species occur over many habitat types, from upland well-drained forests to swamps and Carolina bays (Wike et al. 2006).

Partially as a result of land management practices and industrial usage since the SRS was acquired in 1950, there are a number of different vegetative land cover types that can be identified. The nonforested cover types include grassland/forb and scrub-shrub areas. The forested cover types include evergreen, deciduous, bottomland hardwood, and swamp forests. Carolina bays can exhibit both nonforested and forested regions (Wike et al. 2006).

Nonwoody plants dominate the grassland/forb cover type, with more than 50% of the vegetation cover grasses and forbs. This land cover type occurs primarily on power line rights-of-way, in a few forest openings, or on land from which trees have been cleared recently. The scrub-shrub cover type includes predominantly bare soil, a scrub-shrub canopy of less than 25% closure, or young pine seedlings and saplings less than 5 years old and less than 6 m (20 ft) tall (Wike et al. 2006).

The nonforested portion of Carolina bays can include several bands of herbaceous vegetation, each of which is dominated by grass species. The driest zones of the bays are characterized by broomsedge (*Andropogon virginicus*), but also contains numerous herbs including pitcher plants (*Sarracenia* spp.). Inside these zones, closer to the bay's center, is a band of vegetation dominated by three awn grass (*Aristida affinis*), and in deeper water areas, surrounding the central pool of water, species of maidencane (*Panicum* spp.) are abundant. The pond in the middle of the bay contains typical floating-leaved aquatic plants such as the water lilies *(Nymphaea odorata* and *Nymphoides aquaticum*) (Wike et al. 2006).

The evergreen forest land cover type includes areas with predominately coniferous trees that are at least 6 m (20 ft) tall. Pines, primarily longleaf and loblolly pine, dominate the evergreen forested areas. Slash pine (*P. elliottii*) and shortleaf pine (*P. echinata*) are also common on SRS. The U.S. Forest Service Management Plan established much of this cover type, although some natural coniferous stands still exist. This cover type is generally located on dry upland sites and includes former agricultural fields (Wike et al. 2006).

Upland hardwood cover types include areas where the dominant species are deciduous trees at least 6 m (20 ft) tall. An area is classified as a deciduous forest when deciduous trees compose at least 70% of the canopy layer. The deciduous forest class includes the upland hardwood forest and the mesic hardwood forest described by Whipple et al. (1981). On drier sites and sandy ridges, turkey, bluejack, and black-jack oaks (*Quercus laevis*, *Q. incana*, and *Q. marilandica*) dominate the canopy with longleaf pine present in various densities. In less dry areas, other oaks and hickories (*Carya* spp.) also are present. On mid- and lower slopes, the deciduous forest includes laurel oak (*Q. laurifolia*), yellow poplar (*Liriodendron tulipifera*), blackgum (*Nyssa sylvatica*), sweet-gum (*Liquidambar styraciflua*), red maple (*Acer rubrum*), hickories, and holly (Wike et al. 2006).

Bottomland hardwood forests are found along SRS streams and on the "islands" or "ridges" of the Savannah River swamp and major drainages. Elevations of these ridges are high enough to avoid prolonged flooding during most years. Typical canopy species include water oak (*Q. nigra)*, laurel oak, sweetgum, elms (*Ulmus alata* and *U. americana)*, red maple, and yellow poplar (Wike et al. 2006).

The swamp forest is common along the western boundary of the site, adjacent to the Savannah River. The Savannah River floodplain supports an extensive swamp, covering about 15 square mi of SRS; with a natural levee separating the swamp from the river. This low-lying area is subject to prolonged inundation during one or several periods of the year. The Savannah River and, to a lesser extent, the several streams that empty into it control the hydrology of the swamp. Bald cypress (*Taxodium distichum*) and water tupelo (*Nyssa aquatica*) dominated the historic swamp forest on SRS. Historical reactor operations impacted some areas of this cover type (Wike et al. 2006). Areas that are slightly elevated and well-drained are characterized by a mixture of oak species, as well as red maple, sweetgum, and other hardwood species (DOE 2002).

In 1967, the DOE designated a series of set-aside areas on SRS. The SREL selects and manages these sites as representative of major or unique vegetational communities on site. The set-aside program was initiated to promote ecological research, and began with 10 set-aside areas, which has been expanded to include 30 designated areas and a total of 5,668 ha (14,005 acres) (Figure 3-5). The set-aside program protects a variety of habitat types and includes examples of pine stands, hardwood forest, riparian areas, and swamp forest. Many of these areas contain Carolina bays and are the habitat for the site's threatened and endangered plants. Because the set-asides are within the SRS boundaries, they are subjects of long-term projects that are protected from routine public access. They also provide control areas for other ecological change studies, because they are subject to very few of the disturbances associated with other areas on SRS (Wike et al. 2006).

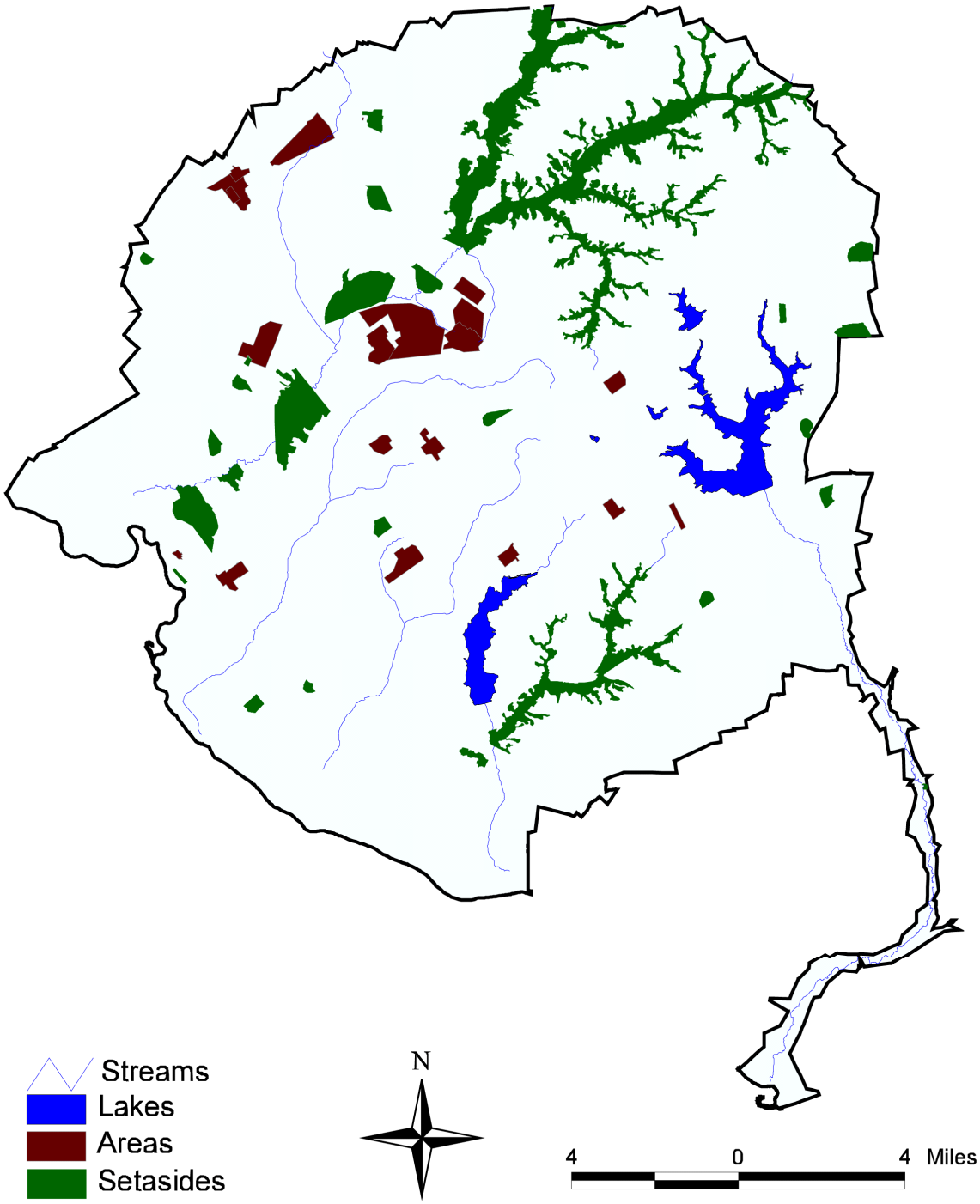


Figure 3-5. Location of SRS Set-Asides

3.2.4.2 Wildlife

The SRS supports abundant herpetofauna because of its temperate climate and diverse habitats. The species of herpetofauna include 17 salamanders, 27 frogs and toads, one crocodilian, 13 turtles, nine lizards, and 36 snakes. The class Amphibia is represented on site by two orders, 11 families, 16 genera, and 44 species. The Reptilia are represented by three orders, 12 families, 41 genera, and 59 species (Wike et al. 2006).

The amphibians and reptiles on SRS are year-round residents and use a variety of aquatic and terrestrial SRS habitats. The highest overall herpetofauna species diversity is in aquatic habitats. More than 70% of the site's amphibian and reptile species use stream corridors and Carolina bays. Only 9% have been documented in the Savannah River. Between 35% and 50% of the species have been collected from terrestrial habitats. The highest diversity is in bottomland hardwood forests, and the lowest is in old fields and clearcuts (Wike et al. 2006).

The SRS supports a diverse bird population that includes migrant, seasonal, and permanent residents, representing 255 species. Waterfowl and wading birds, as well as many upland species, use SRS aquatic habitats year round. Sixty-seven percent use Carolina bays and emergent marshes. Sixty-eight percent of the upland species use this habitat type. Edge or shoreline areas accounted for high numbers of upland birds at Carolina bays and emergent marshes, stream, and small drainage corridors, and river swamp habitats. The aquatic birds are most common in large and small open water habitat (Wike et al. 2006).

The SRS supports a diverse fish population in a variety of aquatic habitats. Since the 1950s, scientists have collected 87 fish species in 23 families from SRS streams, ponds, and lakes, and along the Savannah River adjacent to the site. Fourteen species have been identified from SRS isolated wetlands, although fewer than 10% of these bays and depression wetlands have permanent fish populations (Marcy 2005a).

The fish populations in SRS streams that are not impacted by site operations are typical of similar-sized streams from other areas of the Southeast. However, stretches of Fourmile Branch, Steel Creek, and Pen Branch downstream of C, L, and K Reactors were largely devoid of fish during reactor operations due to temperature fluctuations resulting from cooling water discharges. However, these streams have been recolonized since reactor operations ceased. Fish species composition and abundance, as well as size of individuals, vary with stream size (depth and habitat space) (Marcy 2005a).

Currently, the L Lake fish community includes at least 19 species with the most abundant being brook silversides, yellow perch, bluegill, redbreast sunfish, coastal shiner, largemouth bass, chain pickerel, and spotted sunfish. These species are generally common in southeastern reservoirs with abundant aquatic vegetation. At least 30 species of fish reside in PAR Pond, which differs from most other Southeastern U.S. reservoirs in that it has unusually high densities of largemouth bass and chubsuckers and unusually low densities of shad. High densities of largemouth bass are due in part to a virtual absence of fishing pressure in PAR Pond (Wike et al. 2006).

The Academy of Natural Sciences of Philadelphia began sampling Savannah River fish in 1951 to determine the effects, if any, of discharges from the SRS on the fish community. Between 1980 and 1995, 59 species were collected, with a median number of 33 species per year. The 1980-1995 survey showed little evidence of long-term increases or decreases in species abundance, but many species exhibited temporary increases. No evidence of detrimental effects on the fishery of the Savannah River in the vicinity of the SRS was found (Marcy 2005a).

3.2.4.3 Threatened, Endangered, and Sensitive Species on the Savannah River Site

Within the diverse flora of the SRS, with 1322 species and varieties of 558 genera, exist several unusual and rare species, two of which are listed as Federally endangered; the smooth purple coneflower (*Echinacea laevigata*), and pondberry (*Lindera melissifolia*). In addition to these two species, other plants are expected to be added as species of special concern in South Carolina in the future; the blue wild indigo (*Baptisia australis*), Chapman's sedge (*Carex chapmanii*), Collin's sedge (*C. collinsii*), long sedge (*C. folliculata*), and Candy bulrush (*Scirpus etuberculatus*) (Wike et al. 2006). A list of forty-four sensitive plants occurring on the SRS, including the two endangered species, is provided by Imm (2005).

The shortnose sturgeon (*Acipenser brevirostrum*) is an endangered anadromous fish that occurs in the Savannah River adjacent to the SRS. It spawns upstream and downstream from the SRS, and potential spawning habitat exists in a portion of the river adjacent to the site (Marcy 2005b, Kilgo and Blake 2005).

The American alligator (*Alligator mississippiensis*) has been studied extensively on the SRS. The alligator has benefited from federal protection during the 1960s and 1970s, and as a result has been reclassified from "endangered/threatened" to "threatened due to similarity of appearance" to other species that remain highly endangered. It is estimated that there is a total population of the American alligator in PAR Pond alone of 300 to 350 (Brisbin 2005).

The American wood stork (*Mycteria Americana*) was classified as a federally endangered species in 1984. The species does not breed on the site, but birds from nearby nesting colonies in Georgia use SRS wetlands for foraging. Peaks of foraging occurred when reactor operations influenced water levels and habitat in the Savannah River swamp. Storks continue to forage in the Carolina bays and other ephemeral wetlands on the site (Bryan 2005, Kilgo and Blake 2005).

Bald eagles (*Haliaeetus leucocephalus*), listed as a threatened species since 1995, have used reservoir habitats on the SRS since 1959, and one to two pairs breed annually on the site. Prebreeding eagles and migratory transients also use the SRS (Bryan and Jarvis 2005).

Finally, the endangered red-cockaded woodpecker (*Picoides borealis*) is the single greatest driver of land management activity on the SRS. Many aspects of forest management are geared toward providing nesting and foraging habitat for these woodpeckers. The SRS population declined to a low of four birds in 1985, but has recovered dramatically through intensive management, such that the population totaled 165 birds in 34 groups in 2000, and is expected to reach 250 breeding groups by 2037 (Johnston 2005, Kilgo and Blake 2005).

The SRS has designated 32 species of animals as sensitive, including one insect, eight mollusks, one fish, seven reptiles and amphibians, eight birds, and seven mammals. This designation refers to animals not federally protected under the Endangered Species Act of 1973, but whose population viability is a concern. In addition to the 32 species, the SRS has designated the headwater portion of Pen Branch (above Indian Grave Branch) as a stream worthy of protection due to its unique faunal characteristics (Jarvis 2005).

3.2.5 Soils

3.2.5.1 Geologic Origin and Geomorphology

The SRS is primarily located in the southern portion of the South Carolina upper Atlantic Coastal Plain (Aadland et al. 1995). The northeastern and central portions of the SRS are located on the Aiken Plateau, while the southwestern portions of the SRS are located on erosional terraces, identified by Cooke (1936) as Pleistocene marine terraces (Wike et al. 2006).

The Aiken Plateau is highly dissected and characterized by broad interfluvial areas with narrow, steep-sided valleys (Aadland et al. 1995). The erosional terraces, in the southwestern portions of the SRS represent successive marine recessions during the glacial epoch about 10,000 to 1 million years ago (Figure 3-6). The Brandywine Terrace is the highest and oldest. It lies adjacent to the Aiken Plateau and parallels the Savannah River at elevations between 45 and 76 m (150 and 250 ft) above sea level. The Sunderland Terrace, the second oldest, lies between 27 and 45 m (90 and 150 ft) above sea level. The Wicomico Terrace, the youngest, lies along the Savannah River flood plain, between the river and about 27 m (90 ft) above sea level (Cooke 1936, Langley and Marter 1973, Wike et al. 2006).

Atlantic Coastal Plain sediments are stratified sands, clays, limestones, and gravels that dip gently and thicken southeastward. The sediment ages range from Late Cretaceous to Recent. In the central SRS area, the sedimentary sequence is approximately 213 m (700 ft) thick (Wike et al. 2006).

The Coastal Plain sediments are underlain by either crystalline basement rocks or Triassic-aged consolidated sedimentary rocks (Wyatt and Harris 2000). The primary geologic units exposed at land surface include: Quaternary alluvium, silty, clayey sands and conglomerates of the (probably Miocene) Upland Unit, clayey sands of the Late Eocene Tobacco Road Formation, sands and clays of the Late Eocene Dry Branch Formation, McBean Formation sands, Huber Formation sandy clays and Congaree Formation sands.. A SRS geologic map adapted from Prowell (1996) is presented in Figure 3-7 (Wike et al. 2006).

For much of the SRS, the vadose zone is comprised largely of the Upland Unit (Figure 3-7). The occurrence of cross-bedded, poorly sorted sands with clay lenses in the Aiken Plateau indicates fluvial deposition (high-energy channel deposits to channel-fill deposits) with occasional transitional marine influence. This depositional environment results in wide differences in lithology and presents a very complex system of transmissive and confining beds or zones. The lower surface of the "Upland Unit" is very irregular due to erosion of the underlying formations.

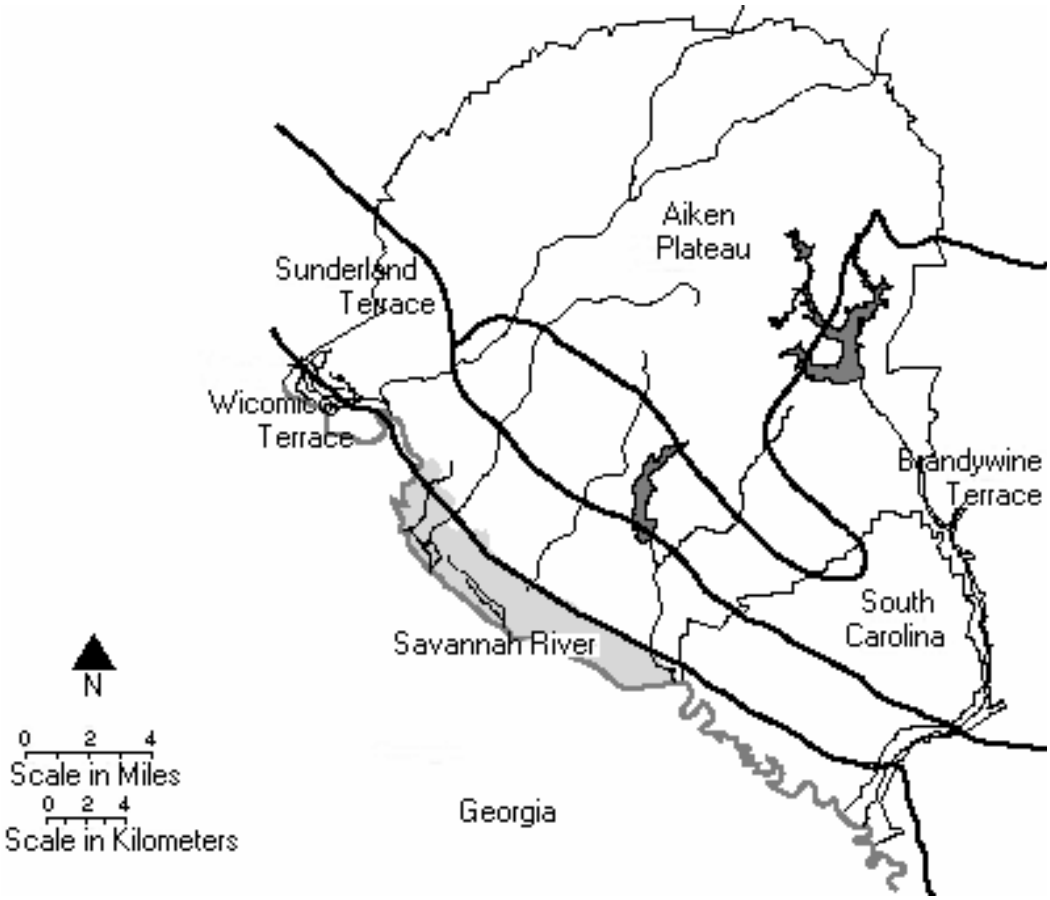


Figure 3-6. Location of Pleistocene Marine Terraces on SRS (Source: Langley and Marter 1973)

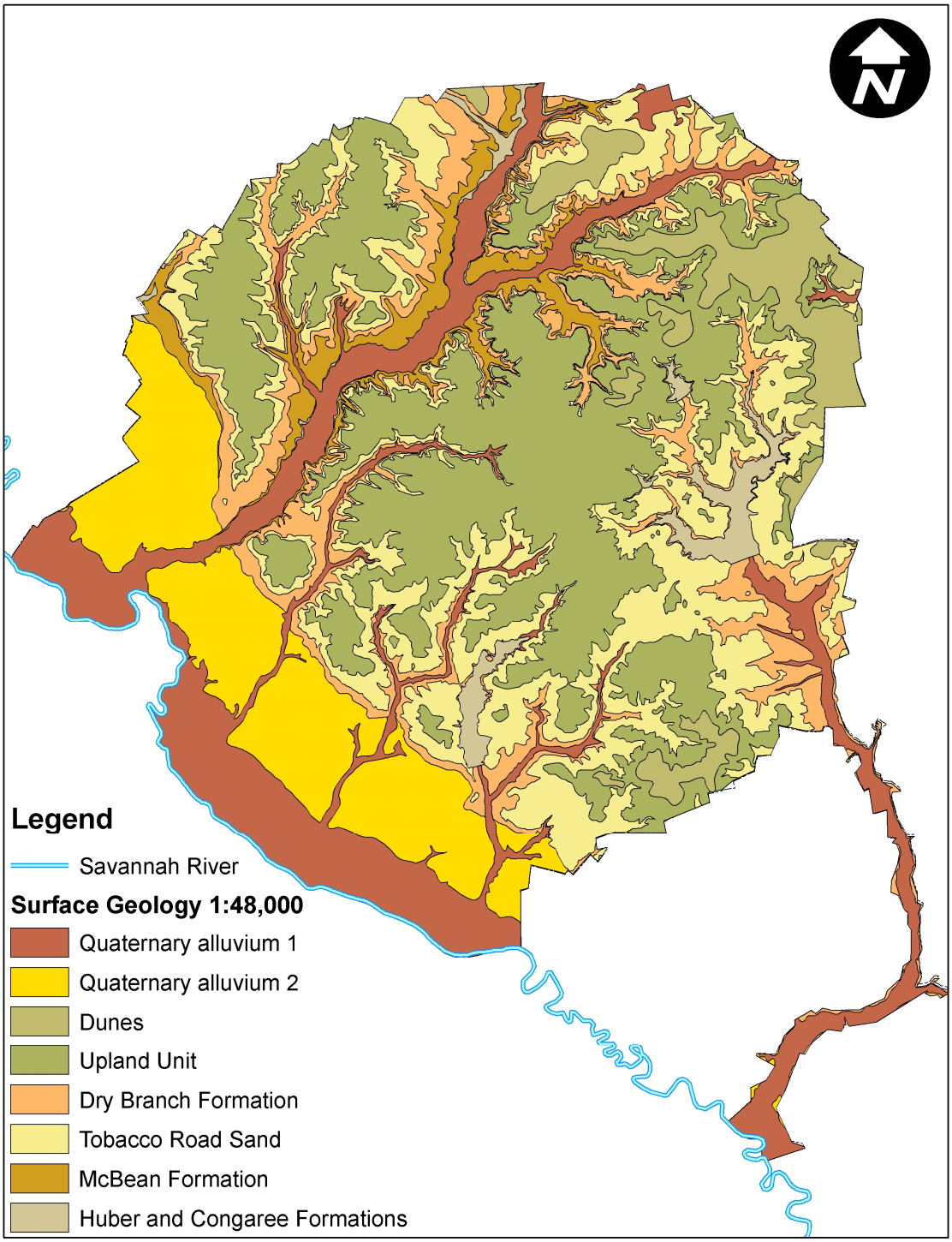


Figure 3-7. SRS Surface Geology (adapted from Prowell 1996)

3.2.5.2 Soil Groups of the Savannah River Site

The distribution of soil types is very much influenced by the creeks on the site, with colluvial deposits on hill-tops and hillsides giving way to alluvium in valley bottoms (Dennehy et al. 1989). In addition to erosion by streams, weathering also affects soil characteristics. Soils at the SRS are typical of soils found in moderately aggressive weathering conditions such as those in the southeastern United States. The temperate climate and relatively high seasonal rainfall produce leached soils with relatively low metal concentrations. The mineralogy of the sands primarily consists of quartz with some feldspar. The mineralogy of the clays is dominated by kaolonite (an aluminum rich clay), which results from the highly weathered feldspars and muscovite (Looney et al. 1990, Nystrom et al. 1991).

Many different soils exist on the SRS, and, in some areas change within a short distance. SRS soils range from seasonally wet and hydric to well-drained. Composition ranges from mostly sand-sized particles with high hydraulic conductivity rates to high clay content with moderately low to low hydraulic conductivity rates. These differences, where the areas are large enough, are shown as a soil series within a mapping unit. A mapping unit is an area dominated by one major kind of soil accompanied by other similar soils. Four orders and 28 soil series are recognized on the SRS. A soil series classification describes the soil's history, evolution, and current characteristics. Detailed soils maps have been constructed for the SRS, and can be found in Rogers (1990).

A more general map of the SRS (Figure 3-8), also from Rogers (1990), shows broader soil-association groups, which are named to coincide with the major soil series within the group (Wike et al. 2006). Each association has a distinctive pattern of soils, relief, and drainage, and a unique natural landscape (Rogers 1990). The SRS is comprised of the seven major soil associations shown, which are described briefly (after Wike et al. 2006).

3.2.5.2.1 <u>Chastain-Tawcaw-Shellbluff Association</u>

The Chastain-Tawcaw-Shellbluff association consists of nearly level soils on the floodplains along the major streams, mainly along the Savannah River. This association makes up about 6% of the survey area. The association is about 60% Chastain soils, 20% Tawcaw soils, 15% Shellbluff soils, and 5% soils of minor extent.

Chastain soils are poorly drained and clayey to a depth of about 100 cm (40 in). Tawcaw soils are somewhat poorly drained and clayey in the upper part and loamy in the lower part. Shellbluff soils are well-drained and loamy to a depth of about 100 cm (40 in).

All the acreage in this association is wooded. The soils are suited for timber production, but not for sanitary facilities or building sites because of flooding and wetness.

3.2.5.2.2 Rembert-Hornsville Association

The Rembert-Hornsville association consists of nearly level soils on stream terraces. Rembert soils are poorly drained, and Hornsville soils are moderately well-drained. A few dirt roads and a railroad cross areas of this association. This association makes up about 7% of the survey area. It is about 30% Rembert soils, 18% Hornsville soils, and 52% other soils of minor extent.

Most of the acreage in this association is woodland. The soils are well suited for timber production. Generally, these soils are poorly suited for sanitary facilities and building sites because of wetness and slow permeability.

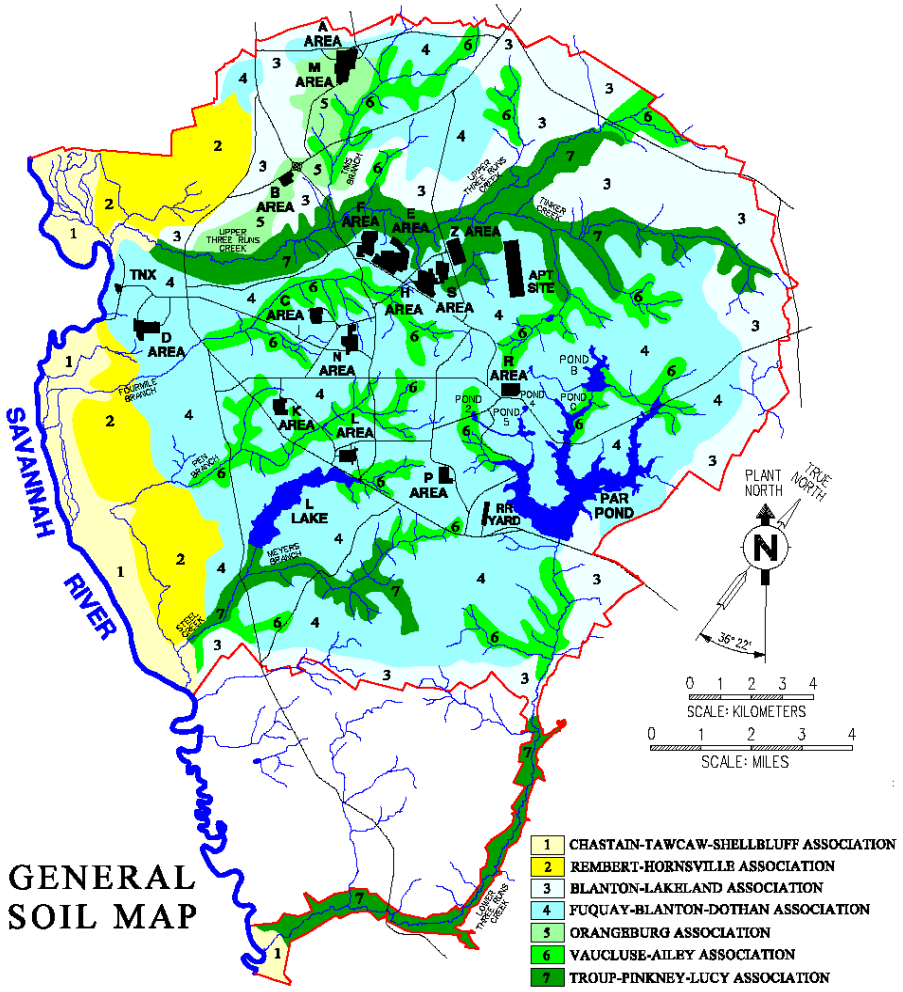


Figure 3-8. SRS General Soils Map (From Rogers 1990)

3.2.5.2.3 Blanton-Lakeland Association

The Blanton-Lakeland association consists of nearly level to sloping soils on uplands. Stands of longleaf (*Pinus palustris*) and loblolly pine (*P. taeda*) with an oak (*Quercus* spp.) understory grow on the broad sandy ridges. This association makes up about 18% of the survey area. It is about 40% Blanton soils, 20% Lakeland soils, and 40% other soils of minor extent.

Blanton soils are somewhat excessively drained. They have thick sandy surface and subsurface layers and loamy subsoil that is 100-200 cm (40-80 in) below the surface. These soils are commonly adjacent to more poorly drained soils. Lakeland soils are excessively drained and are sandy throughout. They are generally higher on the landscape than Blanton soils.

These soils are suited for timber production of species associated with drier sites. Because of the thick surface layer and subsurface layer, these soils are only fairly well suited for sanitary facilities. In most areas the soils are suitable for building sites. The more sloping areas, however, are not as well suited.

3.2.5.2.4 Fuquay-Blanton-Dothan Association

The Fuquay-Blanton-Dothan association consists of nearly level to sloping, well-drained soils on all of the broad upland ridges in the survey area, except for those in the northeastern section of the SRS. This association makes up 47% of the survey area. It is about 20% Fuquay soils, 20% Blanton soils, 12% Dothan soils, and 48% other soils to a minor extent.

Fuquay soils are well-drained. They have moderately thick, sandy surface and subsurface layers and loamy subsoil that contains iron-rich brittle nodules of plinthite. Blanton soils are somewhat excessively drained. Dothan soils are well-drained. They have thick, sandy surface and subsurface layers. They have loamy subsoil that contains iron-rich nodules of plinthite.

These soils are suited for cultivated crops and are well suited for timber production. Most are suited for sanitary facilities. The sandy soils have moderate or severe limitations affecting some sanitary facilities. Most of the soils are suitable for building sites. The more sloping soils, however, are not as well suited.

3.2.5.2.5 Orangeburg Association

The Orangeburg association consists mainly of soils on broad upland ridges and in nearly level to sloping areas northwest of UTR. Slopes generally are smooth. Planted loblolly pine is the dominant vegetation on this soil type. This association makes up about 2% of the survey area. It is about 70% Orangeburg soils and 30% other soils of minor extent. The Orangeburg soils have friable, red, loamy subsoil. The soils of this association are well suited for woodlands, sanitary facilities, building sites, wildlife habitat, and other uses.

3.2.5.2.6 Vaucluse-Ailey Association

The Vaucluse-Ailey association consists of sloping and strongly sloping soils in scattered areas around the head and sides of small drainage ways in the uplands. The areas are long and narrow. The vegetation on this association is mixed pine and hardwoods. This association makes up about 10% of the SRS. It is about 25% Vaucluse soils, 15% Ailey soils, and 60% other soils of minor extent.

Vaucluse soils have a loamy surface layer and subsurface layer that have a combined thickness of less than 50 cm (20 in). Ailey soils have moderately thick, sandy surface and subsurface layers. Both soils have loamy subsoil with a brittle layer.

The soils are fairly well suited for timber production. Because of slow permeability, these soils are poorly suited for sanitary facilities. The soils are poor building sites because of the slope.

3.2.5.2.7 Troup-Pickney-Lucy Association

The Troup-Pickney-Lucy association consists of moderately steep and steep soils on uplands and nearly level soils on the floodplains along streams. The steeper areas are on the southeast bank of UTR and along both sides of Tinker Creek. Areas of this association are long and narrow. This association has soils with the steepest slopes on the SRS. The soils on the floodplains have a higher organic content than the other soils. Their vegetation is mostly hardwoods mixed with loblolly pine. This association makes up about 10% of the survey area. It is about 45% Troup soils, 40% Pickney soils, 10% Lucy soils, and 5% other soils of minor extent.

Troup soils are well drained. They have thick, sandy surface and subsurface layers and loamy subsoil at a depth of 100-200 cm (40-80 in). Pickney soils are poorly drained. They have a thick black surface soil and are sandy throughout. Lucy soils are well drained. They have moderately thick sandy surface and subsurface layers and loamy subsoil at a depth of 50-100 cm (20-40 in).

The soils are fairly well suited for woodlands. These soils generally are poorly suited for sanitary facilities or building sites because of the steep slope and the flooding; however, some areas of more moderately sloping soils are available for such uses. The soils on floodplains are not suited for building sites.

3.2.5.3 General Characteristics of Savannah River Site Soils

Soil texture is the relative proportion of sand, silt, and clay particles in a soil mass. The basic textural classes, in order of increasing proportion of fine particles, are sand, loamy sand, loam, silt loam, silt, sandy clay loam, clay loam, silty clay loam, sandy clay, and clay. The most common surface textures on the SRS are sand and loamy sand, while the subsoil textures are usually sandy loam or sandy clay loam (Wike et al. 2006). Vadose zone soils for E-Area and Z-Area typically fall into these categories (Phifer et al. 2006).

A notable feature of the "Upland Unit" is its compositional variability. This formation predominantly consists of red-brown to yellow-orange, gray, and tan colored, coarse to fine grained sand, pebbly sand with lenses and beds of sandy clay and clay. Generally vertically upward through the unit, sorting of grains becomes poorer, clay beds become more abundant and thicker, and sands become more argillaceous and indurated. In some areas, small-scale joints and fractures, both of which are commonly filled with sand or silt, traverse the unit. The mineralogy of the sands and pebbles primarily consists of quartz, with some feldspars. In areas to the east-southeast, sediments may become more phosphatic and dolomitic (DOE 2002, pp. 3-1 and 3-5).

3.2.5.4 Inorganic and Radionuclide Concentrations in Savannah River Site Soils

Ranges of inorganic and radionuclide concentrations in soils from the SRS are compared to ranges for the southeastern United States and the nation. Data from Looney et al. (1990) and from investigations at SRS waste sites as part of the environmental remediation program were compared statistically. Differences between the two data sets were essentially insignificant; therefore, the data sets were combined to produce SRS-wide range of soil constituent concentrations (Table 3-3).

3.2.6 Geology, Seismology and Volcanology

3.2.6.1 Regional and Site-specific Geology/Topography

3.2.6.1.1 Topography

The elevation of the SRS ranges from 80 ft above mean sea level (msl) at the Savannah River to about 420 ft above msl in the upper northwest portion of the site (Figure 3-9) (USGS 1987, DOE 2002, WSRC 2000a). The Aiken Plateau and the Pleistocene Coastal terraces comprise two distinct physiographic subregions of the Upper Belt of the Atlantic Coastal Plain Province at the SRS (Denham 1999, McAllister et al. 1996). Figure 3-10 shows the location of the SRS relative to the extent of the Aiken Plateau region of the Upper Coastal Plain.

Elevations of the Aiken Plateau, on which most of the SRS lies, range from about 270 to 420 ft above msl. The Aiken Plateau slopes gently to the southeast and is generally well drained except where poorly-drained Carolina bays are present. The relatively flat Aiken Plateau is deeply dissected by local streams. The Savannah River on the southwestern margin of the site is the principal regional drainage basin, serving as the local base-level for the SRS tributaries that drain the site. These tributaries enter the Savannah River at almost right angles. Of the major tributaries on the site, only UTR has its headwaters upgradient from the site. The remaining major tributaries (Fourmile Branch, Pen Branch, Steel Creek, and LTR) have their origins on the SRS (Figure 3-11). Pen branch is not a direct tributary to the Savannah River but discharges into Steel Creek approximately 0.5 mi upstream of the Steel Creek mouth to the Savannah River. (See Section 3.2.9.2.1.4.)

Table 3-3. Metal and Radionuclide Concentration Ranges for the Southeastern United States, Nationwide, and SRS

Constituents	Southeastern U.S.[a]	Nationwide[b,c]	SRS
Metals (mg/kg)			
Aluminum	900 to 46,000	700 to 100,000	715 to 53,530
Antimony	N/A	N/A	<0.21 to 20
Arsenic	N/A	<0.1 to 97	<0.25 to 15.2
Barium	63 to 350	10 to 5,000	0.33 to 3,210
Beryllium	N/A	<1 to 15	0.02 to 5.9
Cadmium	N/A	N/A	<0.098 to 7.57
Chromium	11 to 60	1 to 2,000	<0.41 to 116
Iron	500 to 21,000	100 to >100,000	635 to 79,600<
Lead	2.8 to 26	<10 to 700	<0.16 to 35
Manganese	100 to 410	<2 to 7,000	<0.6 to 566
Mercury	N/A	<0.01 to 4.6	<0.0028 to 0.89
Nickel	2.8 to 18	<5 to 700	<0.1 to 228
Selenium	N/A	<0.1 to 4.3	<0.084 to 29.8
Thallium	N/A	N/A	<0.84 to 49.83
Vanadium	N/A	<7 to 500	<2.25 to 61
Zinc	<25 to 64	<5 to 2,900	<0.23 to 267
Radionuclides (pCi/g)			
Carbon-14	N/A	0.01 to 2.5	<0.0011 to 0.17
Cesium-137	N/A	0.01 to 3.5	<0.0003 to 2.21
Iodine-129	N/A	1E-05 to 9E-05	<7.69E-7 to <32.2
Potassium-40	N/A	3 to 20	0.066 to 11.6
Plutonium-239/240	N/A	0.009 to 0.04	<0.0016 to 4.11
Strontium-90	N/A	0.2 to 4.0	<0.01 to 13.2
Technetium-90	N/A	N/A	<0.0016 to 9.76
Thorium-232 and daughters			
Thorium-232	N/A	0.10 to 3.4	0.31 to 2.53
Actinium-228	N/A	N/A	<0.01 to 2.54
Radium-228	N/A	0.1 to 3.4	0.34 to 2.9
Thorium-228	N/A	N/A	0.21 to 17.9
Lead-212	N/A	N/A	0.013 to 3.2
Uranium-238 and daughters			
Uranium-238	N/A	0.12 to 3.8	0.18 to 2.42
Uranium-234	N/A	0.12 to 3.8	1.0 to 1.2
Thorium-230	N/A	0.12 to 3.8	0.18 to 2.27
Radium-226	N/A	0.23 to 4.2	0.19 to 2.03
Uranium-235	N/A	0.01 to 0.05	<0.0019 to 0.13
Gross alpha	N/A	N/A	<0.01 to 44.53
Nonvolatile beta	N/A	N/A	<0.01 to 54.82

Notes:
Source: PRC 1996.
N/A = Not Available.
mg/kg = Milligram per kilogram.
pCi/g = Picocuries per gram.
[a]Connor and Shacklette 1975.
[b]Shacklette and Boerngen 1984 (for inorganics).
[c]EPA 1994 (for radionuclides).

The Pleistocene Coastal terraces are below 270 ft above msl in elevation, with the lowest terrace constituting the present flood plain along the Savannah River and the higher terraces characterized by gently rolling terrain. The modern floodplain of the Savannah River (Figure 3-11) is approximately 2 to 4 km wide at the site boundary (Denham 1999). The two principal terraces evident on the SRS are the Ellenton Terrace, which stands about 17 to 25 m above the Savannah River, and the Bus Field Terrace, which stands about 8 to 13 m above the river.

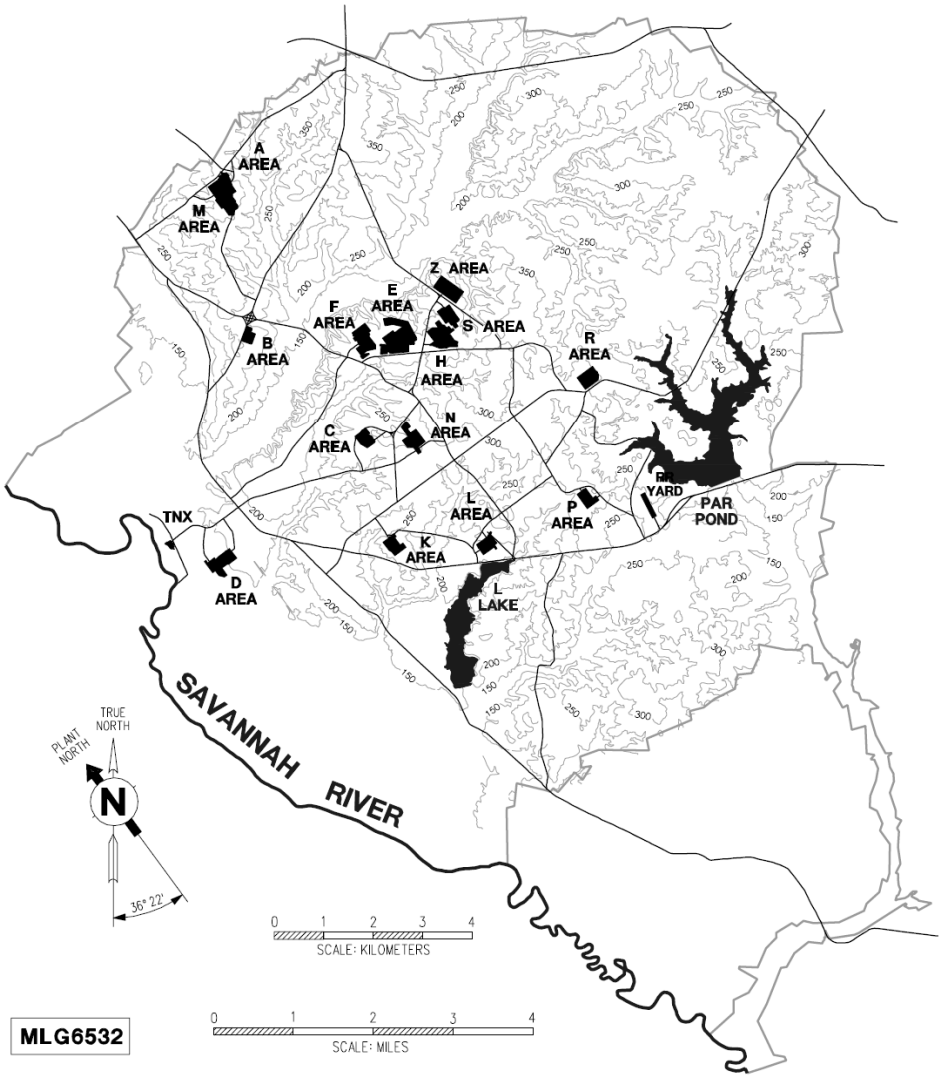


Figure 3-9. Topographic Map of the SRS (from WRSC 2000a)

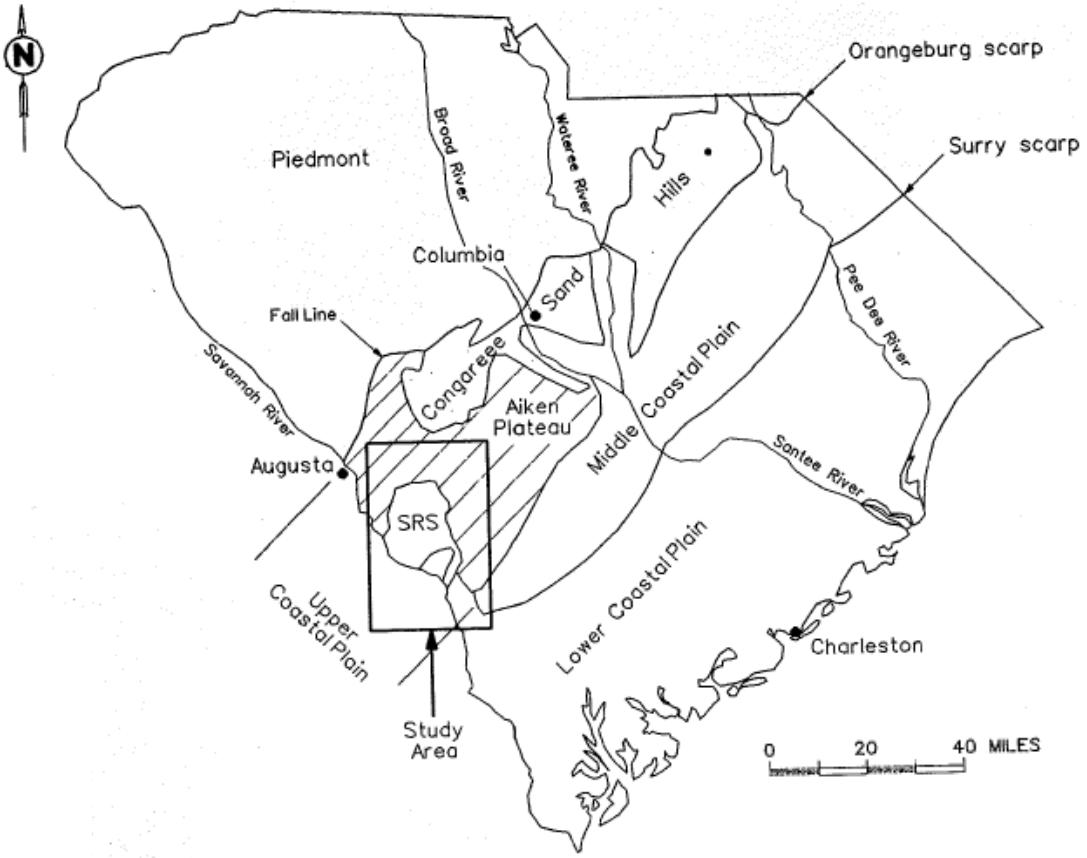


Figure 3-10. Location of the SRS and the Physiography of the Surrounding Region (from Aadland et al. 1995)

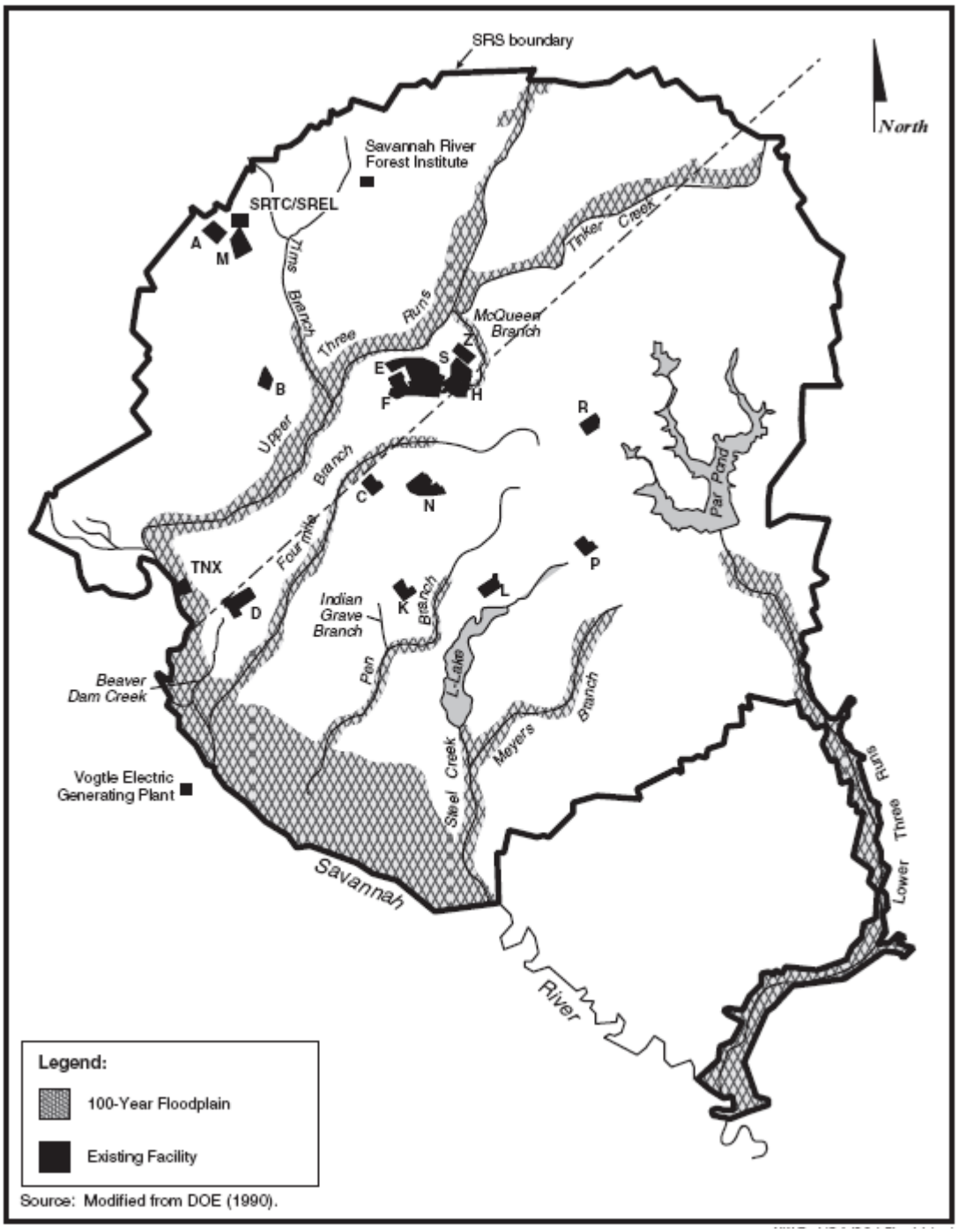


Figure 3-11. Major Tributaries and Floodplain of the Savannah River on the SRS (DOE 2002)

3.2.6.1.2 Geology

The Atlantic Coastal Plain consists of a southeast-dipping wedge of unconsolidated and semi-consolidated sediments, which extends from its contact with the Piedmont Province at the Fall Line to the continental shelf edge. Sediments range in geologic age from Late Cretaceous to Recent and include sands, clays, limestones and gravels. This sedimentary sequence ranges in thickness from essentially zero at the Fall Line to more than 1,219 m at the Atlantic Coast (Siple 1967).

On the SRS, the coastal plain sediments thicken from approximately 213 m at the northwestern boundary to about 430 m at the southeastern boundary of the site (Fallaw et al. 1990) and form a series of aquifers and confining or semi-confining units. Figure 3-12 shows a generalized cross section of the sedimentary strata and their corresponding depositional environments for the Upper Coastal Plain down-dip through the SRS into the Lower Coastal Plain. Figure 3-13 shows the regional lithologic units and their corresponding hydrostratigraphic units at the SRS.

Descriptions of the primary sedimentary units and surface soils are provided below. More detailed descriptions of the geology of the SRS can be found in several historical and recent reports (Aadland et al. 1995, Colquhoun et al. 1983, Denham 1999, Fallaw et al. 1990, Fallaw and Price 1995, Logan and Euler 1989, Nystrom et al. 1991, Siple 1967, Wyatt and Harris 2004).

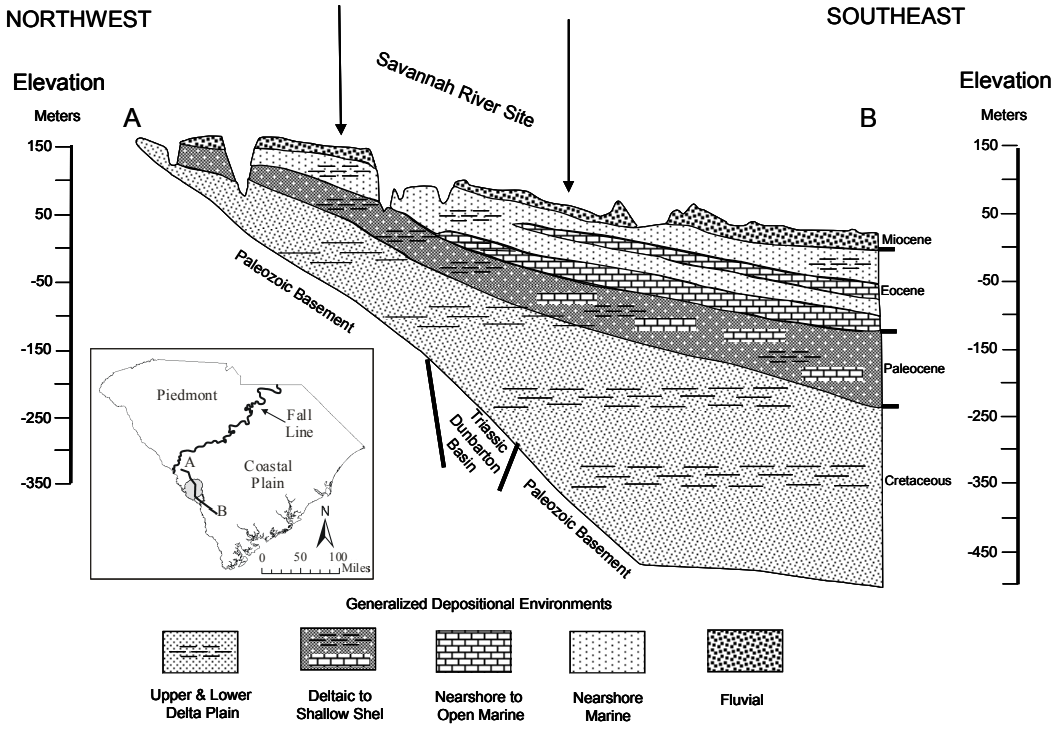


Figure 3-12. Regional NW to SE Cross-Section (from Wyatt and Harris 2004)

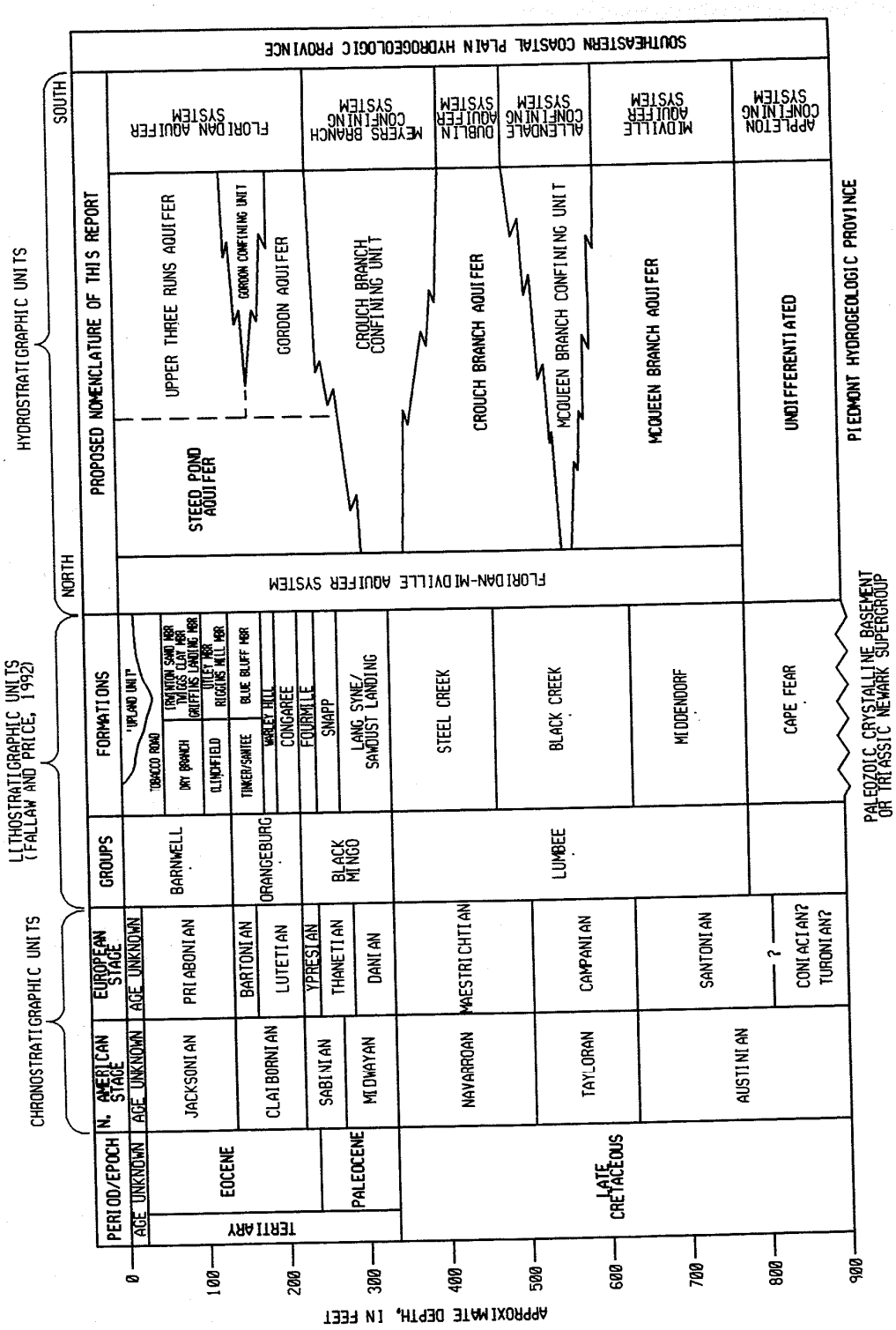


Figure 3-13. Comparison of Chronostratigraphic, Lithostratigraphic and Hydrostratigraphic Units in the SRS Region (from Aadland et al. 1995)

3.2.6.1.2.1 Late Cretaceous Sediments

The Late Cretaceous sediments include, from oldest to youngest, the Cape Fear Formation and the three formations of the Lumbee Group: the Middendorf, Black Creek, and Steel Creek Formations (Figure 3-13). These sediments are approximately 210 m thick at the center of the SRS. At A/M area, these sediments are approximately 130 m thick (Lewis and Aadland 1994). The lowermost Cape Fear Formation rests on a thin veneer of saprolite (weathered rock). It defines the surface of the Paleozoic crystalline basement rock or lies on top of weathered Triassic Basin sedimentary material of the Newark Supergroup in limited areas associated with Triassic-age basement rock faulting. This formation is composed of poorly sorted silty-to-clayey quartz sands and interbedded clays. Bedding thicknesses range from 1.5 to 6 m, with sand beds being thicker than clay beds. The formation is about 9 m thick at the northwestern boundary of the SRS, and it increases to more than 55 m near the southeastern boundary. This formation has not been observed to outcrop in the vicinity of the SRS.

The thickness of the Lumbee Group, which overlies the Cape Fear Formation, varies across the SRS from 120 m in the northwest to more than 230 m near the southeastern boundary. The Middendorf Formation, which directly overlies the Cape Fear Formation, is composed mostly of medium and coarse quartz sand that is cleaner and less indurated than the underlying sediments. Clay casts and pebbly zones occur in several places in the Middendorf Formation. A clay zone up to 24 m thick forms the top of this formation over much of the SRS. In total, the Middendorf Formation ranges from approximately 37 to 73 m thick from the northwestern to southeastern boundary of the SRS (Wyatt and Harris 2000). Outcrops of this formation have been identified northwest of the SRS.

The Black Creek Formation consists of quartz sands, silts, and clays. The lower section consists of fine- to coarse-grained sands with layers of pebbles and clay casts. The upper section changes in composition as it crosses the SRS from northwest to southeast, from massive clay to silty sand with interbeds of clay. Thickness of the Black Creek Formation under the SRS ranges from 34 m in the northwest to 76 m in the southeast. Outcropping in the vicinity of the SRS has not been confirmed. The Black Creek is distinguished from the over-and underlying Cretaceous formations by its better sorted sand, relatively fine-grained texture, and relatively higher clay content.

The uppermost formation in the Lumbee Group is the Steel Creek Formation (previously referred to as the Peedee Formation), which consists of fine-grained sandstone and siltstone with marine fossils. This formation is comparable in age, but lithologically distinct, from the Peedee Formation in southwestern South Carolina. The lower portion of this formation consists of fine- to coarse-grained quartz sand and silty sand, with a pebble-rich zone at its base. Pebbly zones and clay casts are common throughout the lower portion of the Steel Creek Formation. The upper portion of this formation is a clay that varies from more than 15 m to less than 1 m in thickness at the SRS. The Steel Creek Formation is about 18 m thick toward the northwestern SRS boundary and about 53 m thick toward the southeastern boundary (Wyatt and Harris 2000). No nearby outcropping has been identified.

3.2.6.1.2.2 Paleocene-Eocene Black Mingo Group

Paleocene-Early Eocene sediments make up the Black Mingo Group (Figure 3-13). This group consists of the Early Paleocene Lang Syne/Sawdust Landing Formations, the Late Paleocene Snapp Formation, and the Early Eocene Four Mile Formation. This group is about 21 m thick at the northwestern SRS boundary, and thickens to about 46 m near the southeastern boundary.

The Lang Syne/Sawdust Landing Formations together are equivalent to the lithologic unit previously referred to as the Ellenton Formation. These formations are treated as a single unit due to difficulty in mapping them separately and consist mostly of gray, poorly sorted, micaceous, lignitic, silty and clayey quartz sand interbedded with gray clays. They are approximately 12 m thick at the northwestern boundary of the SRS and thicken to about 30 m near the southeastern boundary. These formations outcrop about four mi northwest of the SRS.

The deposits near the SRS that are designated as the Snapp Formation are time-equivalent to the Williamsburg Formation. The sediments are typically silty, medium- to coarse-grained quartz sand interbedded with clay. The Snapp Formation pinches toward the northwestern SRS boundary and thickens to about 15 m near the southeastern boundary. In the central SRS, the distribution of the Snapp Formation is sporadic and not continuous.

Sand immediately overlying the Snapp Formation is identified as the Four Mile Formation. The well-sorted sand of this formation is an average of 9 m in thickness. Clay beds near the middle and top of the formation are a few feet thick. This formation may not be continuous in all areas of the SRS.

3.2.6.1.2.3 Middle Eocene Orangeburg Group

The middle Eocene sediments make up the Orangeburg Group, which consists of the lower middle Eocene Congaree Formation, the upper middle Eocene Warley Hill Formation, and the late middle Eocene Tinker/Santee Limestone Formation (Figure 3-13). The sediments thicken from about 30 m at the northwestern SRS boundary to about 49 m near the southeastern boundary. The dip of the upper surface of this formation is about 2 m/km to the southeast across the site. The Orangeburg Group is about 100 m thick at the coast. The group outcrops at lower elevations in many places near and at the SRS.

The Congaree Formation consists of fine to coarse, well-sorted and rounded quartz sands. Thin clay laminae occur throughout, as do small pebble zones. The sand is glauconitic in places. The formation is up to 26 m thick at the center of the SRS, and approximately 18 m thick at A/M Area.

The Warley Hill Formation, made up of glauconitic sand and green clay beds and thus previously referred to locally as the "green clay," overlies the Congaree Formation. This formation is generally 3 to 6 m in thickness. However, northwest of the central SRS, the Warley Hill Formation is missing or very thin, such that the overlying Tinker/Santee Formation rests unconformably on the Congaree Formation.

The Tinker/Santee Formation consists of calcilutite, calcarenite, shelly limestone, calcareous sands and clays, and micritic limestone. The sands are glauconitic in places and fine- to medium-grained. The sediments comprising this formation have been referred to in the past as the Santee Limestone, McBean, and Lisbon Formations and indicate deposition in shallow marine environments. The McBean Member (Figure 3-13), consisting of tan to white, calcilutite, calcarenite, shelly limestone, and calcareous sand and clay, dominates the Santee in the central SRS region (WSRC 2000a). The Blue Bluff Member (Figure 3-13) consists of fine-grained carbonates, and is found in the southern SRS region (WSRC 2000a). The Tinker/Santee Formation is about 12 to 15 m thick in the center of the SRS. In places where the Warley Hill Formation is absent, the Tinker/Santee Formation rests directly on the Congaree Formation.

3.2.6.1.2.4 Late Eocene Barnwell Group

The Late Eocene sediments make up the Barnwell Group, which consists of the Clinchfield, Dry Branch, and Tobacco Road Sand (Figure 3-13). The Clinchfield Formation, the oldest of the three, is made up of quartz sand, limestone, calcareous sand, and clay. It is generally identified only when the contrasting carbonates of the overlying Dry Branch and underlying Tinker/Santee Formations are present with the sand of the Clinchfield Formation sandwiched between them. It has been identified at several areas within the SRS where it is up to 8 m thick, but it is indistinguishable in the central regions of the SRS, and in A/M area. Sand beds constitute the Riggins Mill Member of the Clinchfield Formation and the carbonate sequence is designated the Utley Limestone Member (WSRC 2000a).

The Dry Branch Formation consists of three distinguishable members: the Twiggs Clay Member, the Griffins Landing Member, and the Irwinton Sand Member. The Twiggs Clay Member is not mapable as a continuous unit within the SRS, but lithologically similar clay is present at various levels within this formation. The tan, light gray, and brown clay of the Twiggs Clay Member has previously been referred to as the "tan clay" at the SRS. The Griffins Landing Member is up to 15 m thick in the southeastern part of the SRS. This member consists mostly of calcilutite and calcarenite, calcareous quartz sand, and slightly calcareous clay. It occurs sporadically and pinches out in the center of the SRS. The remainder of the Dry Branch Formation within the SRS is made up of the Irwinton Sand Member, which is composed of moderately sorted quartz sand with interlaminated clays, which are abundant in places. Clay beds of this member have also been referred to as the "tan clay" at the SRS. The Irwinton Sand is about 12 m thick at the northwestern SRS boundary and thickens to 21 m near the southeastern boundary. It outcrops in many places around and within the SRS.

The Tobacco Road Sand overlies the Dry Branch Formation. This formation consists of moderately to poorly sorted quartz sands, interspersed with pebble layers and clay laminae. The sediments have the characteristics of a lower Delta plain to shallow marine deposits. The upper surface of this formation is irregular due to an incision that accompanied deposition of the overlying "Upland Unit" and later erosion. The thickness is variable as a result of erosive processes, but is at least 15 m in places.

3.2.6.1.2.5 Upland Unit

The "Upland Unit" is an informal stratigraphic term applied to terrestrial deposits that occur at higher elevations in some places in the southwestern South Carolina Coastal Plain (Figure 3-13). This unit overlies the Tobacco Road Formation in the Upper Coastal Plain of western South Carolina, on which the SRS is located. This unit occurs at the surface at higher elevations in many places around and within the SRS, but it is not present at all higher elevations. The sediments are poorly sorted, clayey-to-silty sands, with lenses and layers of conglomerates, pebbly sands, and clays. Clay casts are abundant. The "Upland Unit" is up to 21 m thick in parts of the SRS. The depositional environment is thought to be fluvial and the thickness changes abruptly due to the channeling of the underlying Tobacco Road Formation during the deposition of the "Upland Unit" and the subsequent erosion of the "Upland Unit."

3.2.6.2 Seismology

Throughout the fifty-plus year existence of SRS, extensive regional and site specific geological and seismological investigations have been performed by many organizations in support of Site operations and missions. This section provides a short summary of area seismic studies. Characterization of regional seismicity is dominated by the catastrophic Charleston, SC, earthquake of August 31, 1886, with a retrospectively estimated Richter scale magnitude of 7.5 (DOE 2002). With over three centuries of available historic and contemporary seismic data (since 1698), the Charleston-Summerville area remains the most seismically active region of South Carolina and the most significant seismogenic region affecting the SRS (WSRC 2000a). Other broad regions of South Carolina have also experienced seismic activity, but at very low levels with magnitudes or sizes generally less than or equal to 3 on the Richter scale.

The historical database for the region is essentially composed of two data sets: the first is comprised of mostly qualitative data (1698 - 1974), and the second set covers the instrumentally recorded seismicity (1974 - present) (WSRC 2000a). The qualitative data consists of intensity data, and the scale generally used in North America is the Modified Mercalli Intensity (MMI) scale Table 3-4).

Table 3-4. Modified Mercalli Intensity Scale of 1931

Level	Definition
I.	Not felt except by a very few under especially favorable circumstances (I Rossi-Forel Scale).
II.	Felt by only a few persons at rest, especially on upper floors of buildings. Delicately suspended objects may swing (I and II, Rossi-Forel Scale).
III.	Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motor cars may rock slightly. Vibration like passing truck. Duration estimated (III Rossi-Forel Scale).
IV.	During the day felt indoors by many; outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls made creaking sound. Sensation like heavy truck striking building. Standing motor cars rocked noticeably (IV to V Rossi-Forel Scale).
V.	Felt by nearly everyone; many awakened. Some dishes, windows, etc., broken, a few instances of cracked plaster, unstable objects overturned. Disturbance of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop (V to VI Rossi-Forel Scale).
VI.	Felt by all; many are frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight (VI to VII Rossi-Forel Scale).
VII.	Everybody runs outdoors. Damage negligible in buildings of good structures; considerable in poorly built or badly designed structures; some chimneys are broken. Noticed by persons driving motor cars (VIII Rossi-Forel Scale).
VIII.	Damage slight in specially designed structures; considerable in ordinary substantial buildings with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, and walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Disturbs persons driving motor cars (VIII+ to IX Rossi-Forel Scale).
IX.	Damage considerable in specially designed structures; well designed frame structures thrown out of plumb; great in substantial buildings with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken (IX+ Rossi-Forel Scale).
X	Some well built wooden structures destroyed; most masonry and frame structures destroyed with foundations, ground badly cracked. Rails bent. Landslides considerable from riverbanks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks (X Rossi-Forel Scale).
XI.	Few, if any, masonry structures remain standing. Bridges destroyed. Broad fissures in ground. Underground pipe lines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.
XII.	Damage total. Waves seen on ground surfaces. Lines of sight and level distorted. Objects thrown upward into the air.

3.2.6.2.1 Description of Nearby Seismic Zones and Earthquakes

Results obtained from seismic network data within South Carolina enabled Tarr et al. (1981) to identify two diffuse areas of seismic activity: the Lower Coastal Plain and the Piedmont/Upper Coastal Plain. Through these studies the Lower Coastal Plain area was further subdivided into three distinct clusters or zones of seismicity that include the Bowman Seismogenic Zone, the Middleton Place-Summerville Seismic Zone, and the Jedburg-Adams Run Seismogenic Zone. The most active zone is the Middleton Place-Summerville Seismic Zone, which is the only one to coincide with the meizoseismal area of the 1886 Charleston earthquake. The main two zones, the Middleton Place-Summerville Seismic Zone and the Bowman are discussed in more detail.

3.2.6.2.1.1 Lower Coastal Plain

Middleton Place-Summerville Seismic Zone

The Charleston, SC, area, in particular the Middleton Place-Summerville Seismic Zone, is the most significant source of seismicity affecting SRS in terms of both the maximum historical site intensity and the number of earthquakes recorded at SRS. The earthquake with the greatest intensity recorded at the SRS has been estimated at a MMI of VI-VII and was produced by the intensity MMI X earthquake that struck Charleston, SC, on August 31, 1886. Some of the larger aftershocks of the 1886 Charleston event were also documented in the Aiken-SRS area with intensities equal to or less than MMI IV (Visvanathan 1980)

Initially, Talwani (1985) identified the delineation of two possible intersecting faults in the Charleston area. The first was a shallow, northwest-trending fault striking parallel to the Ashley River, which was named the Ashley River fault. The second fault was labeled the Woodstock fault. The Woodstock fault trends north-northeasterly and is defined by the planar distribution of hypocenters. It intersects and appears deeper than the Ashley River fault.

Recent studies by Madabhushi and Talwani (1993) refined the earlier model by utilizing 58 additional well-recorded events in the Middleton Place-Summerville Seismic Zone from 1980 to 1991. Results of this effort demonstrated that the Ashley River and the Woodstock faults are not simple planar features, but resemble zones composed of short segments of varying strike and dip. Madabhushi and Talwani (1993) concluded that the seismicity in the Middleton Place-Summerville Seismic Zone defines the intersection of two fault zones, which they inferred to be the Ashley River fault zone and the Woodstock fault zone.

Bowman Seismic Zone
The Bowman Seismic Zone located 95 km northeast of SRS was initially defined by Tarr et al (1981) from the location of a series of earthquakes occurring through the 1970s. The largest event occurred February 3, 1972 with a reported Richter scale magnitude of 4.5 (or an MMI V). The SRS area is estimated to have felt this event with an intensity of MMI III-IV. The Bowman area has experienced very low level sporadic activity since that time with only four small events (Richter scale magnitude less than or equal to 2.3) recorded from 1980 through 2005. The last recorded event occurred in 1997 (Richter scale magnitude of 2.25).

3.2.6.2.1.2 Piedmont/Upper Coastal Plain

The second diffuse area of seismic activity includes the Piedmont province and the Upper Coastal Plain. The Upper Coastal Plain is where the majority of the SRS is located. The structure and configuration of Upper Coastal Plain basement features have been interpreted to be the subsurface continuation of Piedmont terrain (Daniels et al. 1983). Earthquake activity occurring within this area, not associated with reservoir-induced activity, shows a lack of clustering and is best characterized by occasional small shallow events associated with strain release near small-scale faults, intrusives, and edges of metamorphic belts (Tarr et al. 1981).

Regional Earthquake Activity
On January 1, 1913, an earthquake struck Union County, SC, approximately 100 mi north-northeast of the SRS. Outside of the Charleston area, this was the largest recorded event near the SRS. It had intensity greater than or equal to MMI VIII. In the Aiken-SRS area, this earthquake was felt with an intensity of MMI II-III (Visvanathan 1980). The most recent estimate of the Richter scale magnitude of this earthquake is 4.5 (DOE 2002).

Earthquake Activity within a 50-mile radius of the SRS
As stated above, the SRS is located within the Coastal Plain physiographic province of South Carolina. However, seismic activity associated with the SRS and the surrounding region displays characteristics more closely associated with the Piedmont province. Epicentral locations for events near the SRS (within 50-mi radius of the site) are presented in Table 3-5. Figure 3-14 shows the distribution of earthquake epicenters within 50 mi of the SRS. No damage or injury has ever been associated with any earthquake activity occurring within the 50-mi radius. The largest event in this series was the magnitude 3.7 Clarks Hill event of November 5, 1974 (event #7 on Table 3-5 and Figure 3-14). It occurred near Clarks Hill Reservoir (now Strom Thurmond Lake) and is attributed to Reservoir Induced Seismicity.

Recent Earthquake Activity at the SRS
On June 9, 1985, an intensity MMI III earthquake with a local duration magnitude of 2.6 occurred at SRS (Talwani et al. 1985). Duration magnitude is a method of estimating magnitudes of local earthquakes from signal duration. Workers at the western edge of the central portion of the site experienced and reported the event.

Table 3-5. Historic and Instrumental Earthquakes Recorded within 50 mi of the SRS

Event #	Date	Latitude	Longitude	Depth (km)	Magnitude
1	5/6/1897	33.3	-81.2	unknown[1]	Felt
2	5/9/1897	33.9	-81.6	unknown[1]	Felt
3	5/24/1897	33.3	-81.2	unknown[1]	Felt
4	5/27/1897	33.3	-81.2	unknown[1]	Felt
5	8/14/1972	33.2	-81.4	unknown[1]	3.2
6	10/28/1974	33.79	-81.92	unknown[1]	3.0
7	11/5/1974	33.73	-82.22	unknown[1]	3.7
8	9/15/1976	33.0883	-81.4480	12.0	2.4
9	6/5/1977	33.0520	-81.4120	3.50	2.7
10	2/21/1981	33.5933	-81.1476	6.61	2.0
11	1/28/1982	32.9800	-81.3900	7.00	3.4
12	6/9/1985	33.2225	-81.6842	5.81	2.6
13	2/17/1988	33.5113	-81.6966	11.73	2.5
14	8/5/1988	33.1873	-81.6290	2.26	2.0
15	7/13/1992	33.4798	-81.1920	7.60	1.9
16	10/2/1992	33.4990	-81.2020	3.00	2.4
17	12/12/1992	33.2798	-81.8328	11.80	1.2
18	6/29/1993	33.4652	-81.2210	4.90	2.2
19	8/8/1993	33.5893	-81.5852	10.18	3.2
20	8/8/1993	33.5885	-81.5812	9.22	1.6
21	9/18/1996	33.6915	-82.1248	2.38	2.8
22	5/17/1997	33.2118	-81.6765	5.44	2.5
23	10/8/2001	33.3240	-81.6650	3.90	2.6
24	10/8/2001	33.3193	-81.6733	4.19	1.0
25	10/8/2001	33.3317	-81.6762	4.15	1.4
26	10/14/2001	33.3467	-81.6627	3.14	0.7
27	10/15/2001	33.3475	-81.6938	4.67	0.8
28	12/17/2001	33.3283	-81.6745	4.13	1.1
29	12/27/2001	33.3310	-81.6652	3.76	0.1
30	3/6/2002	33.3313	-81.6792	4.61	1.4
31	1/18/2005	33.6063	-82.1631	8.76	2.5
32	1/18/2005	33.5976	-82.1681	15.4 ?	2.3
33	1/18/2005	33.5786	-82.1621	17.4 ?	2.0

Source: SEUSSN Bulletins, Virginia Tech Publication for events through December 2000; SRS unpublished data for events from January 2001 through September 20, 2006; question marks indicate depths that are estimates; numbers on the table correspond to the numbers on the map showing historical earthquakes; magnitudes are expressed in terms of the Richter scale.

[1] Historic data based upon qualitative information rather than instrumentally recorded seismicity.

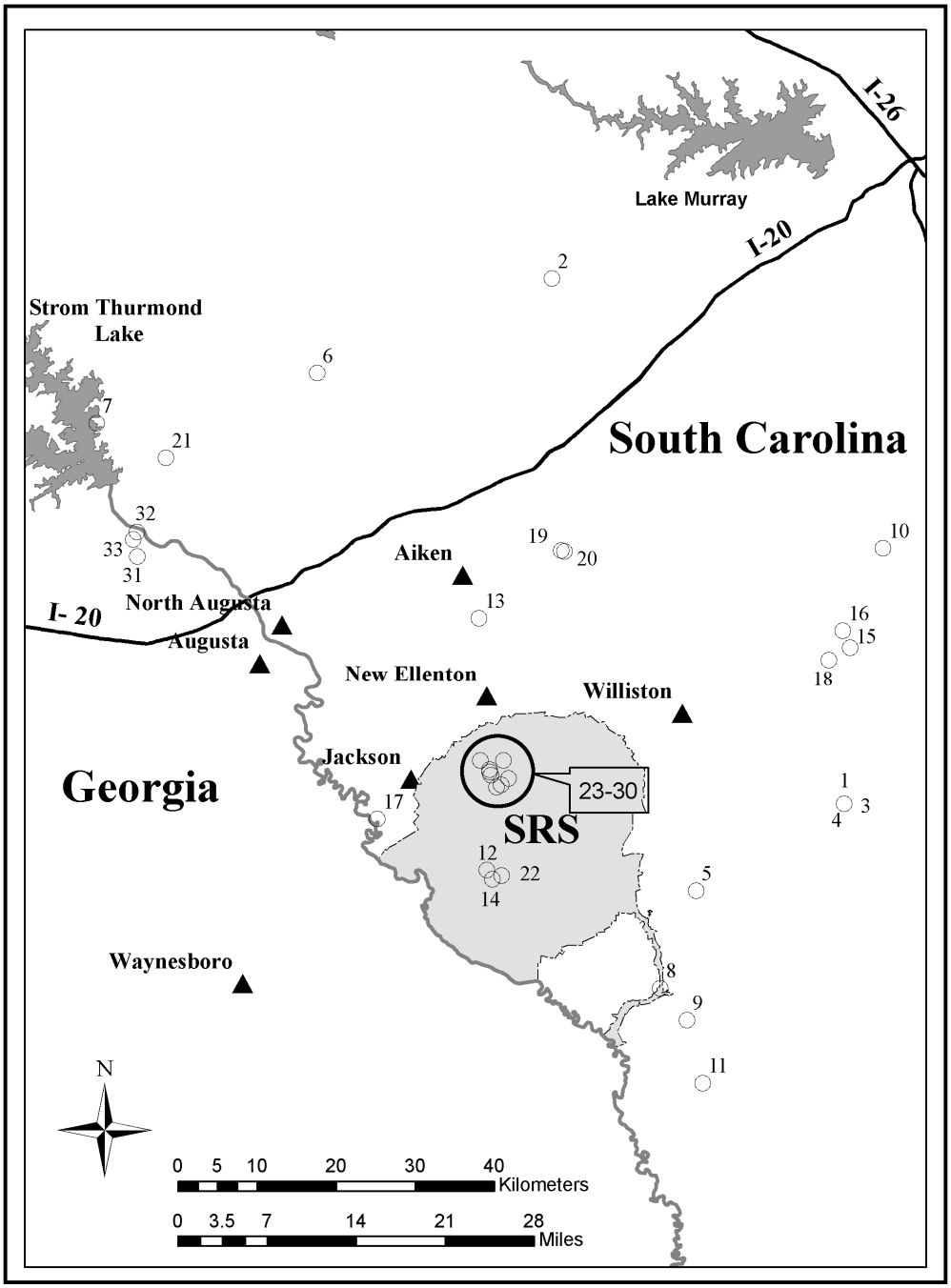


Figure 3-14. Historical Earthquakes within 50-mi Radius of the SRS

Another event occurred at the SRS on August 5, 1988. This earthquake had an intensity of MMI I-II and a local duration magnitude of 2.0. A survey of SRS personnel in 1988 indicated that it was not felt at the site (Stephenson 1988). Neither of these earthquakes triggered SRS seismic alarms, which were set to trigger when ground accelerations equaled or exceeded 0.002 times the earth's gravitational acceleration (set point 0.002 g_o) (Stephenson et al. 1985 and Stephenson 1988).

On the evening of May 17, 1997, an earthquake with a duration magnitude of approximately 2.3 occurred within the boundary of the SRS. Workers in K-Area and Wackenhut guards at a nearby barricade reported the earthquake. No strong motion accelerographs were triggered as a result of this event with the closest instrument located only 3 mi southeast of the epicenter (trigger threshold set at 0.006 g_o).

Eight small earthquakes were recorded and located between October 8, 2001 and March 6, 2002. None of these events was strong enough to trigger strong motion instrumentation installed in facilities throughout SRS. The largest earthquake occurred on October 8, 2001 with a local duration magnitude of 2.6. It was located near UTR in the north central area of the SRS. A series of seven small aftershocks followed the main event with the last one occurring March 6, 2002. Projection of hypocenters onto a nearby seismic reflection line showed no apparent relationship to interpreted basement faults.

Detailed analyses of collected data showed a strong relationship to a small northwest (NW-SE) trending gravity and magnetic feature (Stevenson and Talwani, 2004). This small basement feature runs counter to the regional structure (NW-SE). The shallowness, small aerial extent, and its relationship to a small basement feature running counter to the regional structure indicated that this activity was extremely localized and not related to any large scale regional feature.

Faults at the Savannah River Site

Faults involving Coastal Plain sediments that are considered regionally significant based on their extent and amounts of offset include the ATTA, Crackerneck, Martin, Pen Branch, and Tinker Creek faults shown in Figure 3-15 (Stieve and Stephenson 1995). The Crackerneck and Pen Branch Faults are relatively well constrained with borings. The other faults are projected only from geophysical data and their parameters are less known. DOE (2002) concludes that faults located beneath SRS are not "capable," i.e., have not moved at or near the ground surface within the past 35,000 years or is associated with another fault that has moved within the past 35,000 years.

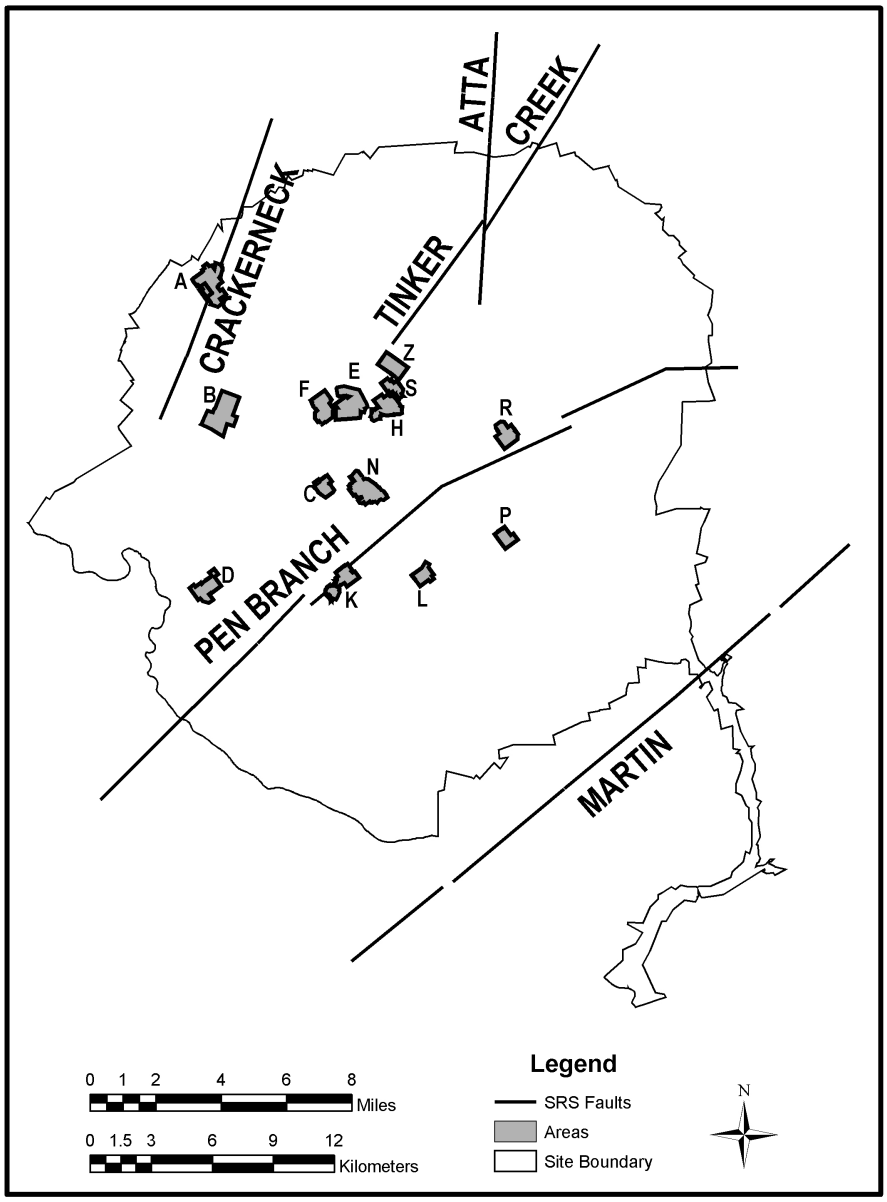


Figure 3-15. Locally Significant Faults at the SRS

Of these faults the Pen Branch fault has been regarded as the primary structural feature at SRS that has the characteristics necessary to pose a potential seismic risk. As a result, the Pen Branch fault has been extensively studied in order to determine the capability of the fault to release potentially damaging seismic energy as defined by NRC regulatory guidelines, 10 CFR 100, Appendix A. Results from the Pen Branch Fault Program showed that the most recent faulting on this fault was older than 500,000 years. In a study designed to examine only the sediments with an age of 1 million years or less, no deformation of the sediments was found to exist (Hanson et al. 1993). In the end, research findings from the program indicated that the Pen Branch Fault was not capable of producing damaging earthquakes (Hanson et al. 1993, Stieve et al. 1991, Stieve et al. 1994, WSRC 2000a).

Contemporary shallow state of stress values gathered from direct in-situ measurements together with focal mechanisms of recent earthquakes have shown a consistent northeast-southwest (NE-SW) direction of maximum horizontal compressive stress (N 55-70°E) for the southeastern United States (Moos and Zoback 1993). Overall, the state of stress for the SRS has been found to agree with these measurements (Moos and Zoback 1992). The significance of these findings is that reactivation of the SRS predominantly northeast striking faults would not induce potentially damaging earthquakes (Moos and Zoback 1992, WSRC 2000a).

3.2.6.2.2 Projected Recurrence of Earthquakes

Talwani and Schaeffer (2001) presented analyses of 15 years of paleoliquefaction investigations in the South Carolina Coastal Plain to arrive at an estimated recurrence interval for large earthquakes in the region. Their investigation concluded that the Charleston area had a recurrence interval for magnitude 7+ earthquakes (similar to the Charleston 1886 event) between 500 and 600 years. The recurrence of earthquakes associated with other known seismic zones in the region is not expected to be of greater intensity nor cause greater disturbance at the SRS.

3.2.7 Meteorology and Climatology

3.2.7.1 General Savannah River Site Climate

The SRS region has a humid subtropical climate characterized by relatively short, mild winters and long, warm, and humid summers. Summer-like conditions typically last from May through September, when the area is frequently under the influence of a western extension in the semi-permanent Atlantic subtropical anticyclone (i.e., the 'Bermuda' high). Winds in summer are light and cold fronts generally remain well north of the area. Daily high temperatures during the summer months exceed 90°F on more than half of all days on average. Scattered afternoon and evening thunderstorms are common (WSRC 2007a).

The influence of the Bermuda high begins to diminish during the fall as continental air masses become more prevalent, resulting in lower humidity and more moderate temperatures. Average rainfall during the fall is usually the least of the four seasons (WSRC 2007a).

In the winter months, mid-latitude low pressure systems and associated fronts often migrate through the region. As a result, conditions frequently alternate between warm, moist, subtropical air from the Gulf of Mexico region and cool, dry polar air. The Appalachian Mountains to the north and northwest of the SRS help to moderate the extremely cold temperatures that are associated with occasional outbreaks of Arctic air. Consequently, less than one-third of winter days have minimum temperatures below freezing on average, and days with temperatures below 20°F are infrequent. Measurable snowfall occurs an average of once every two years (WSRC 2007a).

Although spring weather is somewhat windy, temperatures are usually mild and humidity is relatively low (Kabela 2008, WSRC 2007a). Although rare in South Carolina, most tornadoes that do occur happen during the period of March, April, May and November (Ruffner 1985, WSRC 2007a). For the 49-year period of 1950-1998, an average of 11 tornadoes per year occurred in South Carolina (Storm Prediction Center, 2009). Between 1880 and 1995, a total of 17 significant tornadoes were reported in Aiken and Barnwell Counties, South Carolina, and Burke County, Georgia. Nine tornadoes have caused damage on the SRS, one with estimated wind speeds as high as 150 mph. None have caused damage to buildings on the SRS (DCS 2002).

Tropical storms or hurricanes affect South Carolina about once every 2 years. Most do little damage and affect only the coastal areas, decreasing in intensity as they move inland. Those that do move far inland can cause considerable flooding (Ruffner 1985). Thirty-six hurricanes caused damage in South Carolina between 1700 and 1992, and the interval between them has ranged from 2 months to 27 years. About 80% have occurred in August and September. The only hurricane-force winds measured at the SRS were associated with Hurricane Gracie on September 29, 1959, when wind speeds of 75 mph were measured at F-Area (DCS 2002, WSRC 2007a).

3.2.7.2 Local Meteorology

3.2.7.2.1 Data Sources

Meteorological data are critical input to atmospheric transport and dose models used to estimate the effects of releases from SRS facilities. A number of sources of data are used to describe the local climatology at the SRS. These include the eight meteorological towers adjacent to each of eight areas (A-, C-, D-, F-, H-, K-, L-, and P-Areas), the Central Climatology Site (CCS) located near N-Area, a meteorological instrument shelter in A-Area, and the NWS office at Bush Field in Augusta, GA (WSRC 2007a). The Bush Field station is located about 12 mi west-northwest of the SRS. In addition, the Tall Tower facility near Beech Island, South Carolina, provides a set of high-quality meteorological measurements that is unique to the southeast U.S. (Kabela 2008).

Locations of onsite towers are shown in Figure 3-16. The eight onsite towers near the major operations areas are equipped to measure wind direction, wind speed, temperature, and dew point at a height several hundred feet above ground. Temperature and dew point are also measured at 2 m. The ninth tower near N-Area (the CCS) is instrumented with wind, temperature, and dew point sensors at four levels: 6.6 feet (13.2 feet for wind), 59.4 feet, 118.8 feet, and 201.3 feet. The CCS is also equipped with an automated tipping bucket rain gauge, a barometric pressure sensor, and a solar radiometer near the tower at ground level. Data collected at the A-Area instrument shelter consist of temperature, daily precipitation, and relative humidity.

Data acquisition units at each of the nine onsite tower locations record a measurement from each instrument at 1-second intervals. Every 15 minutes, 900 data points are processed to generate statistical summaries for each variable, including averages and instantaneous maxima, and the results are uploaded to a relational database for permanent archival retention (Kabela 2008).

The Tall Tower near Beech Island utilizes fast-response sonic anemometers, water vapor sensors, barometric pressure sensors, slow-response temperature sensors and relative humidity sensors. Data are collected at 100 feet, 200 feet, and 1,000 feet above ground level. Spread-spectrum modems at each measurement level transmit raw data to a redundant set of personal computers at the SRNL. Data processing software on the personal computers determine mean values and other statistical quantities every 15 minutes and uploads the results to the relational database (WSRC 2008b). Data collected from the Bush Field station have been summarized by the National Climatic Data Center (NOAA 2004, NOAA 2007).

Precipitation measurements are collected from a network of 13 rain gauges across SRS (Figure 3-16). Twelve of these gauges are read manually by security or operations personnel once per day, usually around 6 am. The daily data are reported to the SRNL Atmospheric Technologies Center, where it is technically reviewed and manually entered into a permanent electronic database. The other is an automated rain gauge at the CCS previously addressed.

3.2.7.2.2 Summarized Data

The description of local climatology provided below is based on the summary of the Bush Field data (NOAA 2004, NOAA 2007) in addition to data provided by the Savannah River National Laboratory Atmospheric Technologies Group and summarized in recent reports (Blake et al. 2005b, DCS 2002, WSRC 2002a, Hunter and Tatum 1996, Mamatey 2006, Ruffner 1985, and Weber 1998). Precipitation, air temperature, and atmospheric dispersion data descriptive of the SRS are summarized below.

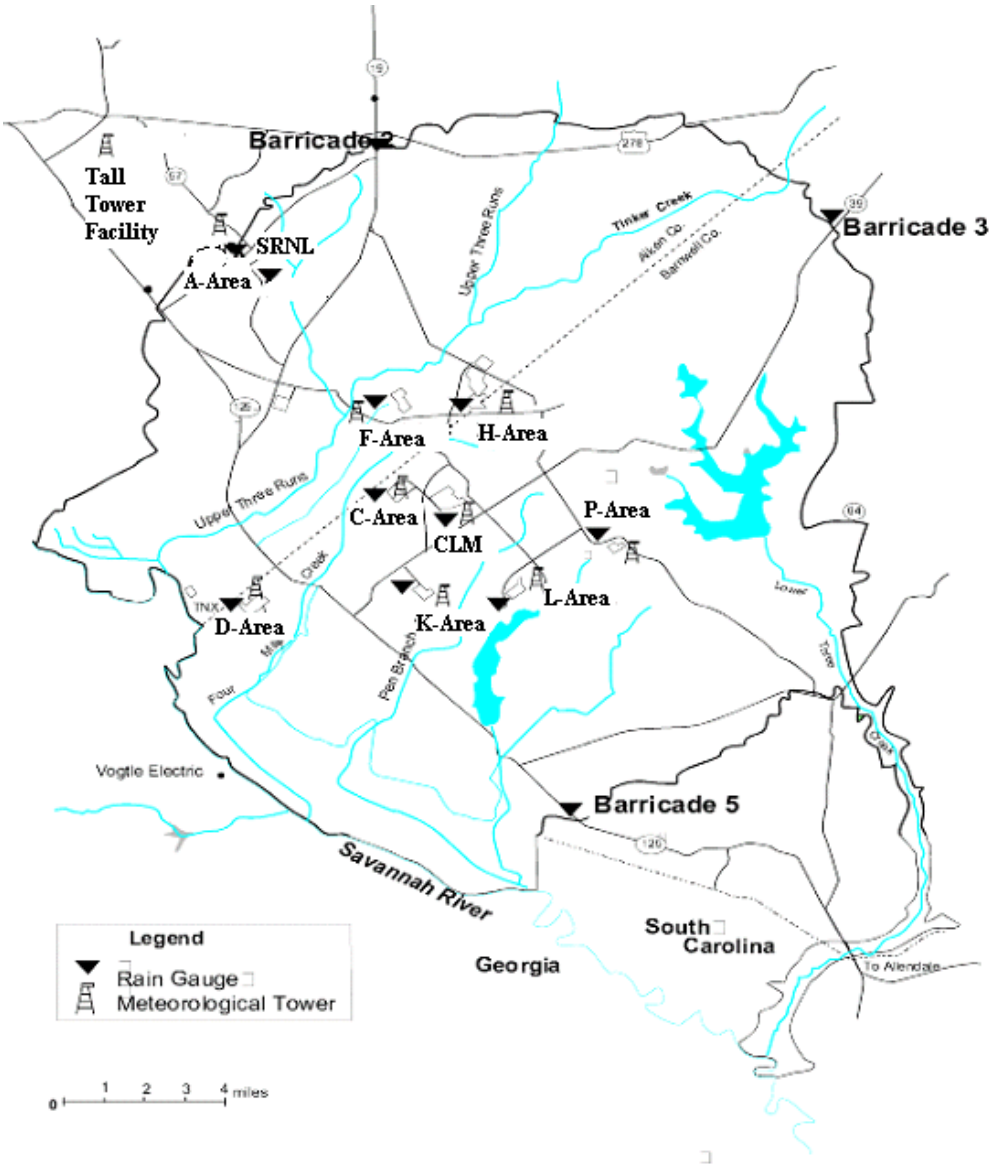


Figure 3-16. SRS Meteorological Monitoring Network (WSRC 2008b)

3.2.7.2.2.1 Precipitation

Average annual rainfall at the SRS is approximately 122 cm and at Bush Field average annual rainfall is approximately 113 cm (Table 3-6). Average monthly precipitation at the SRS ranges from 6.6 cm in November to 13.1 cm in July. April, May, October and November are typically the driest months at the SRS (Blake et al. 2005b). Rainfall events that are greater than 2 cm are common with an average occurrence of about 20 times a year. A rainfall event greater than 5 cm can be expected at least once a year and rainfall events greater than 10 cm in a 24-hour period can be expected every 5 to 10 years (Blake et al. 2005b).

Table 3-6. Summary of Local Climatology Data

Climate Data	Cited in Blake et al. 2005b[1]	Used in Dose Calculations in Mamatey, 2006[2]	NOAA Data for Augusta, GA[3]
Average Rainfall	122.5 cm/yr	122.4 cm/yr	113.3 cm/yr
Average Annual Air Temperature	18 °C	17.8 °C	17.3 °C
Range and Average Percent Relative Humidity[4]	36-97% 70%	Not reported	51-87% 69%
Average Wind Speed	not reported	3.83 m/s	2.9 m/s

[1]based on SRS meteorological data from A-Area station; data from 1952-2001 for rainfall; data from 1964-2001 for temperature and % relative humidity
[2]based on SRS meteorological data from H-Area station, 1997-2001; data on CD accompanying the Savannah River Site Environmental Report for 2005
[3]based on NOAA data for Augusta, GA; data from Bush Field Airport, 1971-2000 for rainfall and temperature (www.ncdc.noaa.gov/oa/climate/normals/usnormals.html); for average wind speed and % relative humidity, data from Augusta station(s) that are active or sites comparable in exposure (http://www1.ncdc.noaa.gov/pub/data/ccd-data/CCD-2006_fixed.pdf); data covers 1964-2006 for % relative humidity; data covers 1951-2006 for average wind speed
[4]based on monthly and annual means, minimums and maximums

Although annual average rainfall is 122 cm at the SRS, 1964 and 1972 were abnormally wet years with 186.6 cm falling in 1964 and 162.5 cm falling in 1972. In contrast, SRS received relatively little rainfall in 1954 (73.2 cm) and 2001-2002 (91.5 cm) (Blake et al. 2005b).

In general, the SRS receives little measurable snowfall. Annual snowfall averages approximately 3.6 cm at Bush Field in Augusta (NOAA 2004). At the SRS, the greatest monthly snowfall on record occurred in February 1973, with 35.6 cm. Freezing rain can also be expected to occur one to three times per winter (Blake et al. 2005b, Ruffner 1985).

3.2.7.2.2.2 Air Temperature and Relative Humidity

The average annual temperature at the SRS is about 18°C (Table 3-6). January is the coldest month, with an average monthly temperature ranging from 1.7 to 12.8°C, and July, the warmest, averaging 26.7 to 29.4°C (Blake et al. 2005b).

Below freezing temperatures can be expected from late October through early April, however extreme low temperatures are more typically in December and January (Blake et al. 2005b). Data from 1964 through 2006 for Augusta, GA, show that on average December has 14 days with minimum temperature of 0°C or less. January averages 15 days and February 12 days. May through September typically have no freeze days (where temperatures are below 0°C) (NOAA 2007).

Average annual relative humidity at Bush Field ranges from 87% in the early morning to 51 % in the afternoon (Table 3-6). In July and August, the early morning relative humidity averages 89-90%, with afternoons averaging 55%. At the SRS, comparable values of 97% and 50% are recorded for August (WSRC 2007a). Heavy fog with visibility less than 0.40 km occurs on an average of about 30 days per year. Heavy fog occurs throughout the year but is most likely in the fall and winter (Blake et al. 2005b).

3.2.7.2.2.3 Atmospheric Dispersion

Parameters pertinent to atmospheric dispersion include wind speed and direction, temperature, dew point, and horizontal and vertical turbulence intensities. Since the mid-1970s, a five-year database of meteorological conditions at the SRS is updated in order to support dose calculations. This allows data to be retrieved specific to each of the major operations areas at the site.

Average wind speed at the SRS is reported to be 3.83 m/s, as shown in Table 3-6) Wind-rose plots for each of the eight SRS towers near the major operations areas are shown in Figure 3-17 and Figure 3-18 for the time period from 1992 to 1996 (WSRC 2007a). As indicated in these plots, there is no strongly prevailing wind direction at the SRS.

Updated wind plots are available for each tower from the SRS Atmospheric Technologies Center.

Pasquill stability classes are commonly used to characterize atmospheric dispersion. The percent occurrence of Pasquill stability class for each of the eight operations area towers is summarized in Table 3-7. Stable conditions were observed between 20 and 30 percent of the time during 1992 to 1996. Wind rose plots by stability class for each tower can be found in the *Documented Safety Analysis Support Document* (WSRC 2007a).

Estimates of morning and afternoon mixing heights are shown in Table 3-8. These values were interpolated from radiosonde observations during the 5-year period, 1960-1964, as described in the *Documented Safety Analysis Support Document* (WSRC 2007a).

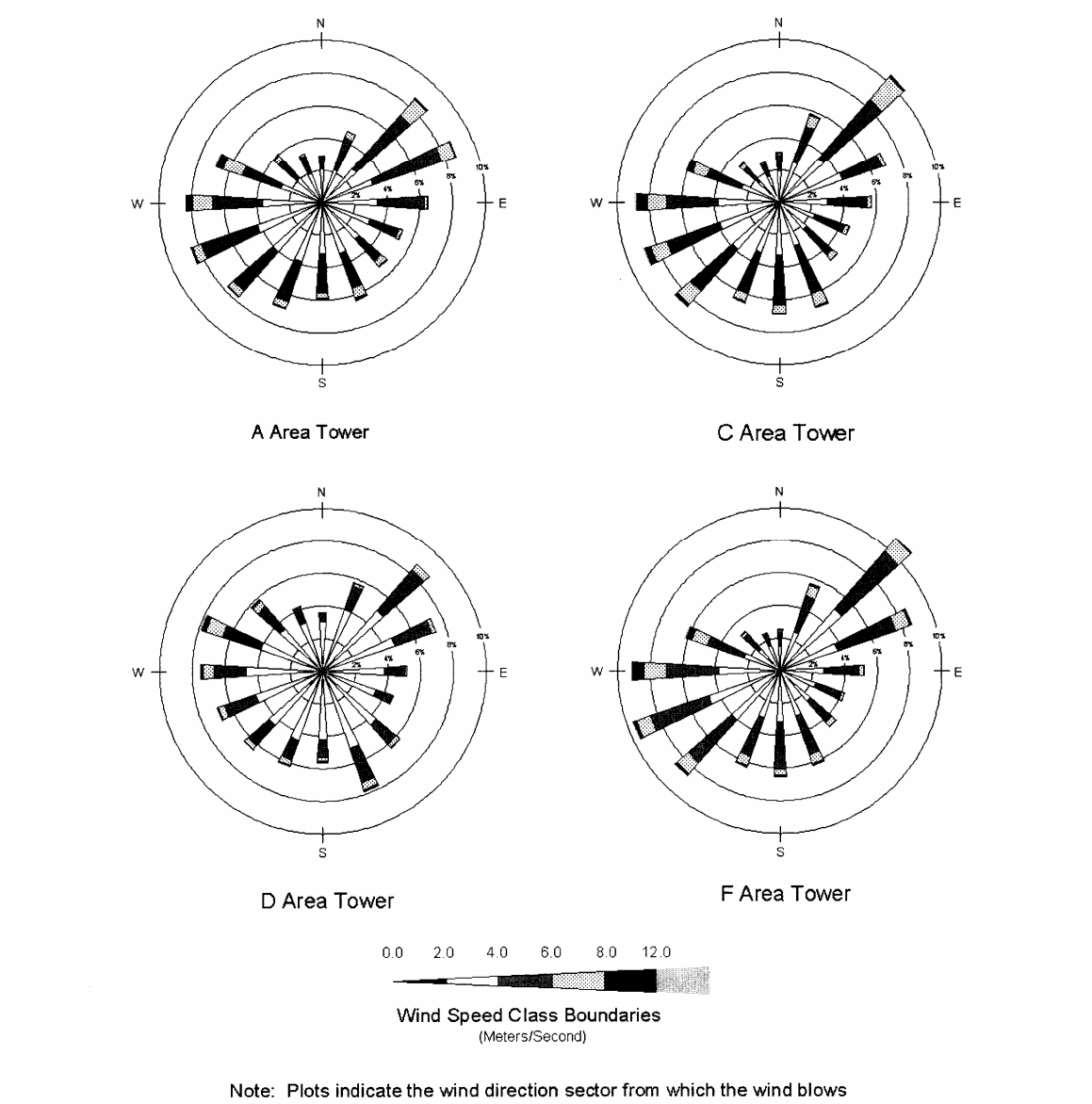
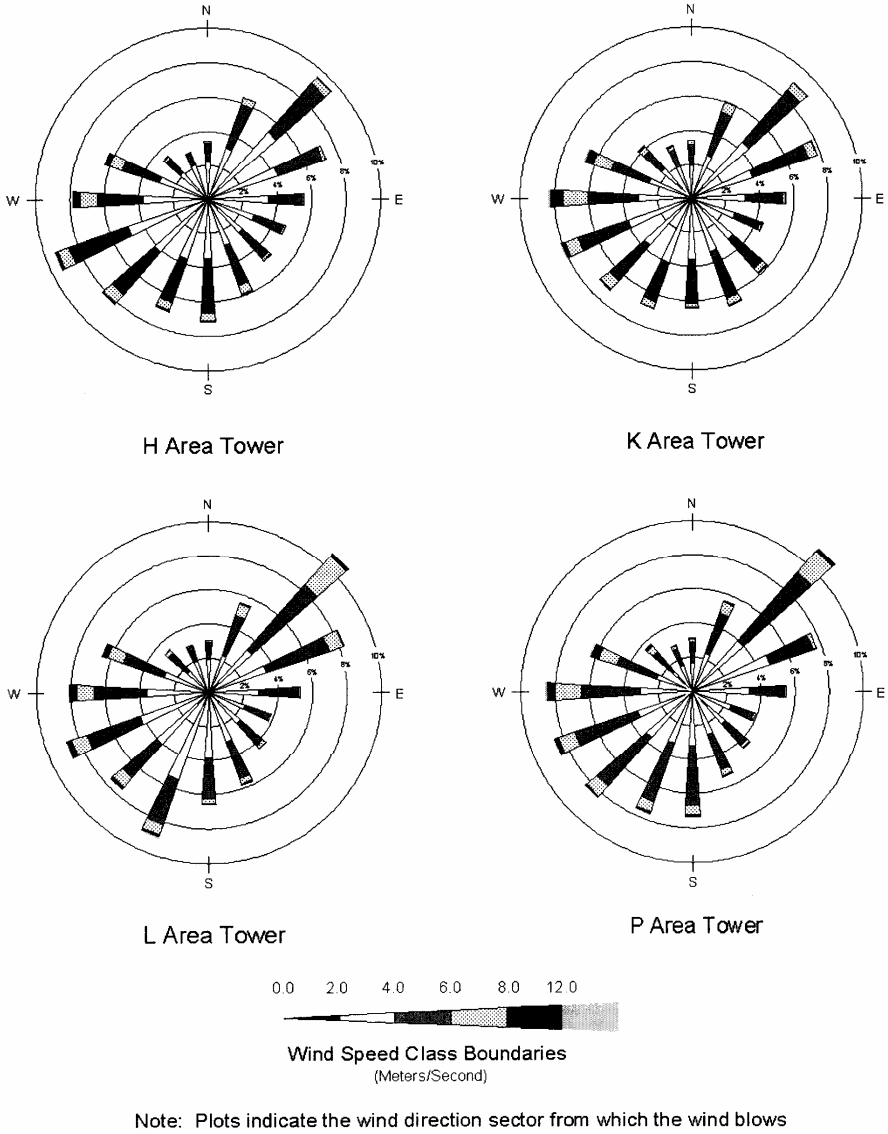


Figure 3-17. Wind-Rose Plots for A, C, D, and F Areas, 1992-1996 (from WSRC 2007a)



Note: Plots indicate the wind direction sector from which the wind blows

Figure 3-18. Wind-Rose Plots for H, K, L, and P Areas, 1992-1996 (from WSRC 2007a)

Table 3-7. Percent Occurrence of Atmospheric Stability Class for SRS Meteorological Towers (1992-1996)

Stability Class	Percent Occurrence Per Year							
	A-Area	C-Area	D-Area	F-Area	H-Area	K-Area	L-Area	P-Area
A	17.5	15.6	20.5	13.3	25.9	15.4	16.8	14.9
B	10.6	8.8	11.9	8.3	13.2	9.8	10.2	9.4
C	17.6	15.7	19.4	15.2	20.1	17.0	18.0	16.4
D	26.6	27.1	24.9	28.6	22.1	25.4	25.1	26.5
E	19.6	20.6	17.4	24.9	15.5	21.2	18.7	21.1
F/G	8.0	12.1	6.0	10.6	3.2	11.1	11.1	11.8
From Table 1.4-13 in WSRC (2007a)								

Table 3-8. Estimated Mixing Height for the SRS (meters)

Season	Mixing Height (m)	
	Morning	Afternoon
Winter	1148	3362
Spring	1230	5576
Summer	1312	5904
Fall	984	4592
Annual	1230	4756
From WSRC (2007a) – based on observations from 1960-1964.		

3.2.8 Background Infiltration and Water Balance

Numerous water balance and infiltration studies have been conducted in and around the SRS by various organizations including the SRL, the United States Geological Survey (USGS), the State University of New York at Brockport, the Pennsylvania State University, the University of Arizona, and the Desert Research Institute (DRI) (Phifer et al. 2007). Findings from eight such studies are reported in Section 3.2.8.1 and summarized in Section 3.2.8.2.

3.2.8.1 Background Water Balance and Infiltration Studies Findings

3.2.8.1.1 <u>Cahill 1982</u>

The USGS (Cahill 1982) conducted a study at the Barnwell Low-Level Radioactive Waste Disposal Facility (LLRWDF) (now called CNSI's LLRWDF) "to determine the geologic and hydrologic conditions near the burial site and to measure migration of leachates from buried waste into the surrounding unconsolidated sediments." The Barnwell LLRWDF is located immediately to the east of the SRS. As part of this study hydrologic budget estimates were made.

Precipitation estimates for the Barnwell facility were based upon the National Weather Service (NWS) station near Blackville, SC. The mean annual precipitation reported from this station for the years 1951 to 1980 was 46.62 in. Aquifer recharge (i.e., infiltration) estimates were made based upon measurements of discharge to local streams (i.e., stream flow measurements over defined steam reaches). It was assumed that on average aquifer recharge (i.e., infiltration) was equal to discharge from the aquifers to local streams.

Cahill (1982) made three infiltration estimates based upon stream flow measurements over defined steam reaches of Mary's Branch Creek, LTR, and the Duncannon Creek. Table 3-9 provides the results of the infiltration estimates made by Cahill (1982). Cahill (1982) concluded that infiltration "appears to range from about 14 to 17 inches/year." Based upon these infiltration estimates Cahill (1982) selected an infiltration rate of 15 in/yr to utilize within a 3-dimensional finite-difference groundwater flow model of the Barnwell facility. For the Barnwell facility Cahill (1982) stated, "Overland flow" (i.e., runoff) "is rare, occurring only during intense rainfall where forest litter has been removed, as in cultivated fields and along roadways." Therefore, Cahill (1982) assumed essentially no runoff from the Barnwell facility.

Although evaporation measurements using 4-ft diameter pans were made at the NWS station and at the Barnwell facility, this data was not utilized by Cahill (1982) to make an estimate of evapotranspiration. The evapotranspiration estimate was made simply by subtracting the infiltration and runoff estimates from the average annual precipitation. In summary Cahill (1982) reported that of the mean annual precipitation of 46.62 in (1951 to 1980) at the Barnwell facility about 60 to 70 percent of it returned to the atmosphere by evapotranspiration (i.e., 28 to 32.6 in), 30 to 40 percent became infiltration (i.e., 14 to 19 in), and essentially zero percent became runoff.

Table 3-9. Barnwell LLW Disposal Facility Infiltration Estimates by Cahill (1982)

Stream	Infiltration Estimate (in/yr)	Cahill (1982) Comment
Mary's Branch Creek	14.5	This estimate may be low since some underflow may not be accounted for
Lower Three Runs	17.6	This estimate may be high due to the effect of leakage from PAR Pond
Duncannon Creek	14	This estimate may be low since some underflow may not be accounted for

3.2.8.1.2 Hubbard and Emslie 1984

The State University of New York at Brockport in conjunction with the SRL (Hubbard and Emslie 1984) conducted a water budget evaluation for the SRP ORWBG. Precipitation was measured at F-Area from 1963 to 1973 and at the ORWBG from 1974 to 1982. An average annual precipitation of 47.32 in/yr was measured during this time period (1963 to 1982). Evapotranspiration was evaluated by four different methods as outlined in Table 3-10.

Although the average evapotranspiration for the four methods of estimation was 32.57 in/yr, Hubbard and Emslie (1984) took an evapotranspiration of 30 in/yr for determination of an estimated infiltration. Runoff was estimated at 2 in/yr based upon stream flow records, a soil survey inspection, and visual observations of runoff at the ORWBG. Infiltration was estimated by subtracting evapotranspiration and runoff from the average annual precipitation, resulting in an estimated value of 15 in/yr. Based upon these estimates Hubbard and Emslie (1984) produced the following annual water balance for the ORWBG:

- Precipitation = 47 in/yr
- Evapotranspiration = 30 in/yr
- Runoff = 2 in/yr
- Infiltration = 15 in/yr

Table 3-10. Evapotranspiration Estimates by Hubbard and Emslie (1984)

Method of Estimation	Evapotranspiration (in/yr)
Stream basin water balance method: The stream flow of three nearby streams (South Fork of the Edisto River, Upper Three Runs Creek, and Mary's Branch) expressed as water yield in inches was subtracted from the average annual precipitation	31.17
Thornthwaite Method based upon the average monthly temperature and precipitation	36.13
1974 through 1978 monthly evaporation pan data from Blackville, South Carolina multiplied by monthly pan coefficients developed at the USDA research watersheds at Tifton, Georgia	33.27
SRP ORWBG lysimeter water balance data	29.7
Average	**32.57**

3.2.8.1.3 Hubbard 1986

The State University of New York at Brockport in conjunction with the SRL (Hubbard 1986) provided an updated ORWBG water balance based upon an evaluation of five years worth of data from the Defense Waste Lysimeter study modified with data from evaporation pan climatic data and other watershed, lysimeter, and runoff studies. The Defense Waste Lysimeters were six or ten ft in diameter, were filled with soil from the ORWBG, and did not allow runoff because the lysimeters extended above the ground surface. The vegetative cover of most lysimeters consisted of a sparse coverage of Bahia grass and herbaceous plants; however, some of the ten-ft diameter lysimeters contained 10 to 14 ft high pine trees. Analysis of five and a half years worth of data resulted in the following average annual water balance for the Defense Waste Lysimeters from 1980 through 1985:

- Rainfall = 47.0 in/yr
- Infiltration and runoff = 19.6 in/yr
- Evapotranspiration =27.4 in/yr

Hubbard (1986) considered the following data in arriving at an estimated ORWBG water balance:

- Runoff estimates from other sources ranging from 1 to 3 in/yr
- Other evapotranspiration estimates ranging from 30 to 34 in/yr

Based upon this data Hubbard (1986) produced the following estimated annual water balance for the ORWBG:

- Precipitation = 48 in/yr
- Evapotranspiration = 30 in/yr
- Runoff = 2 in/yr
- Infiltration = 16 in/yr

Hubbard (1986) also made the following observations regarding the water balance data produced from the Defense Waste Lysimeters:

> "Lysimeter studies indicate that about 12 inches more water is lost annually to the atmosphere by evapotranspiration with deep-rooted pine trees present than in areas where bare soil or shallow-rooted grass cover occur …. In forested areas near the burial ground, evapotranspiration is estimated to be about 40 inches annually, and therefore recharge to the water table is about 6 inches."

3.2.8.1.4 Parizek and Root 1986

The Pennsylvania State University (Parizek and Root 1986) conducted a hydrologic water budget study of the McQueen Branch watershed, located in the central portion of the SRS as part of the development of a groundwater model. The McQueen Branch watershed consisted of approximately 67% pine trees, 18% hardwood, 15% grass and unvegetated of which approximately 2% are impervious surfaces such as pavement. The field study was conducted from November 1, 1982 to May 18, 1984. In order to arrive at the hydrologic water budget Parizek and Root (1986) measured precipitation within the watershed; took stream flow measurements in order to determine the stream baseflow and runoff; calculated runoff from impervious areas; took water level measurements within water table wells in order to determine changes in groundwater storage; took soil-moisture measurements in order to determine changes in vadose zone water storage; estimated evapotranspiration using the Penman-Monteith evaporation method based upon watershed weather and vegetation data; estimated underflow through the McBean beneath the stream gauge; and estimated leakage through the Green Clay aquitard. Table 3-11 provides the three hydrologic water budgets that Parizek and Root (1986) produced from their study: annual water budget based upon >18 months of data; annual water budget based upon 13 months of data; and assumed average annual water budget.

Table 3-11. Parizek and Root (1986) McQueen Branch Watershed Water Budget Summary

Water Budget Parameter	>18 Month Annual Water Budget (in/yr)	13 Month Annual Water Budget (in/yr)	Assumed Average Annual Water Budget (in/yr)
Precipitation	52.40	48.44	47.78
Runoff	3.41	3.32	2
Evapotranspiration	34.07	30.95	30.78
Infiltration	15.26	14.17	15

3.2.8.1.5 Hubbard and Englehardt 1987

The State University of New York at Brockport in conjunction with the University of Arizona (Hubbard and Englehardt 1987) utilized the Chemicals, Runoff, and Erosion from Agricultural Management Systems (CREAMS) model to produce estimated annual water balances for the SRP Burial Ground utilizing site specific weather data from 1961 to 1986. "The CREAMS model was developed by the US Department of Agriculture Research Service for the Environmental Protection Agency under the Clean Water Act, to give 'reasonable estimates' of the water balance used to calculate diffuse pollution transported with surface runoff in small, cropped agricultural areas." Additionally the infiltration, percolation, and evapotranspiration routines of the HELP model (Schroeder et al. 1994) are almost identical to those used in the CREAMS model. Table 3-12 provides the CREAMS water balance produced for the minimum, average, and maximum site-specific precipitation recorded from 1961 to 1986.

Table 3-12. SRS Burial Ground Estimated Water Balance from 1961 to 1986

Parameter	Precipitation (in/yr)	Runoff (in/yr)	Evapotranspiration (in/yr)	Infiltration (in/yr)
Minimum	34.67	0.13	29.06	4.97
Average	48.51	1.21	32.60	14.70
Maximum	71.88	4.12	35.92	32.14
Standard Deviation	8.73	0.90	2.64	6.04

3.2.8.1.6 Dennehy and McMahon 1989

The United States Geological Survey (USGS) (Dennehy and McMahon 1989) conducted a study at the Barnwell LLRWDF "of water movement in and adjacent to trenches excavated in the unsaturated zone and assesses the principal factors affecting this movement." As part of this study hydrologic budget estimates were made for an undisturbed portion of the site. The mean annual precipitation reported from the NWS station near Blackville, South Carolina station for the years 1883 to 1983 was 47.8 in (121.5 cm). However, precipitation at the Barnwell facility was also measured with an onsite meteorologic station. Precipitation for the study period from July 1983 through June 1984 was measured at 56.7 in (144 cm) with the onsite station. The actual evapotranspiration was estimated using the Bowen ratio/energy budget method with necessary input variables measured with the onsite meteorologic station. Evapotranspiration for the study period from July 1983 through June 1984 was estimated at 39.8 in (101 cm) using this methodology.

"Zero runoff was assumed to have occurred during this period." "The net change in the unsaturated zone was negligible", based upon tensiometer measurements. "The hydrologic budget is given by the equation $R = P - ET - RO \pm dS / dt$ where R = recharge to the saturated zone; P = precipitation ET = evapotranspiration; RO = runoff; and dS/dt = storage changes in the unsaturated zone. Based upon this equation, the recharge (i.e., infiltration) for the study period from July 1983 through June 1984 was estimated at 16.9 in (43 cm). Based upon a mean annual precipitation of 47.8 in, the results of this study, and an assumption of a linear relationship between precipitation, and evapotranspiration and infiltration, a mean annual evapotranspiration and infiltration of 33.5 in and 14.3 in, respectively, result. Where land surface conditions are different such that measurable runoff is possible, Dennehy and McMahon (1989) assume that infiltration would probably be less.

3.2.8.1.7 Young and Pohlmann 2001

The DRI (Young and Pohlmann 2001) conducted both deterministic and probabilistic (100 Monte Carlo runs) modeling utilizing the computer code HYDRUS 2-D (finite difference model solving Richard's equation) to estimate infiltration at the SRS E-Area. Estimates of evapotranspiration were made using the model of Feddes et al. (1978). The following types of data were utilized as input to the modeling effort:

- E-Area stratigraphic layering based upon cone penetrometer (CPT) data
- E-Area soil texture, saturated hydraulic conductivity, and water retention properties determined from the laboratory testing of undisturbed soil samples taken with depth
- E-Area soil water content with depth from water content reflectometer and neutron probe measurements
- E-Area soil water potential (matrix potential) with depth from tensiometers
- Meteorological data from weather stations at SRS and Augusta, Georgia

Based upon this methodology Young and Pohlmann 2001 produced the following estimated E-Area infiltration:

- The deterministic modeling resulted in an infiltration estimate of 9.5 in/yr
- The probabilistic (100 Monte Carlo runs) modeling resulted in a median infiltration estimate of 9.1 in/yr with a 90% confidence interval of 8.1 to 9.8 in/yr

While the modeling effort by Young and Pohlmann (2001) produced evapotranspiration estimates in the process of estimating infiltration, they did not report their evapotranspiration estimates within the document. However based upon their infiltration estimates, the associated evapotranspiration estimates would have had to be relatively high (at least in the 30s of in/yr range).

3.2.8.1.8 Young and Pohlmann 2003

The DRI (Young and Pohlmann 2003) refined the modeling that they performed in 2001 (Young and Pohlmann 2001) by the following:

- Incorporating additional E-Area soil texture, saturated hydraulic conductivity, and water retention data determined from the laboratory testing of undisturbed soil samples taken with depth
- By apportioning potential evaporation between soil evaporation and plant transpiration
- Considering plant cover at 0, 50, and 100 percent

They conducted probabilistic (100 Monte Carlo runs) modeling utilizing the computer code HYDRUS 2-D (finite difference model solving Richard's equation) to estimate infiltration at the SRS E-Area under conditions of 0, 50, and 100 percent plant cover. Estimates of evapotranspiration were made using the model of Feddes et al. (1978).

Based upon this methodology Young and Pohlmann (2003) produced E-Area infiltration estimates of 11.7, 7.1, and 5.0 in/yr for plant cover of 0, 50, and 100 percent, respectively.

While the modeling effort by Young and Pohlmann (2003) produced evapotranspiration estimates in the process of estimating infiltration, they did not report their evapotranspiration estimates within the document. However, based upon their infiltration estimates, the associated evapotranspiration estimates would have had to be relatively high (at least in the 30s of in/yr range).

3.2.8.2 Background Water Balance and Infiltration Studies Summary

Eight water balance and infiltration studies were evaluated (Phifer et al. 2007). They included both field and modeling studies and ranged in scale from 55-gallon drum lysimeters to entire watersheds. Table 3-13 provides the nominal water balance and infiltration estimate produced from each of the eight studies along with the median of the nominal water balance values of the eight studies. Table 3-14 provides the range of values reported by Hubbard and Englehardt (1987) for precipitation ranging from 34.7 to 71.9 in/yr.

As seen in Table 3-13 and Table 3-14, precipitation is distributed, in decreasing order, into evapotranspiration, infiltration, and runoff. Precipitation is seen to range from 35 to 72 in/yr with a median of the eight studies' nominal values of 47.8 in/yr. Evapotranspiration is seen to range from 29 to 36 in/yr with a median of the eight studies' nominal values of 31.2 in/yr. Infiltration is seen to range from 5 to 32 in/yr with a median of the eight studies' nominal values of 14.8 in/yr. Runoff constitutes very little of the water balance; it is seen to range from 0.1 to 4 in/yr with a median of the eight studies' nominal values of 1.6 in/yr. Clearly evapotranspiration dominates the water balance distribution of precipitation at the SRS.

Based upon the results of these background water balance and infiltration studies, the nominal annual average background infiltration at SRS has been historically taken as 15 inches/year for modeling purposes (Shine 2008). Additionally the 1,000-year average background infiltration rate standard deviation has been determined to be 0.17 inches/year, based upon an analysis of this historical data and a 100-year simulation history of background infiltration (Phifer et al. 2007; Shine 2008).

3.2.9 Surface Water Hydrology

The Aiken Plateau, on which much of the SRS is located, slopes to the southeast approximately 5 ft/mi. The plateau is dissected by streams that drain into the Savannah River. The Savannah River Basin (Figure 3-19) is one of the major river basins in the southeastern United States and is the principal surface-water system near the SRS. It has a drainage area of 10,577 mi^2, of which 8,160 mi^2 are upstream of SRS. The River Basin is located in three physiographic regions or provinces: the Mountain, the Piedmont, and the Coastal Plain (WSRC 2007a).

Table 3-13. Nominal Water Balance and Infiltration Estimate Produced from Each of Eight Studies[a]

Source	Nominal Annual Precipitation (in/yr)	Nominal Annual Runoff (in/yr)	Nominal Annual Evapotranspiration (in/yr)	Nominal Annual Infiltration (in/yr)
Cahill (1982)	46.62	0	31.62	15
Hubbard and Emslie (1984)	47	2	30	15
Hubbard (1986)	48	2	30	16
Parizek and Root (1986)	47.78	2	30.78	15
Hubbard and Englehardt (1987)	48.51	1.21	32.60	14.70
Dennehy and McMahon (1989)	47.8	0	33.5	14.3
Young and Pohlmann (2001)	10-yr Augusta, GA data from 1977 to 1987	Assumed to be 0	Determined but not reported within the document [b]	9.1
Young and Pohlmann (2003)	10-yr Augusta, GA data from 1977 to 1987	Assumed to be 0	Determined but not reported within the document [b]	11.7
Median of the eight Studies Nominal Values [c]	47.79	1.6	31.2	14.85

[a] In this table all of these studies assumed that the change in water storage was a minor water budget component

[b] Based upon the infiltration estimates, the associated evapotranspiration estimates would have had to be relatively high (at least in the 30s of in/yr range).

[c] The median of the eight studies' nominal values does not include precipitation, runoff, and evapotranspiration from Young and Pohlmann (2001 and 2003)

Table 3-14. Hubbard and Englehardt (1987) Water Balance Range

Parameter	Precipitation (in/yr)	Runoff (in/yr)	Evapotranspiration (in/yr)	Infiltration (in/yr)
Range	34.7 to 71.9	0.1 to 4.1	29.1 to 35.9	5.0 to 32.1

3.2.9.1 The Savannah River

3.2.9.1.1 Upstream and Downstream of the SRS

The headwaters of the Savannah River are in the Blue Ridge Mountains of North Carolina, South Carolina, and Georgia, at the junction of the Tugaloo and Seneca Rivers approximately 100 mi northwest of SRS (See Figure 3-19). The river empties into the Atlantic Ocean near Savannah, GA, approximately 95 mi (about 160 river mi) southeast of the SRS (DOE 2002, WSRC 2007a). From the Hartwell Reservoir Dam to the Savannah Harbor, the river runs a course of 289 river mi.

Three large reservoirs on the Savannah River upstream of the SRS (Strom Thurmond, Richard B. Russell, and Hartwell) provide hydroelectric power, flood control, and recreation. The closest of these to the SRS is Strom Thurmond Reservoir (formally Clarks Hill Lake), which was completed in 1952, is approximately 35 mi upstream of the site, and encompasses approximately 2.51 million acre-ft. These three reservoirs, along with the New Savannah River Bluff Lock and Dam (located near Augusta, GA), stabilize stream flow in the vicinity of the SRS. River flow averages about 10,000 cfs at the SRS (DOE 2002, CSRACT 2007, and WSRC 2007a).

Upstream of the SRS, the river supplies domestic and industrial water needs for Augusta, GA, and North Augusta, SC. The river receives treated wastewater from these municipalities and from Horse Creek Wastewater Treatment Facility (Aiken, SC). VEGP (a commercial power plant) withdraws an average of 92 cfs from the river for cooling and returns an average of 25 cfs. The Urquhart Steam Generating Station at Beech Island withdraws approximately 261 cfs of once-through cooling water. No uses of the Savannah River for irrigation have been identified in either South Carolina or Georgia (WSRC 2007a).

Downstream of the SRS, the Coastal Plain has a negligible gradient, with an elevation change of 200 ft over approximately 95 mi to the ocean (~ 2 ft/mi). The soils of this region are primarily stratified sand, silts, and clays. The Coastal Plain contains about 31% of the total Savannah River drainage area, and includes the city of Savannah, GA. In the Coastal Plain, the Savannah River is slow moving. Tidal effects may be observed up to 40 mi upriver, and a salt front extends upstream along the bottom of the riverbed for about 20 mi (WSRC 2007a).

The Savannah River downstream from Augusta, GA, is classified by the State of South Carolina as a Class B waterway, which is suitable for agricultural and industrial use, the propagation of fish, and after treatment, domestic use. Below the mouth of UTR, raw water is pumped from the Savannah River for drinking water supplies. The City of Savannah Industrial and Domestic Water Supply Plant (Savannah I&D), near Port Wentworth, GA (Figure 3-19) withdraws water to supply an estimated consumer population of about 26,300. The Beaufort-Jasper Water and Sewer Authority (BJWSA) has two treatment plants near Hardeeville, SC (Figure 3-19), which supply consumer populations of approximately 58,000 (Purrysburg facility) and 77,000 (Chelsea facility). Use of the Savannah River downstream of the SRS, including water-contact recreation, is less extensive than it is upstream of SRS (Mamatey 2008).

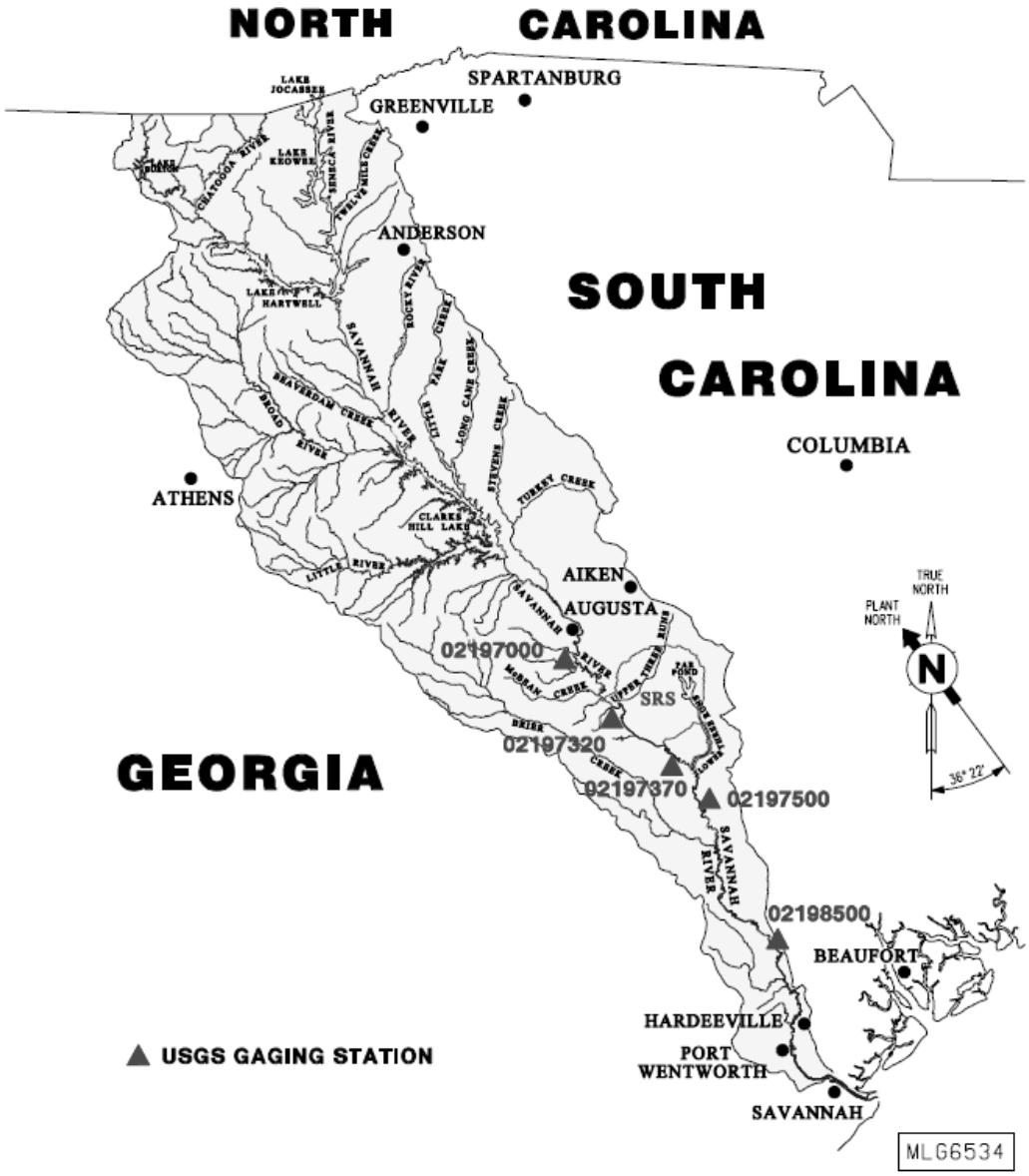


Figure 3-19. Savannah River Basin (Source: WSRC 2000a)

3.2.9.1.2 Savannah River Site Vicinity

The Savannah River is about 340 ft wide and from 9 to 16 ft deep in the vicinity of the SRS under normal conditions. The river is gauged above the SRS near Augusta, GA, at a few locations where the river borders the site, and below the SRS at U.S. Highway 301 Bridge (Station 02197500) and 3 mi north of Clyo, GA (station 02198500) (see Figure 3-19) (WSRC 2007a). From the SRS, river water usually reaches the coast in approximately 5 to 6 days, but can take as few as 3 days. The average flow at Augusta, GA, since the filling of Thurmond Lake (Clarks Hill) has been approximately 9,600 cfs. Flows increase below Augusta, GA, to about 12,000 cfs near Clyo, GA, about 100 mi downriver. The seven-day low flow, 10-yr recurrence (7Q10) value for the Savannah River at four measurement locations near the SRS is given in Table 3-15. The 7Q10 value is a measure of the dependability of flow. The 7Q10 flow is derived from the frequency curve of the yearly 7-day low flow statistics over the period of record at that stream or river location (WSRC 2007a).

Table 3-15. Low Flow Statistics for the Savannah River and SRS Streams (from WSRC 2007a)

Gauging location	7Q10 (cfs)[a, b]
Savannah River	
at Augusta, GA	4332
at SRS Boat Dock (See Figure 4-2)	4293
at U.S. Highway 301 Bridge[c] (See Figure 4-2)	4411
at Clyo, GA (where Highway 119 crosses the Savannah River)	5211
Upper Three Runs	
at Highway 278 (See Figure 3-1)	56
at SRS Road C (just downstream of F-Area)	100
at Highway 125	100
Beaver Dam Creek	
at D-Area	0.01
Fourmile Branch	
at SRS Site 7	0.58
Pen Branch	
at SRS Road B (near K-Area; see Figure 3-1)	0.27
at SRS Road A-13 (near the Savannah River Swamp; see Figure 3-11)	5.5
Steel Creek	
at Highway 125	12.9
Lower Three Runs	
below PAR Pond	1.2
near Snelling, SC	16

This table was calculated from U.S. Geological Survey (USGS) stream measurement data after construction of Thurmond Dam (formally Clarks Hill Dam).

[a]The data set from which statistics were derived consisted of daily average flows with varying periods of record (from 2 to 81 years) for SRS streams and the Savannah River.

[b] Values listed for 7Q10 flow (7-day low flow, ten yr recurrence) are based on adjusted "natural" flows, i.e., without the effects of cooling water discharges from SRS reactors.

[c]Eleven years are missing between 1971 and 1982.

The Savannah River overflows its channel and floods the swamps bordering the site when its elevation rises higher than 88.5 ft above msl (which corresponds to flows equal to or greater than 15,470 cfs). River elevation measurements made at the SRS Boat Dock indicate that the swamp was flooded approximately 20% of the time (74 d/yr on the average) during the period from 1958 through 1967. Since the construction of the upstream reservoirs, the maximum average monthly flow has been 43,867 cfs for the month of April. The minimum flow that is required for navigation downstream from Strom Thurmond Dam is 5,800 cfs (WSRC 2007a).

The major tributaries at the SRS that flow to the Savannah River include UTR, Fourmile Branch, Pen Branch, Steel Creek, and LTR (Figure 3-11). While Pen Branch reaches the Savannah River swamp, it does not discharge directly into the Savannah River; rather it joins Steel Creek in the Savannah River floodplain swamp (Wike et al. 2006). Beaver Dam Creek, the smallest of the six SRS tributaries of the Savannah River, is located north of Fourmile Branch, primarily in the floodplain of the Savannah River. Without discharges from SRS operations in D-Area, flow within Beaver Dam Creek is negligible (Wike et al. 2006). Tinker Creek and Tims Branch are tributaries of UTR; Castor Creek is a tributary of Fourmile Branch; and Indian Grave Branch is a tributary of Pen Branch. Each creek originates on the Aiken Plateau and descends 49 to 200 ft before discharging to the Savannah River (WSRC 2007a).

The Savannah River Swamp lies within the SRS floodplain for a distance of about 10 mi and averages about 1.5 mi wide. A small embankment or natural levee has built up along the north side of the river from sediments deposited during periods of flooding. The top of the natural levee is approximately 3 to 6 ft above the river during normal flow (river stage 85 ft) at the SRS boat dock. Three breaches in this levee (at the confluences with Beaver Dam Creek, Fourmile Branch, and Steel Creek) allow discharge of stream water to the river. During periods of high river level (above 88 ft), river water overflows the levee and stream mouths and floods the entire swamp area. The water from these streams mixes with river water and then flows through the swamp parallel to the river and combines with the Pen Branch flow. The flows of Steel Creek and Pen Branch converge 0.5 mi above the Steel Creek mouth. However, when the river level is high, the flows are diverted parallel to the river across the offsite Creek Plantation Swamp, which borders the SRS portion of the Savannah River Swamp to the southeast (Figure 3-20). Ultimately, the flows join the Savannah River flow near Little Hell Landing (Figure 3-20) (WSRC 2007a).

SRS was once a major user of water from the Savannah River but since all SRS reactors are now shut down, river water withdrawals are minimal. Past operations typically removed about 9% of the average annual Savannah River flow, but river water usage only averaged 0.133 cfs during the second quarter of 1995. Currently, one pump is operated to provide water from the Savannah River to the site (Pumphouse #3), which can supply up to 5,000 gallons per minute (gpm). This supply rate is typically more than is needed for system uses on the SRS. The water is used to maintain the L Area fire system and L Lake levels, K Area stream outfall and as standby for maintenance of PAR Pond level (e.g., drought conditions) if needed. Two additional 30,000 gpm pumps are still operable if needed (WSRC 2007a).

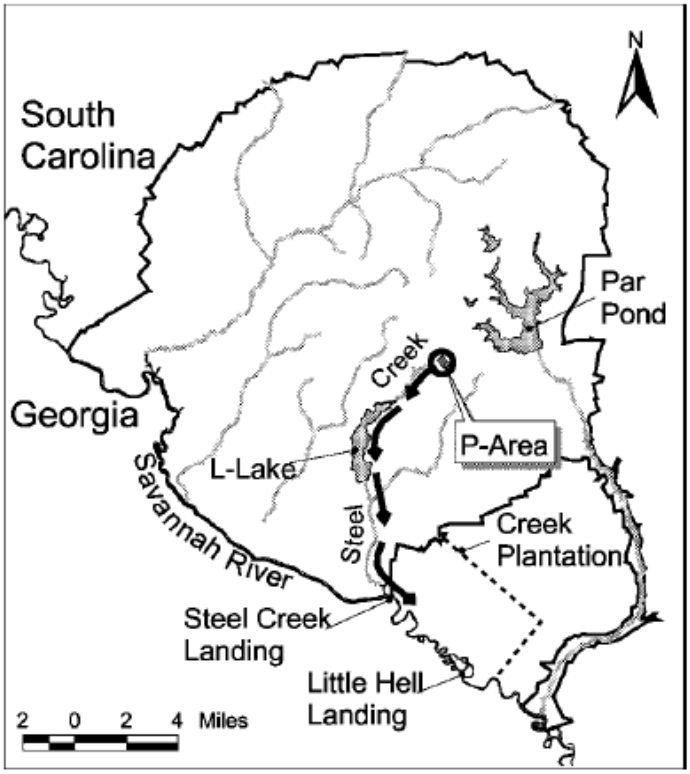


Figure 3-20. Location of the Creek Plantation Swamp and Little Hell Landing (Arnett and Mamatey 2002)

Three SRS studies have contributed to an understanding of the degree of mixing of effluents from site streams with the Savannah River water in the vicinity of the SRS (Heffner et al. 1998). Two of these studies involved thermal infrared aerial photographs showing parts of the Savannah River which receive discharges from site tributaries. The third study involved tritium sampling done in the Savannah River and the Savannah River Swamp. Sampling for tritium is routinely carried out because the largest contributions of tritium to the Savannah River in the SRS vicinity originate in tritiated groundwater that migrates into Savannah River tributaries located on the site (Heffner et al. 1998).

In March 1983, thermal infrared aerial photographs surveyed parts of the Savannah River where site tributaries discharged hot water. Conditions at the time of this survey included a high river stage and a flooded swamp. The photographs (reproduced in Figure 3-21) show clearly that in this flow regime, waters discharged from Fourmile Branch and Pen Branch emerge into the swamp and migrate close to the river bank for several mi, then merge and mix with the main river flow downstream of Steel Creek landing (Figure 3-20). The left side of Figure 3-21 shows the heated discharge from Fourmile Branch and from Pen Branch; the right side shows how the heated water from the tributaries maintains its identity close to the bank for about 5 mi and does not enter the river's main channel until some point downstream of the mouth of Steel Creek (which comes in where the swamp narrows) (Heffner et al. 1998).

In March and August of 1981, thermal infrared aerial photographs were taken in the same area. Conditions at the time of these surveys were low river flow, without swamp flooding. The corresponding photographs (Figure 3-22) show clearly that in these conditions, waters from Fourmile Branch and Pen Branch enter the Savannah River and mix thoroughly within 2000 ft of downstream flow. The left side of the figure shows the discharge from Fourmile Branch in March; the right side shows the discharge in the same area in August (Heffner et al. 1998).

In February 1990, under conditions of high river flow and flooded swamp, water samples were collected in a transect from the submerged normal riverbank location landward toward the floodstage riverbank. Analyses of these samples showed low tritium concentrations in the river channel and near the normal bank. Concentrations were low along most of the transect but were elevated for the few hundred feet of swamp nearest the temporary bank (Heffner et al. 1998).

These historical studies show two distinctly different patterns in the mixing of water from the Savannah River with water from SRS tributaries to the river. When the river stage is high, the swampy area along the SRS is flooded, with slow, shallow, low-turbulence flow. Under these conditions, Fourmile Branch and Pen Branch water enters the swamp and stays close to the bank, onsite, not mixing fully with the Savannah River until it moves downstream past the Steel Creek Landing. At low stages of the Savannah River, Fourmile Branch flows directly into the river channel, and mixes thoroughly with the river water within a few thousand ft. Regardless of river stage, the Savannah River at the U.S. Highway 301 Bridge (downstream of Steel Creek, at river mile 118.8; see Figure 4-2) is fully mixed with all tributaries that feed it from the SRS (Heffner et al. 1998).

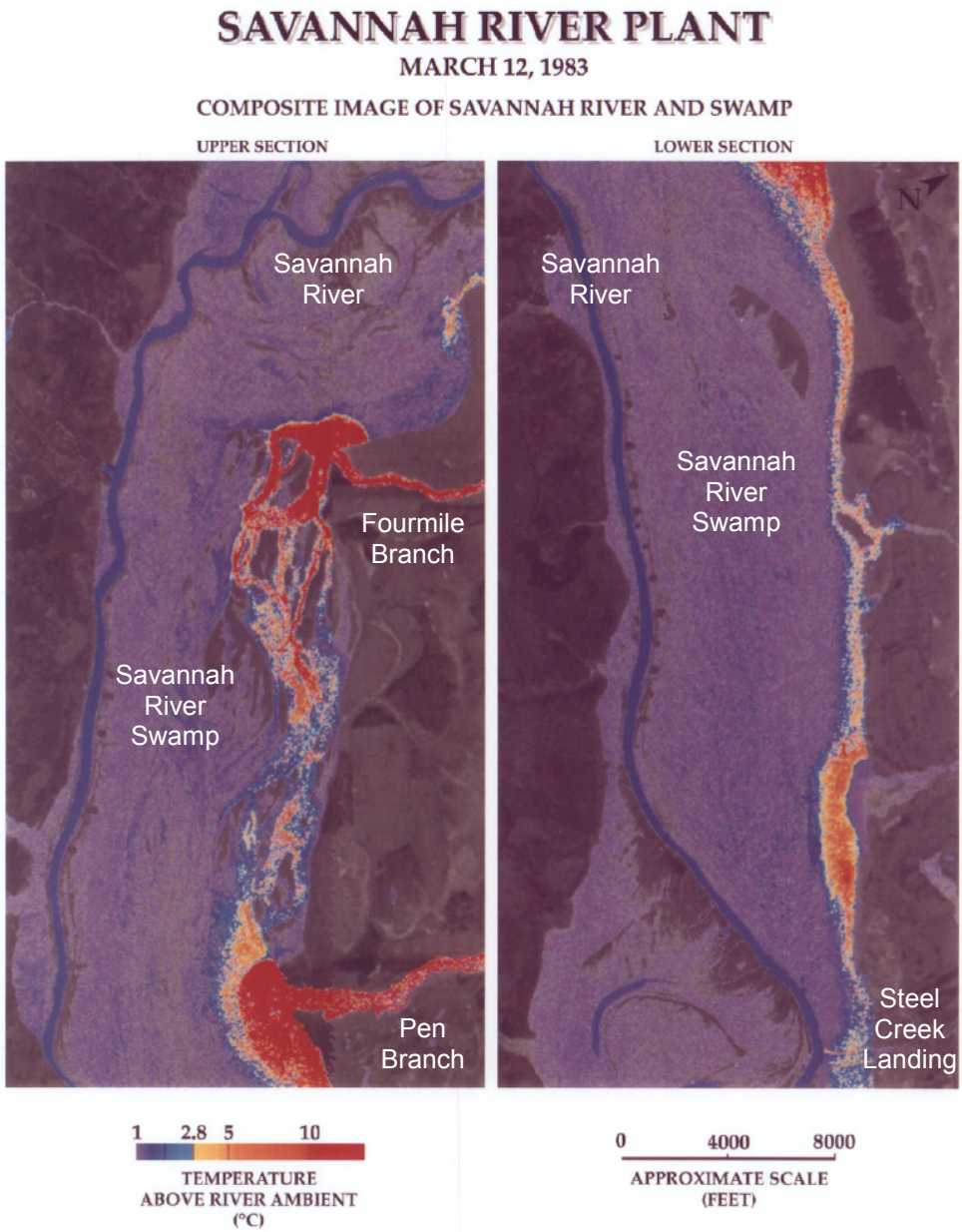


Figure 3-21. Aerial Photograph of the Savannah River, Thermal Infrared Image at High River Stage (from Heffner et al. 1998)

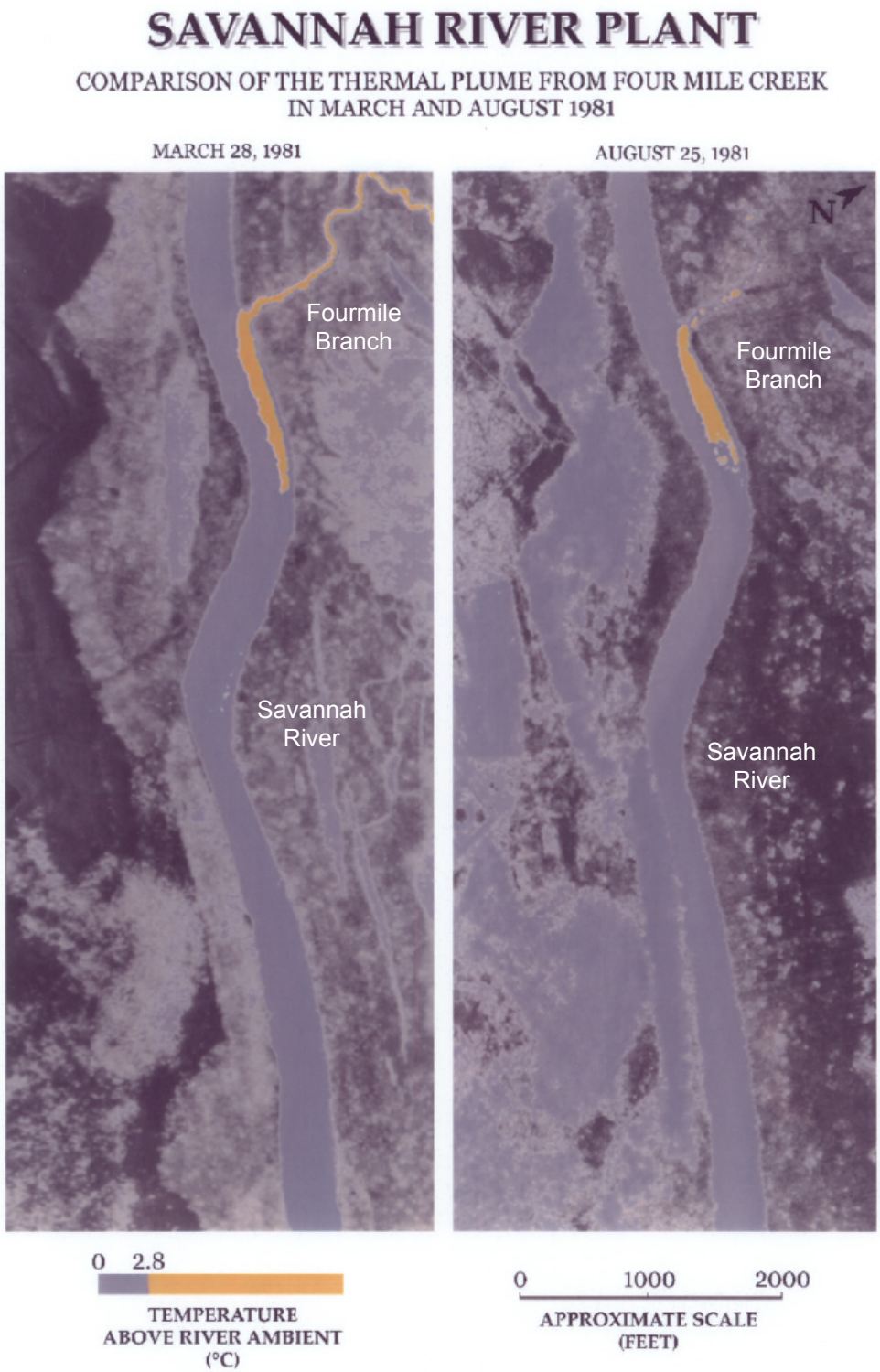


Figure 3-22. Aerial Photograph of Savannah River, Thermal Infrared Image at Low River State (Heffner et al. 1998)

3.2.9.2 Onsite Streams and Impoundments

The source of most of the surface water on the SRS is either natural rainfall, which averages 48 inches annually, water pumped from the Savannah River to facilities, or groundwater discharging to the surface streams (WSRC 2007a). The streams, which historically have received varying amounts of effluent from SRS operations, are not commercial sources of water (DOE 2002). Table 3-16 provides an overview of time periods that onsite streams received reactor cooling water discharges.

The watersheds associated with the major SRS tributaries to the Savannah River are depicted in Figure 3-23. These tributaries drain all of SRS with the exception of a small area on the northeast side, which drains to a tributary of the Salkehatchie River (Kolka et al. 2005). This area does not contain any SRS facilities and was not considered in the CA analyses.

3.2.9.2.1 Savannah River Site Streams

3.2.9.2.1.1 Upper Three Runs

Upper Three Runs (Figure 3-11) is the longest of the site's streams. It drains an area of over 195 mi^2 and differs from the other five onsite streams in two respects. It is the only stream with headwaters originating offsite and it is the only stream that never received heated discharges of cooling water from the production reactors (WSRC 2007a). It is approximately 25 mi long, with its lower 17 mi within the SRS, and receives more water from underground sources than any other SRS stream.

The northern portion of the UTR watershed within the site boundary includes portions of A-Area, M-Area, and the SRNL. The southern portion of the UTR watershed includes the majority of the B-Area Administrative Center, S-Area, and Z-Area, as well as portions of the E-Area waste management complex, F- and H- Areas, and R Reactor Area (DOE 2005). In late 1988, UTR began receiving effluents from the F-/H-Area Effluent Treatment Facility (ETF), which discharges just downstream of the Road C bridge (below Station 02 in Figure 3-24) which is downstream from F-Area (See Figure 3-1) (Wike et al. 2006).

The UTR stream channel has a low gradient and is meandering, especially in the lower reaches. Its floodplain ranges in width from 0.25 to 1 mi and contains extensive stands (about 98% coverage) of bottomland hardwood forest. Within SRS, the UTR valley is asymmetrical, having a steep southeastern side and a gently sloping northwestern side (WSRC 2007a).

Table 3-16. Reactor Operations and Cooling Water Discharges to Onsite Streams

Reactor	Period of Reactor Operations	Reactor Cooling Water Discharge
C	March 28, 1955 – 1985	• 1955 – 1985 discharged to Fourmile Branch through Castor Creek
K	August 14, 1954 – 1988; June 8, 1992 (test run)	• 1954 – 1988 discharged to Pen Branch through Indian Grave Branch • K-Area Cooling Tower completed in 1992 • One day test run on June 8, 1992 discharged to the K-Area Cooling Tower
L	August 11, 1954 – 1968; 1985 - 1988	• 1954 – 1968 discharged directly to Steel Creek • L Lake completed in 1985 (Impoundment of Steel Creek) • 1985 – 1988 discharged to L Lake
P	February 20, 1954 – 1988	• 1954 – 1964 discharged directly to Steel Creek • 1958 PAR Pond completed (impoundment of upper reaches of Lower Three Runs Creek and its tributaries, Poplar Branch and Joyce Branch) • 1964 – 1988 discharged to PAR Pond
R	December 28, 1953 – 1964	• 1953 – 1958 discharged to Lower Three Runs through Joyce Branch • 1958 PAR Pond completed (impoundment of upper reaches of Lower Three Runs Creek and its tributaries, Poplar Branch and Joyce Branch) • 1958 – 1964 discharged to PAR Pond

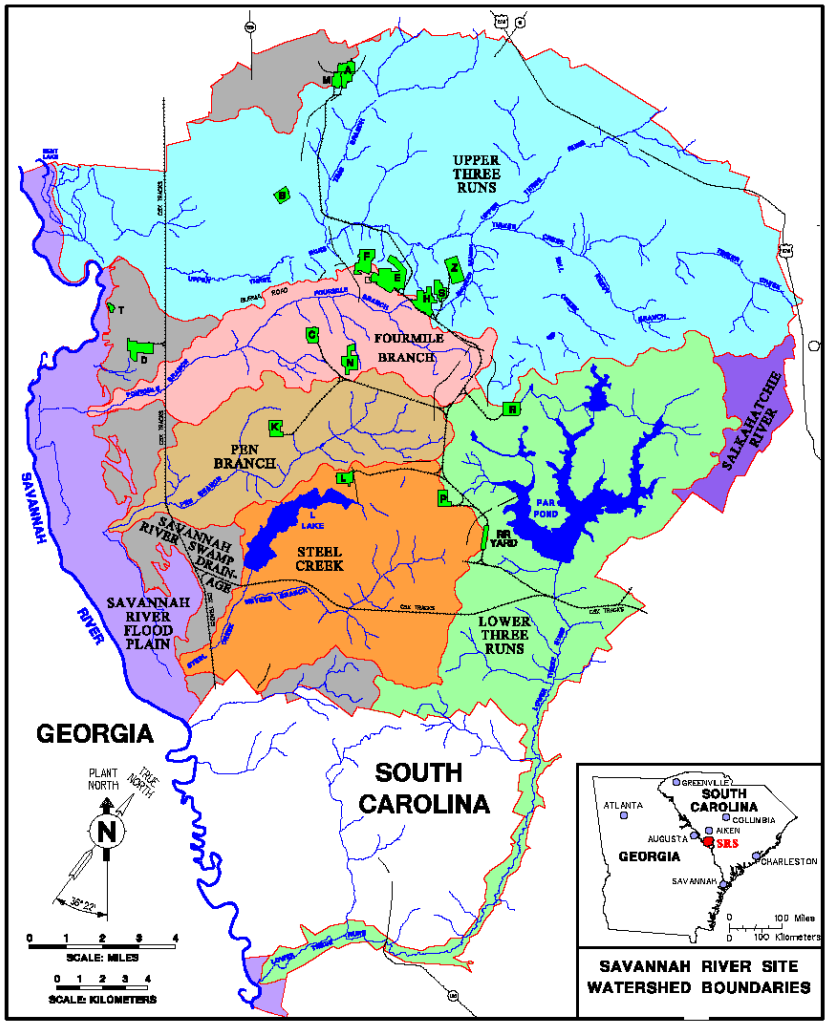


Figure 3-23. SRS Watershed Boundaries and Major Tributaries

Upper Three Runs is gauged near Highway 278 (see Figure 3-1) (station 02197300 relocated downstream), at SRS Road C (station 02197310), and at Highway 125 about 3 mi above the confluence of UTR with the Savannah River (station 02197315) (see Figure 3-1). Figure 3-24 shows the location of the flow measurement stations for UTR (from Wike et al. 2006). The Highway 278 station is a National Hydrologic Benchmark Station. Benchmark streams are measured monthly for water flow, temperature, and quality to provide hydrologic data on river basins governed by natural conditions (WSRC 2007a). The mean UTR flow reported for the time period between 1975 and 2000 is 240 cfs (Table 3-17) (Kolka et al. 2005).

Tims Branch, McQueen Branch, and Crouch Branch (Figure 3-11) are major tributaries to UTR which receive SRS effluents. The Tims Branch tributary receives industrial wastes from A-Area, M-Area, SRNL, and SREL, including nonprocess cooling water (Wike et al. 2006, WSRC 2007a). McQueen Branch and Crouch Branch originating in F- and H-Areas carry ambient temperature cooling waster and other effluents (Wike et al. 2006).

The Tims Branch (Figure 3-11) drainage basin is about 22.5 mi^2, most of which lies within SRS. The stream flow is to the south-southeast into UTR. It has a gradient ranging from 10 to 30 ft per mi. The valley is V-shaped, and the sides vary from steep to gently sloping. The floodplain is up to 1,000 ft wide. The drainage basin outside the SRS boundary includes areas of low density population and some farmland. For many years, Steed Pond was maintained; in recent years, DOE decided to not rebuild the failed dam structure on Steed Pond. Mean annual flow on Tims Branch is 5.63 cfs just before discharging into UTR (WSRC 2007a).

McQueen Branch (Figure 3-11) drains 4.4 mi^2 within SRS. McQueen Branch flows north into Tinker Creek, just above its intersection with UTR. The stream valley is V-shaped, with relatively steep sides (up to 100 ft high) and little floodplain. The hilltops within the drainage basin are gently sloping (WSRC 2007a). Stream flow measurements were taken in McQueen Branch from 1982 to 1984 as part of a hydrogeological study that included the effects of 48 storms. Three of these storms produced runoff exceeding the 30 cfs capacity of a weir located downstream of the DWPF site (drainage area 3.47 mi^2). The runoff included rainwater from H-Area, south of the DWPF site. The study indicated that the time lag between the beginning of a storm and an increased flow at the weir was usually less than 2 hours. The water-level rise generally took 1 to 3 hours to peak; the water-level decline took only slightly longer, and the impact of large storms usually was gone within 2 to 3 days (WSRC 2007a).

Crouch Branch (Figure 3-11) drains 1.2 mi^2 within SRS. Crouch Branch flows northwest into UTR and has a topography similar to that of McQueen Branch. Flow determinations have not been made for Crouch Branch (WSRC 2007a).

3.2.9.2.1.2 Fourmile Branch

Within various SRS documents Fourmile Branch has also been denoted as Four Mile Creek or Fourmile Creek. Within the text of this document the stream will be denoted as Fourmile Branch. However, figures within this document will retain the designation originally used within the source documents.

Fourmile Branch (Figure 3-11) drains about 23 mi^2 within SRS, including much of the F-, H-, and C-Areas. The creek flows to the southwest into the Savannah River Swamp and then into the Savannah River. The valley is V-shaped, with the sides varying from steep to gently sloping. The floodplain is up to 1,000 ft wide (WSRC 2007a). Portions of F- and H-Area, and all of C-Area, lie in the Fourmile Branch watershed (DOE 2005).

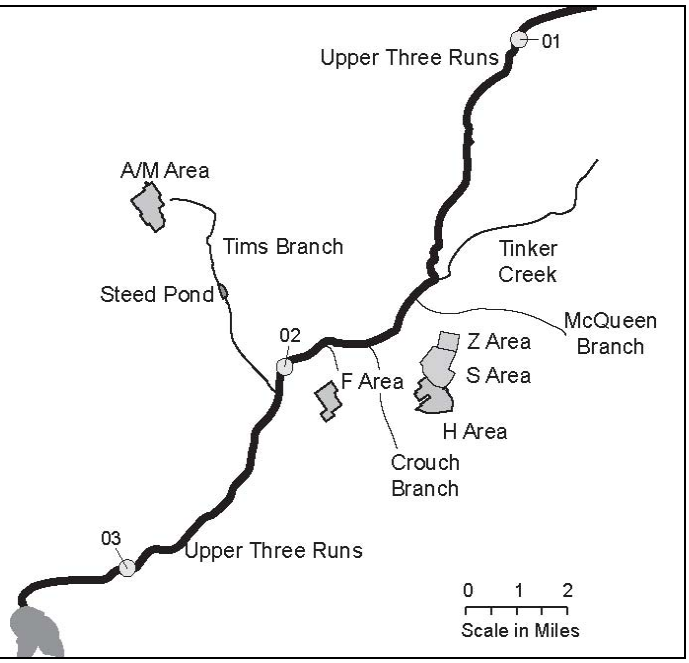
01 near New Ellenton (Highway 278) 02 just downstream of F-Area 03 at Highway 125 at SRS	

Figure 3-24. Flow Measurement Stations for Upper Three Runs (from Wike et al. 2006)

From 1955 until June 1985, Fourmile Branch received large volumes of cooling water from the production reactor in C-Area via Castor Creek (Table 3-16). The creek valley has been modified by the cooling water discharge, which has created a delta into the Savannah River Swamp (WSRC 2007a). Fourmile Branch receives effluents from National Pollutant Discharge Elimination System (NPDES) outfalls in C-, F-, H-, and N-Areas. Beginning in 1995, effluent from the 1.05-million gallon per day Centralized Sanitary Wastewater Treatment Facility also began discharging to Fourmile Branch (Wike et al. 2006).

Table 3-17. Mean SRS Stream Flows with and without Reactor Cooling Water Discharges

Stream	Flow with Reactor Cooling Water Discharges		Flow without Reactor Cooling Water Discharges [2]	
	cfs	Data Dates	cfs	Data Dates
Upper Three Runs	NA [1]	NA [1]	240 [3]	1975 – 2000
Beaver Dam Creek [2]	80 [3]	1976 – 2000	Negligible [4]	Not available
Fourmile Branch	390 [5]	1955 – 1985 (during periods of C-Reactor operations)	33 [3]	1987 – 2000
Pen Branch	290 [6]	1985 – 1988 (with K-Reactor operations)	36 [3]	1993 – 2000
Steel Creek	260 [6]	1986 – 1988 (with L-Reactor operations discharging to L Lake)	35 [7] 60 [3]	Estimated natural flow prior to L Lake 1994 – 2000 (after L Lake impoundment)
Lower Three Runs	Stream flow increase of approximately 200 cfs with R-Reactor Operations [4]	1953 – 1958 (Prior to PAR Pond completion)	82 [3, 8]	1975 – 2000

[1] Not Applicable: Upper Three Runs never received reactor cooling water discharges.
[2] While Beaver Dam Creek has never received reactor cooling water discharges, its flow prior to SRS operations was probably only intermittent or very low. Beaver Dam Creek received cooling water discharges from the Heavy Water Plant from 1952 to 1982 and from the D-Area coal-fired power plant from 1952 to the present. The flow data provided above is shown in relation to flow with and without the D-Area cooling water discharges.
[3] Kolka et al. 2005
[4] Wike et al. 2006
[5] DOE 1986 (DOE/EIS-0121D)
[6] DOE 1990 (DOE/EIS-0147); During actual reactor operations, K-Area discharged approximately 400 cfs to Pen Branch through Indian Grave Branch; L-Area discharged approximately 400 cfs to L-Lake.
[7] DOE 1984 (DOE/EIS/0108)
[8] During operations of P and R reactors, PAR Pond was operated as a recirculating cooling water reservoir. Water losses due to evaporation, seepage, and overflow from PAR Pond were compensated by pumping makeup water from the Savannah River. Therefore from a flow standpoint, flow in Lower Three Runs after PAR Pond operations began was essentially unimpacted by Reactor operations (DOE 1990 [DOE/EIS-0147], Kolka et al. 2005).

Water flow measurements have been made at several locations on Fourmile Branch from 1976 to 2002 (Figure 3-25 and Wike et al. 2006). The mean flow rate reported for this stream during periods of C Reactor operations (1955 through 1985) is 390 cfs (Table 3-17). After C Reactor was shut down, the mean flow rate for Fourmile Branch is reported to be on the order of 33 cfs (Kolka et al. 2005).

01 just downstream of C-Area
02 near D-Area

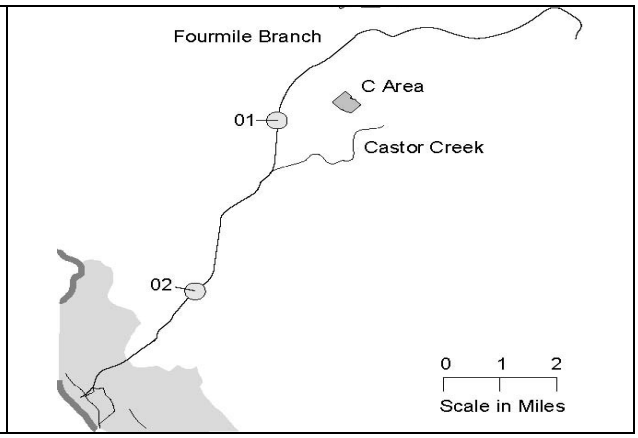


Figure 3-25. Flow Measurement Stations for Fourmile Branch (Wike et al. 2006)

3.2.9.2.1.3 Beaver Dam Creek

Beaver Dam Creek (Figure 3-11), which is approximately 3 mi long and drains an area of about 0.85 mi^2, originates at the effluent outfall canal of D-Area and flows south, parallel to Fourmile Branch, to the Savannah River (Figure 3-26). The creek is deep, narrow and channelized in its headwaters and near its mouth, but opens into a slough-like channel in its midreaches (Wike et al. 2006). D-Area is the only SRS operational area in the Beaver Dam Creek watershed.

Observations on the drainage topography of Beaver Dam Creek indicate that it was an intermittent flowing stream before SRS operation. The stream is located 1 to 2 mi west of Fourmile Branch and flows in a southwest direction from D-Area through the swamp to the Savannah River. Beaver Dam Creek received effluent from both the heavy water production plant and the coal-fired generating station from 1952 until 1982. The heavy water production plant was placed on standby in 1982.

Currently, Beaver Dam Creek receives cooling water from the coal-fired powerhouse located in D-Area, which provides electricity and steam for site use, as well as sanitary waste water, ash basin effluent waters, and various laboratory waste waters. The mean flow rate reported for Beaver Dam Creek between 1976 and 2000 is 80 cfs, which represents the flow including cooling water discharges (Table 3-17) (Kolka et al. 2005).

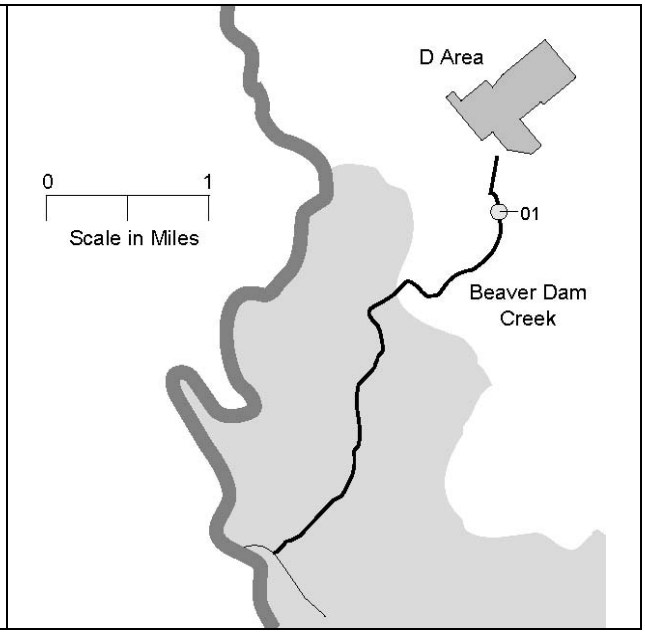
01 at D-Area	

Figure 3-26. Flow Measurement Stations on Beaver Dam Creek (Wike et al. 2006)

3.2.9.2.1.4 Pen Branch and Indian Grave Branch

Pen Branch (Figure 3-11) follows a path roughly parallel to Fourmile Branch until it enters the Savannah River Swamp. The only significant tributary to Pen Branch is Indian Grave Branch (Figure 3-11), which flows into Pen Branch about 5 mi upstream from the swamp. Pen Branch enters the swamp about 3 mi from the Savannah River. It flows directly toward the river for about 1.5 mi and then turns and runs parallel to the river for about 5 mi before discharging into Steel Creek at about 0.5 mi from its mouth. Pen Branch and Indian Grave Branch drain about 22 mi^2 of watershed upstream from the swamp. The Pen Branch watershed includes the entirety of K-Area (K Reactor), portions of N-Area (Central Shops), and waste units associated with L-Area (L Reactor) (DOE 2005).

Pen Branch received heat exchanger cooling water from K Reactor through Indian Grave Branch from 1954 to 1988 (Table 3-16 and Figure 3-27) (WSRC 2007a). Currently, the Pen Branch system receives nonthermal effluents, including nonprocess cooling water, sanitary wastewater, and other effluents from K-Area and sanitary effluent from N-Area.

Gauging station 01 in Figure 3-27 is upstream of the confluence of Indian Grave Branch with Pen Branch, and as expected, flow rates measured at this upstream location were considerably smaller than those measured downstream (Station 02) of the point of discharge during the time period including reactor operations (WSRC 2007a). Mean flow in Pen Branch when K Reactor was operating is reported to be on the order of 290 cfs (Table 3-17). After shutdown of K Reactor, mean flow is reported to be on the order of 36 cfs (Table 3-17).

01 at K-Area
02 near the Savannah River Swamp; see Figure 3-11

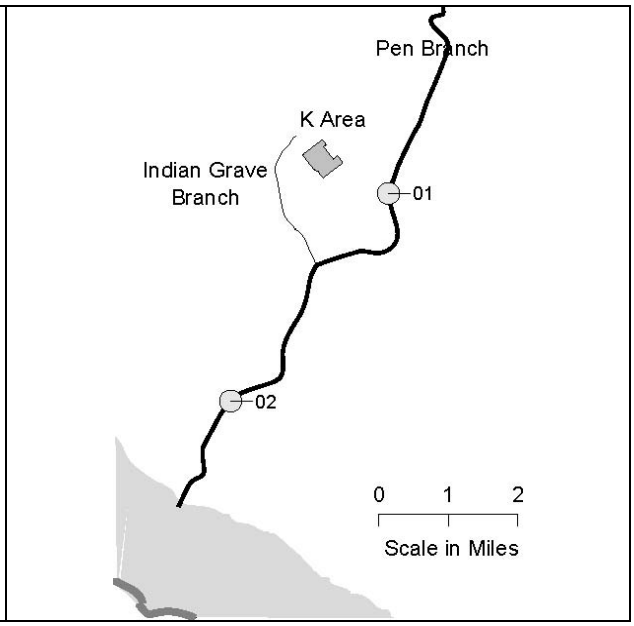


Figure 3-27. Flow Measurement Stations for Pen Branch (Wike et al. 2006)

3.2.9.2.1.5 Steel Creek

The headwaters of Steel Creek (Figure 3-11) originate near P Reactor, southwest of PAR Pond. The creek flows southwest for about 1.8 mi before it enters the headwaters of L Lake, an impoundment of the creek by L Dam, constructed in 1985 (Figure 3-28). The lake is 4.3 mi long and relatively narrow, with an area of about 1,034 acres. Flow from the outfall of L-Lake dam travels about 3 mi before entering the Savannah River Swamp and then another 1.8 mi before entering the Savannah River. In the swamp, it is joined by the flow from Pen Branch and part of the flow from the Fourmile Branch/Beaver Dam Creek system (Wike et al. 2006). The drainage area of Steel Creek and its main tributary, Meyers Branch, is about 35 mi^2 and includes portions of P and L Areas (DOE 2005).

Meyers Branch, the main tributary of Steel Creek, flows approximately 6.2 mi before entering Steel Creek. The confluence of Steel Creek and Meyers Branch is downstream from the L-Lake dam and upstream from Highway 125 (Figure 3-1 and Figure 3-28) (Wike et al. 2006).

Steel Creek formerly received cooling water discharges from two reactors (L and P). The discharge of cooling water effluent from P-Reactor to Steel Creek was discontinued in 1964 when coolant supply for this reactor was switched to recirculated cooling water from PAR Pond (see Table 3-16). Thermal discharge from L Reactor ceased in 1968, when the reactor was placed in standby condition. L Reactor was restarted in 1985, after construction of L Lake, and shutdown in 1988 (see Table 3-16) (WSRC 2007a). L Lake was constructed to receive and cool the heated effluents from L Reactor prior to their release into Steel Creek, but did not provide recirculated cooling or makeup water to L Reactor.

Prior to 1985, the USGS maintained a continuous flow recorder on Steel Creek at Old Hattiesville Bridge (station 02197359), which is located about 0.5 mi upstream of the confluence with the onsite swamp. This station has since been moved to Highway 125 (USGS station 021973565, shown as Station 01 in Figure 3-28). The mean flow reported for Steel Creek during the time span in which the reactor was operating (1986-1988) is 260 cfs (Table 3-17). The estimated mean natural flow prior to the construction of L Lake is 35 cfs, while the reported mean flow after reactor operations ceased is 60 cfs (Table 3-17).

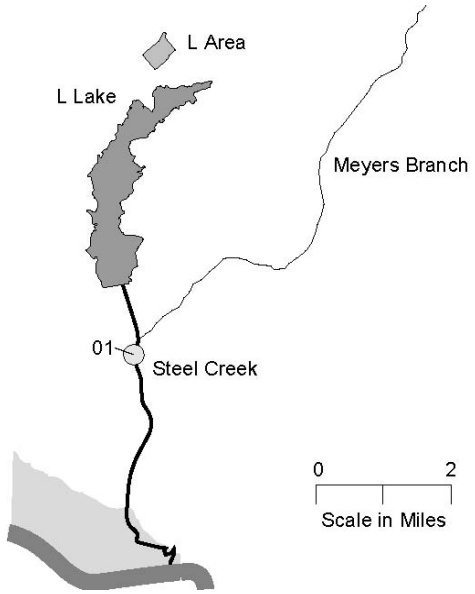


Figure 3-28. Flow Measurement Station for Steel Creek at Highway 125 (Wike et al. 2006)

3.2.9.2.1.6 Lower Three Runs

Lower Three Runs (Figure 3-11) has the second largest watershed (about 178 mi^2) of streams within the SRS. PAR Pond, a 2,500-acre impoundment completed in 1958 by constructing an earthen dam (Cold Dam) on LTR, has three main branches which follow the former streambeds and drainage areas of the upper reaches of LTR and its tributaries, Poplar Branch and Joyce Branch. Below the dam, LTR flows in a southerly, then southwesterly course for about 24 mi to the Savannah River (Figure 3-29). Several small tributaries draining portions of SRS flow into the creek in its lower reaches (Wike et al. 2006, WSRC 2007a). The majority of the LTR watershed is located outside of the SRS boundary (Kolka et al. 2005). R-Area, a portion of P-Area, ecological laboratories and various soil and groundwater project waste sites lie within the watershed, as do industrial facilities located outside of the eastern SRS boundary (DOE 2005).

Several other ponds were constructed in the headwaters above PAR Pond to improve cooling of reactor effluent. Pond B, the largest, has an area of about 180 acres (Wike et al. 2006). Before construction of PAR Pond, effluent cooling water from R Reactor was discharged through Joyce Branch to LTR. From 1958 through 1964, PAR Pond received effluents only from R Reactor (Table 3-16). R Reactor was shut down in 1964, and all P Reactor cooling water was diverted from Steel Creek to PAR Pond in that year, and continued until P Reactor was shut down in 1988 (Table 3-16).

PAR Pond overflows to the LTR. In addition, about 5.3 cfs seeps through and under the dam to enter LTR. This seepage is usually several degrees cooler than the surface water in the pond during the summer months.

The mean flow rate reported for LTR below PAR Pond from 1975 to 2000 is 82 cfs (Table 3-17). Although this time period of measurement includes a period of P Reactor operations (until 1988, Table 3-16), flow in LTR below the dam was essentially unimpacted by reactor operations due to the operation of PAR Pond as a recirculating reservoir (DOE 1990, Kolka et al. 2005). Prior to PAR Pond construction, when LTR received R Reactor effluents via Joyce Branch, mean flow estimates are not available. However, during this early period, it is estimated that natural flow was augmented by about 200 cfs during reactor operations.

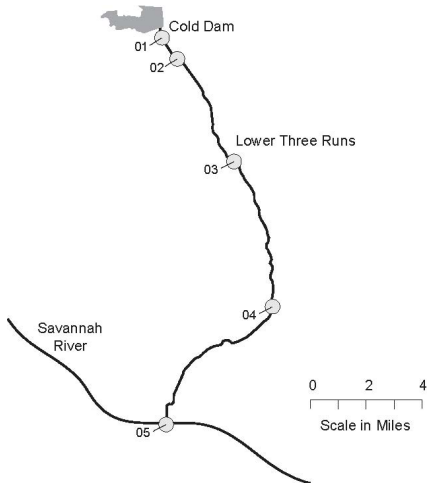


Figure 3-29. Lower Three Runs below PAR Pond (numbered circles refer to biological sampling stations, from Wike et al. 2006)

3.2.9.2.2 Impoundments on the Savannah River Site

Several artificial impoundments exist on the SRS, with the largest being PAR Pond and L Lake. PAR Pond, the largest impoundment on SRS (Figure 3-11), is an artificial lake located in the eastern part of the site that covers approximately 2,500 acres. PAR Pond was created in 1958 by constructing an earthen dam on LTR and was used to receive thermal effluent from P and R Reactors. A pumphouse on PAR Pond recirculated water from PAR Pond to P-Area, where it was mixed with makeup water from the Savannah River and reused in reactor operations. R Reactor used PAR Pond from August of 1958 until it ceased operation in July 1964; P Reactor used PAR Pond from 1964 until it was shut down in 1988 (Table 3-16). In 1995, DOE decided to allow the water level in PAR Pond to fluctuate naturally near its operating level (200 ft above msl), but not allowing the water level to fall below 195 ft (WSRC 2007a). Water has not been pumped from the Savannah River to PAR Pond since 1995 (Wike et al. 2006). Pond C, a 140-acre "precooler" body of water, is separated from PAR Pond by a dam and is part of the P Reactor effluent canal system. PAR Pond receives discharges from all storm sewer outfalls from the deactivated R-Area and from a few storm sewers in P-Area.

A second large artificial impoundment, L Lake (Figure 3-11), lies in the southern portion of SRS and covers approximately 1,034 acres. As noted earlier, L Lake was created by the impoundment of the Steel Creek by L Dam, constructed in 1985, as a once-through cooling water reservoir for L Reactor (Figure 3-28). The lake is 4.3 mi long and relatively narrow (0.4 to 0.8 mi) (Wike et al. 2006). L Reactor discharged thermal effluents to L Lake via an effluent canal, which along with P-Area process water and ambient Steel Creek flow were released through the dam to be returned to the Savannah River (Wike et al. 2006). Water from the Savannah River is still pumped to L Lake to maintain the normal operating level of 190 ft and the flow in Steel Creek (downstream of L Lake) greater than 10 cfs (WSRC 2007a).

Water from both PAR Pond (200 ft) and L Lake (190 ft) drains to the south via LTR and Steel Creek, respectively, into the Savannah River. Water is also retained intermittently in natural lowland and upland marshes and natural basins, some of which are Carolina bay depressions (WSRC 2007a).

3.2.9.3 Location of Surface Water Relative to Savannah River Site Operational Areas

3.2.9.3.1 General Separations Area

3.2.9.3.1.1 E-Area

Area surface water bodies near the E-Area LLW Burial Grounds consist of Fourmile Branch and UTR, and their tributaries (Figure 3-30). There are no surface waters, including intermittent streams, within 2,000 feet of the E-Area LLW Burial Grounds, and this facility is not located in flood-prone areas (WSRC 2000a). From the original E-Area burial grounds (ORWBG), surface flow is southwest towards a small tributary of Fourmile Branch, which empties into the Savannah River (WSRC 2007a). From the currently operating ELLWF, surface flow is northeast towards Crouch Branch, a small tributary of UTR, which empties into the Savannah River.

3.2.9.3.1.2 F- Area

F-Area is on a relatively elevated portion of the SRS, between UTR and an unnamed tributary of Fourmile Branch (Figure 3-30). Approximately half of F-Area drains into each of the two streams. Surface elevations of F-Area range from about 270 to 320 ft above msl (DOE 2002). For example, the F-Canyon building site is at an elevation of over 300 ft above msl. The nearest significant stream to F Canyon is UTR. It is located about 0.7 mi north and west of the F-Canyon facility. This creek flows at elevations below 150 ft. Runoff from precipitation is diverted into storm sewers, then discharged to an unnamed tributary of UTR, which empties into the Savannah River (WSRC 2007a).

3.2.9.3.1.3 H- Area

H-Area is also located on a relatively elevated portion of the SRS between UTR and Fourmile Branch (Figure 3-30). Similarly to F-Area, about half of H-Area drains into each of the two streams. Surface elevations of H-Area range from about 270 to 320 ft above msl (DOE 2002). The nearest significant stream is UTR, located about 1.6 mi north of H-Area. UTR flows at elevations less than 150 feet above msl. Runoff from precipitation is carried away from structures by natural contours or catch basins that divert water into the UTR watershed. UTR empties into the Savannah River (WSRC 2007a).

3.2.9.3.1.4 S-and Z-Areas

S- and Z-Areas are located on a local topographic high (minimum grade level 275 ft msl). S-Area is within the Savannah River drainage basin at the divide between Crouch Branch and McQueen Branch watersheds. Z-Area is located north of S-Area. Runoff from Z-Area is diverted indirectly to McQueen Branch (Figure 3-30). McQueen Branch drains into Tinker Creek near its junction with UTR, and Crouch Branch drains directly into UTR. All streams in the area are at substantially lower elevations than S- and Z-Areas (WSRC 2007a).

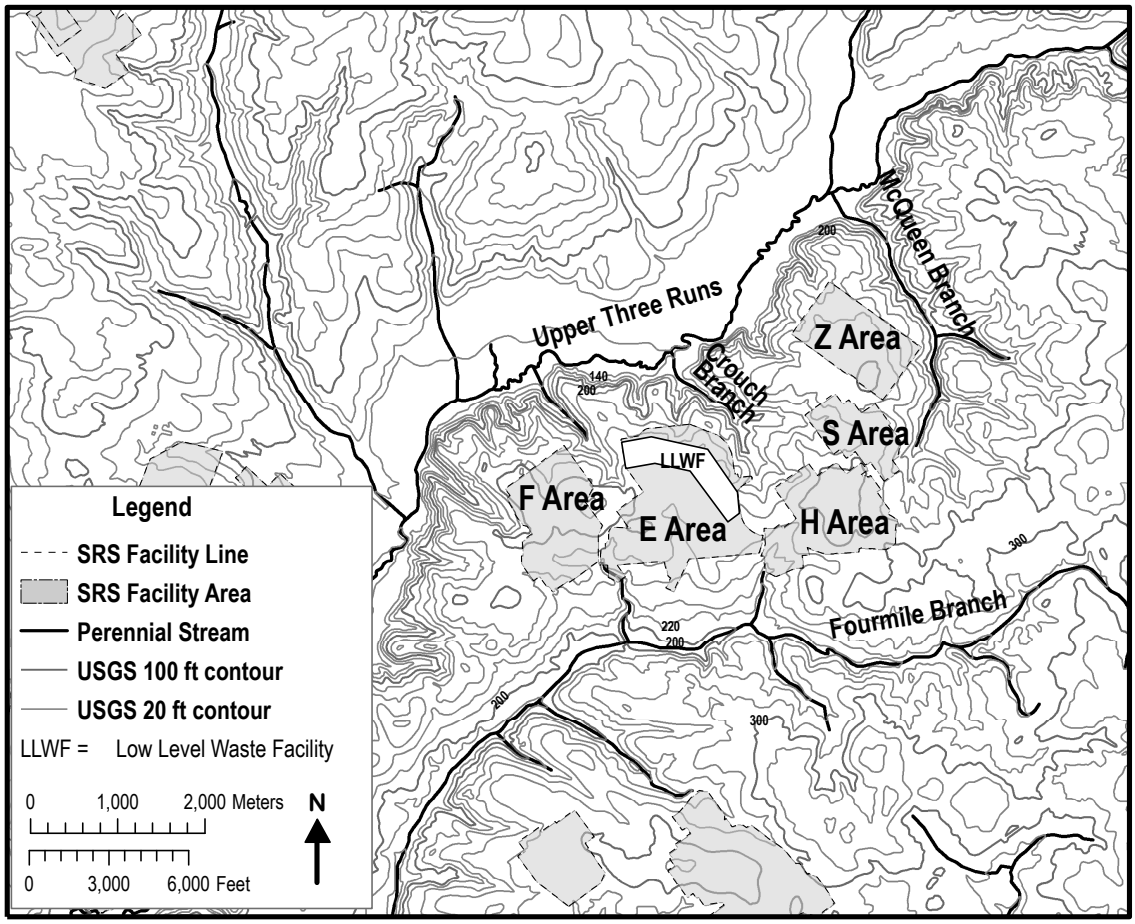


Figure 3-30. Topographic Map Showing the Location of McQueen and Crouch Branch Tributaries to Upper Three Runs in the General Separations Area

3.2.9.3.2 A/M Areas

3.2.9.3.2.1 A-Area and the Savannah River National Laboratory

A-Area, which includes SRNL, is located on a hilltop at elevations ranging from 350 to 390 ft above msl. Surface drainage is away from this area, with radial flow to the east toward Tims Branch. Tims Branch historically flowed into an 11.2 acre impoundment called Steed Pond (until the dam barrier failed and was not repaired) about 1.25 mi southeast of the 3/700-Areas before draining into UTR. Runoff from this area that does not reach Tims Branch either infiltrates the soil or drains to the northwest into an unnamed, intermittent surface channel. This surface channel flows southward into the Savannah River Swamp (WSRC 2007a).

3.2.9.3.2.2 M-Area

M-Area is located on a hilltop at elevations ranging from 350 to 380 ft above msl. Surface drainage is to the east toward Tims Branch, which is located about 0.75 mi east of M-Area (Figure 3-24), and flows at elevations below 250 ft. Tims Branch historically flowed into an 11.2 acre impoundment called Steed Pond (until the dam barrier failed and was not repaired) about 1.25 mi southeast of M-Area before draining into UTR. Runoff from M-Area that does not reach Tims Branch either infiltrates the soil or drains to the northwest into an unnamed, intermittent surface channel. This surface channel flows southward into the Savannah River Swamp. The western portion of M-Area drains into a small, unnamed, intermittent stream outside SRS. This stream flows around the northwest boundary of SRS and empties into a swamp near the Savannah River (WSRC 2007a).

3.2.9.3.3 D/T Areas

3.2.9.3.3.1 D-Area

D-Area is located at an elevation slightly above the maximum flood level for the Savannah River (above 118 ft above msl), approximately 1 mi east of the Savannah River (WSRC 2000a, 2007a). D-Area is drained by Beaver Dam Creek (See Figure 3-26), which also receives effluent flow from power generating facilities and miscellaneous operations in D-Area (WSRC 2007a).

3.2.9.3.3.2 T-Area

T-Area (alternatively referred to as TNX Area) is located one quarter mi east of the Savannah River, on a terrace between UTR to the north, and Fourmile Branch to the south (Figure 3-11), at an elevation 150 ft above msl (Nichols 1993). The local topography is relatively flat, with a slope eastward away from the Savannah River. A portion of the Savannah River floodplain lies immediately west of the area, at 95 ft above msl. A small levee divides the flood plain and serves as the bank of the river during high stages.

3.2.9.3.4 C-Area

C-Area is located on the Aiken Plateau, at an average ground surface elevation of 285 ft above msl. The area is bounded to the west by Fourmile Branch, to the south by Castor Creek, to the north by unnamed tributaries of Fourmile Branch (WSRC 2002b). C-Area is within the Fourmile Branch watershed (Figure 3-23 and Figure 3-25) on a hilltop with radial drainage of surface water (WSRC 2004a, b). The two principal tributaries to Fourmile Branch in the vicinity of C-Area are the Twin lakes tributary, and Castor Creek, which receive runoff during rain events (WSRC 2004b).

3.2.9.3.5 K-Area

K-Area is located within the Pen Branch Watershed (Figure 3-23 and Figure 3-27), approximately 100 ft above the elevation of Pen Branch. Surface drainage is to the southeast toward an unnamed tributary of Pen Branch in the north and east portion of the area (WSRC 1998a). In the western portion of K-Area, surface drainage is toward the west-southwest, to an unnamed tributary of Indian Grave Branch (WSRC 1998b), the latter being a tributary to Pen Branch (Figure 3-27).

3.2.9.3.6 L-Area

L-Area is located on a surface water divide between the Pen Branch watershed to the northwest, and the Steel Creek watershed to the southeast (Figure 3-23). In the southeastern part of L-Area, surface water flows into L-Lake (WSRC 2005a).

3.2.9.3.7 P-Area

Within the P Reactor fence in P-Area, the elevation is greater than 330 ft, and higher than the surrounding land. Surface drainage on the west side of P-Area is to the southwest, toward Steel Creek (Figure 3-11). Headwaters of Steel Creek are at approximately 280 ft above msl. Surface drainage on the east side of P-Area is toward unnamed tributaries to PAR Pond (WSRC 2004c).

3.2.9.3.8 R-Area

R-Area is located about 1.7 mi northwest of PAR Pond (Figure 3-11). Topographic relief in this area is low, with elevations generally ranging from 290 to 300 ft above msl. In the northern portion of the area, where the seepage basins lie, surface runoff flows both north and west into Mill Creek (a tributary to UTR), or east into the R-Effluent Canal (WSRC 2002c). The R-Reactor Discharge Canal, located northeast of R-Area, carried water from R-Reactor to PAR Pond after 1958 (WSRC 2006b).

3.2.9.3.9 N-Area

N-Area lies within the Fourmile Branch Watershed, approximately equidistant from Pen Branch to the southeast and Fourmile Branch to the northwest (See Figure 3-23). The topography is relatively flat, sloping gently to the north. Surface waters appear to drain into an intermittent stream, which feeds directly into Fourmile Branch, approximately 2 mi to the northwest (WSRC 1998c).

3.2.10 Groundwater Hydrology

The Southeastern Coastal Plain hydrogeologic province comprises a multilayered hydraulic complex in which retarding beds composed of clay and marl are interspersed with beds of sand and limestone that transmit water more readily. Ground water flow paths and flow velocity for each of these units are governed by the unit's hydraulic properties, the geometry of the particular unit, and the distribution of recharge and discharge areas.

Figure 3-31 shows several key boring locations within the SRS boundaries from which hydrogeologic data have been extracted. The primary purpose for installing the majority of wells at the SRS was to monitor the migration of dissolved contaminants emanating from waste disposal units. Samples are routinely collected from these wells, and water levels monitored, and both groundwater chemistry data and data describing water levels are stored in the Environmental Restoration Data Management System (ERDMS) database (Hiergesell and Jones 2003). Much of the following discussion has been derived from the large quantity of data collected over the years of site operations.

The basement complex, designated the Piedmont hydrogeologic province (Figure 3-13), consists of Paleozoic crystalline rocks, and consolidated to semi-consolidated Upper Triassic sedimentary rocks of the Dunbarton basin. The hydrogeology of this province was studied intensively at the SRS to assess the feasibility of storing radioactive waste in these rocks. The rocks are massive, dense, and practically impermeable except where fracture openings are encountered. The low aquifer permeability and poor water quality render these basement rocks undesirable for water supply in the SRS region (WSRC 2000a).

This section is organized as follows. Site wide groundwater hydrology corresponding to hydrologic units within the Coastal Plain province will be introduced first. Regional potentiometric surface maps corresponding to the hydrologic units are provided, as well as a generalized description of the units present throughout the site. Following the general SRS groundwater hydrology subsection, groundwater hydrology details specific to the SRS operational areas will be provided in separate subsections. In the more detailed subsections, local hydraulic properties, recharge and discharge areas, and local water table potentiometric surfaces are discussed. Plume maps, when applicable, are provided.

3.2.10.1 Savannah River Site General

The groundwater flow system at the SRS consists of four major aquifers separated by confining units. Flow in recharge areas generally migrates downward as well as laterally – eventually either discharging into the Savannah River and its tributaries, or migrating into the deeper regional flow system (Figure 3-32) (Mamatey 2006).

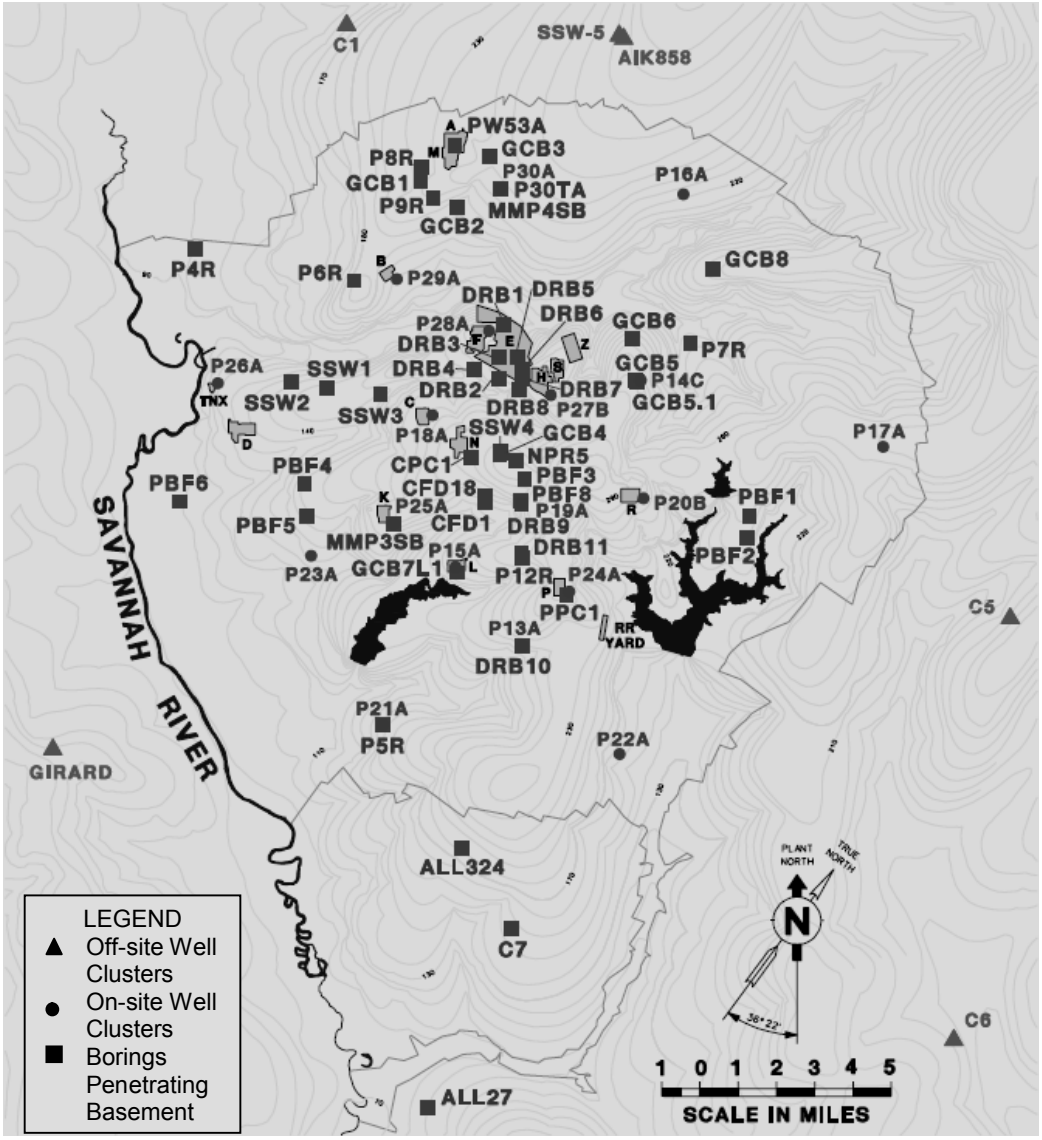


Figure 3-31. Location and Types of Reference Wells for Hydrostratigraphic Units (from WSRC 2000a)

The hydrologic units recognized in the SRS area are referred to in terms of hydrogeologic systems (WSRC 2000a). These systems have been grouped into three aquifer systems divided by two confining systems, all of which are underlain by the Appleton confining system. The Appleton separates the Southeastern Coastal Plain hydrogeologic province from the underlying Piedmont hydrogeologic province. The regional lithostratigraphy and hydrostratigraphic subdivisions of the Southeastern Coastal Plain hydrogeologic province are shown in Figure 3-13, and a cross-sectional view showing the hydrologic units in relation to the respective hydrogeologic systems is provided in Figure 3-33 (WSRC 2000a).

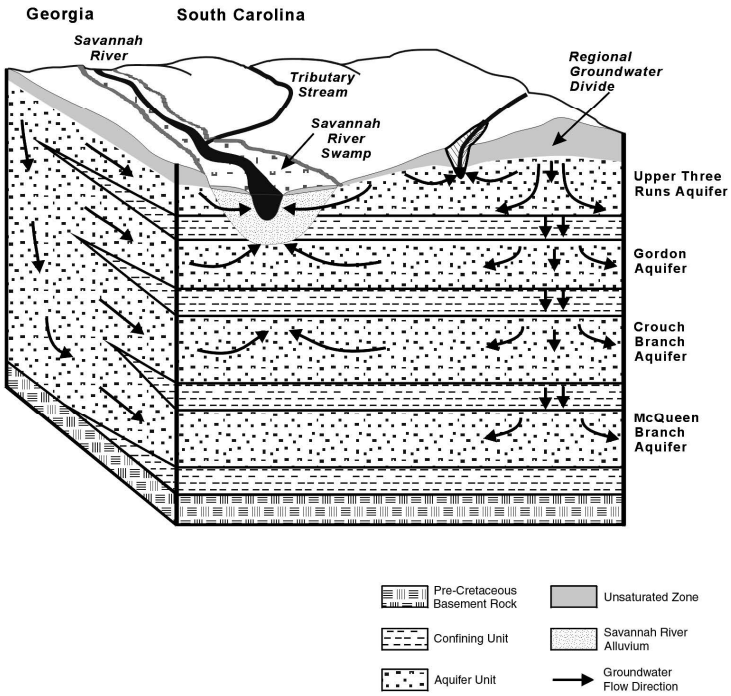


Figure 3-32. Generalized Groundwater Flow at the SRS (from Mamatey 2006)

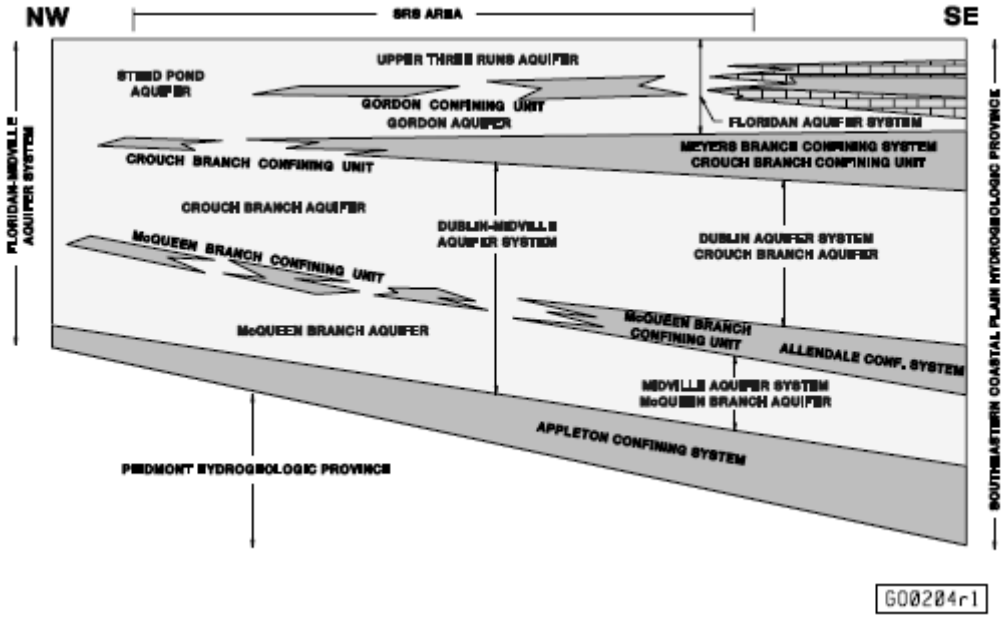


Figure 3-33. Cross-Section Showing Hydrologic Units in Relation to Hydrogeologic Systems at the SRS (from WSRC 2000a)

In descending order, the aquifer systems beneath SRS are the Floridan Aquifer System, the Dublin Aquifer System, and the Midville Aquifer System (Figure 3-13). In the northern portion of the site, only one system is recognized – the Floridan-Midville System, although confining units exist within this system such that individual aquifer units are defined. In descending order, the confining systems are the Meyers Branch confining system, the Allendale confining system, and the Appleton confining system (see Figure 3-13).
Figure 3-34 shows the limits of the recognized hydrogeologic systems on the SRS, in relation to the extent of the confining systems present. Beneath the SRS, the Midville and Dublin Aquifer Systems each consists of a single aquifer, the McQueen Branch aquifer and Crouch Branch aquifer, respectively (WSRC 2000a). The Floridan Aquifer System beneath the SRS consists of two aquifers in the southern and central portion of the site, and one aquifer in the northern portion of the site.

3.2.10.1.1 Floridan Aquifer System

The Floridan Aquifer System consists of two aquifers in the SRS region, the UTR aquifer unit, and the underlying Gordon aquifer unit, which are separated by the Gordon confining unit. Northward, the Gordon and UTR aquifer units coalesce to form the Steed Pond aquifer (Figure 3-33). This system overlies the Meyers Branch Confining system throughout the lower two-thirds of the SRS region. Toward the north, the confining beds of the Meyers Branch confining system thin, become intermittent, and the entire Floridan Aquifer System coalesces with the Dublin-Midville Aquifer System to form the Floridan-Midville Aquifer System (see Figure 3-13 and Figure 3-33).

3.2.10.1.1.1 Upper Three Runs Aquifer Unit

The Upper Three Runs aquifer unit occurs between the water table and the Gordon Confining Unit (when continuous), and includes the heterogeneous sediments in the Barnwell Group as well as sandy sediments of the Tinker/Santee Formation (Figure 3-13). The UTR aquifer is defined by the hydrogeologic properties of the sediments penetrated in well P-27, near UTR in the center of the SRS (Figure 3-31). In the center of the SRS, the aquifer is 40 m thick, consisting of sand, clayey sand, and interbedded clay (WSRC 2000a).

On the SRS, the hydraulic-head distribution characteristic of the UTR aquifer is controlled by the location and depth of incisement of creeks that dissect the site. The incisement of these naturally-occurring streams and their tributaries has divided the interstream areas of the water table aquifer into “groundwater islands,” each behaving as an independent hydrologic subset of the aquifer with unique recharge and discharge areas. The streams act as the groundwater discharge boundary for the interstream area (Hiergesell 1998). The head distribution pattern in these groundwater islands tends to follow topography, with groundwater divides being present near the center of the interstream areas (WSRC 2000a). Water table elevation contours for the SRS (Figure 3-35) reach approximately 80 m (260 ft) above msl in the northern areas of the site, reach a high of about 94 m (310 ft) above msl near the centers of the site, and decrease to as low as about 21 m (70 ft) above msl near the mouth of LTR, where it flows into the Savannah River (Hiergesell 1998).

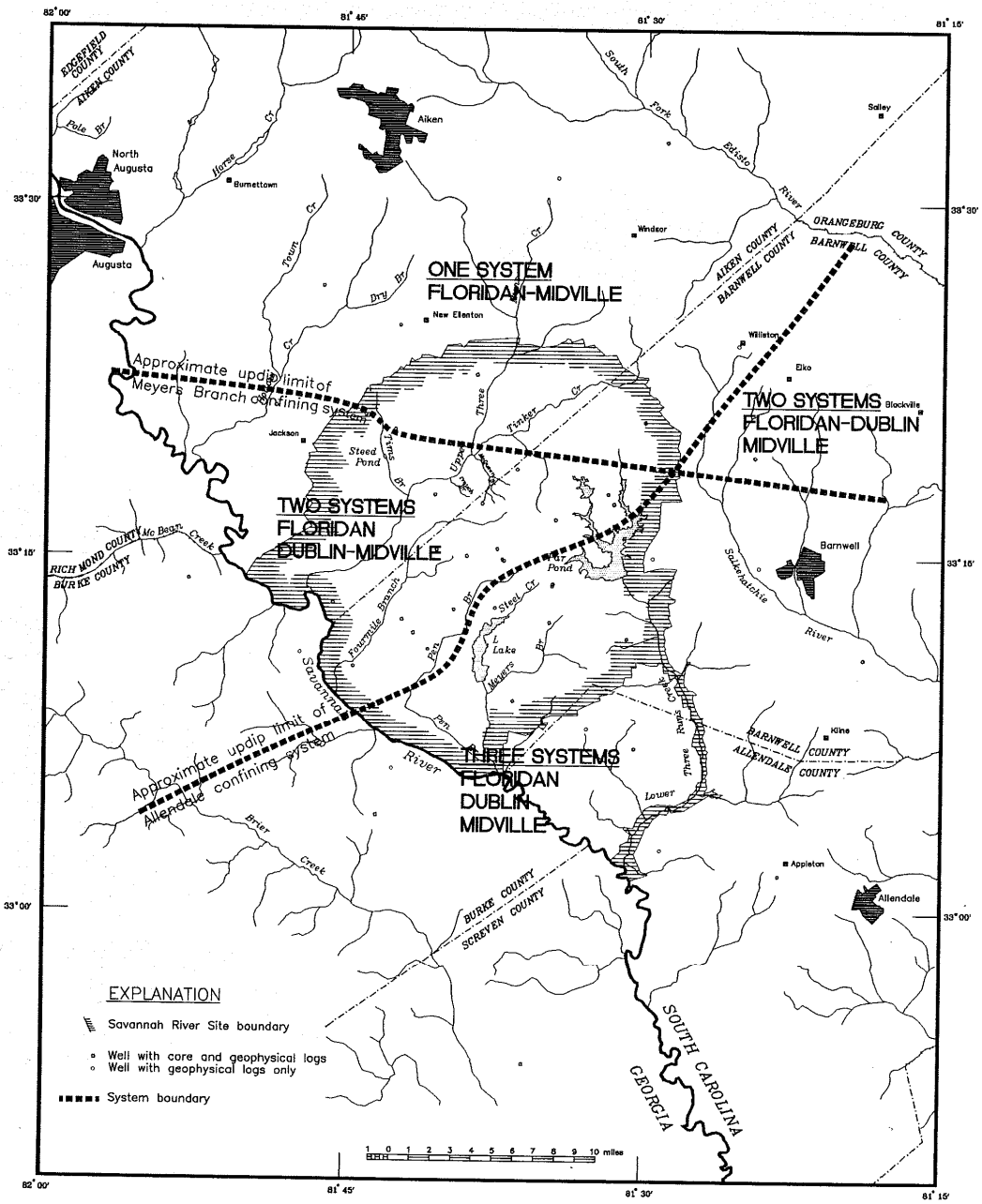


Figure 3-34. Limits of Meyers Branch and Allendale Confining Systems in the SRS Region (from Aadland et al. 1995)

Recharge to the UTR aquifer occurs at the water table by infiltration downward from the land surface. In the “upper" aquifer zone, part of this groundwater moves laterally toward the bounding streams while part moves vertically downward. The generally low vertical hydraulic conductivities of the “upper" aquifer zone and the intermittent occurrence of the “tan clay" confining zone retard the downward flow of water, producing vertical hydraulic-head gradients in the "upper" aquifer zone and across the "tan clay” confining zone (WSRC 2000a).

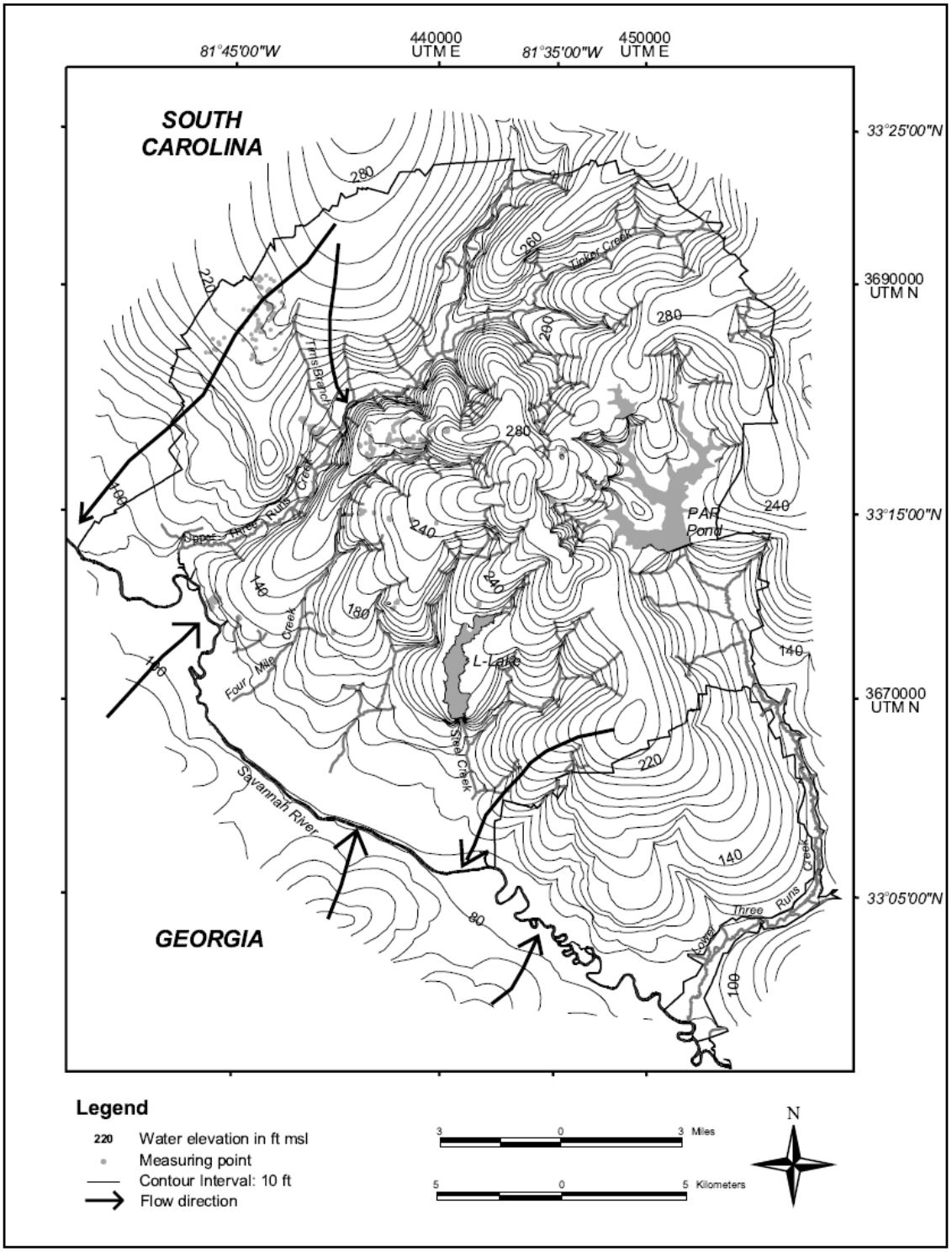


Figure 3-35. Head Elevations for the Water Table Aquifer (UTR aquifer) for the SRS, in ft above msl (from Mamatey 2007)

Porosity and permeability of the UTR aquifer are variable across the site, as is hydraulic conductivity. In the northern and central regions, the aquifer yields only small quantities of water, owing to the presence of interstitial silt and clay and poorly sorted sediments that combine to significantly reduce permeability. Local lenses of relatively clean, permeable sand however, may yield sufficient quantities for domestic use. Such high-permeability zones have been observed in the General Separations Area near the center of the SRS and may locally influence the movement of groundwater (WSRC 2000a). The majority of hydrogeologic data available on the UTR aquifer is from wells in the GSA (WSRC 2000a). These data are presented in the discussion devoted to the GSA below.

3.2.10.1.1.2 Gordon Confining Unit

Clayey sand and clay of the Warley Hill Formation and clayey, micritic limestone of the Blue Bluff Member of the Santee Limestone constitute the Gordon confining unit (Figure 3-13). The Gordon confining unit separates the Gordon aquifer from the overlying UTR aquifer (Figure 3-33). The unit has been informally termed the "green clay" in previous SRS reports (WSRC 2000a).

In the SRS region, the thickness of the Gordon confining unit ranges from about 1.5 to 26 m. The unit thickens to the southeast. From UTR to the vicinity of L Lake and PAR Pond, the confining unit generally consists of one or more thin clay beds, sandy mud beds, and sandy clay beds intercalated with subordinate layers and lenses of quartz sand, gravelly sand, gravelly muddy sand, and calcareous mud. Southward from L Lake and PAR Pond, however, the unit undergoes a stratigraphic facies change to clayey limestone and limey clay typical of the Blue Bluff Member (WSRC 2000a).

Northward from UTR, the Gordon Confining Unit is thin and intermittent, such that the UTR aquifer and Gordon aquifer coalesce to form the Steed Pond aquifer. This occurs in A/M-Area (Figure 3-1) (WSRC 2000a).

3.2.10.1.1.3 Gordon Aquifer Unit

The Gordon aquifer consists of all the saturated strata that occur between the Gordon confining unit and the Crouch Branch confining unit in both the Floridan-Midville Aquifer System and the Meyers Branch confining system (Figure 3-13 and Figure 3-33). The aquifer is semi-confined, with a downward potential from the overlying UTR aquifer observed in interfluvial areas, and an upward potential observed along the tributaries of the Savannah River where the UTR aquifer is incised (WSRC 2000a).

The thickness of the Gordon aquifer ranges from 12 m at well P-4A to 56 m at well C-6 (see Figure 3-31) and generally thickens to the east and southeast. The Gordon aquifer is partially eroded near the Savannah River and UTR. A potentiometric surface map of the Gordon aquifer at the SRS (Figure 3-36) indicates that major deviations in the flow direction, especially in the region affected by UTR, are present where the aquifer is deeply incised by streams that drain water from the aquifers (WSRC 2000a).

The Gordon aquifer is recharged directly by precipitation in the outcrop area and in interstream drainage divides in and near the northern outcrop areas. South of the outcrop area, the Gordon is recharged by leakage from overlying and underlying aquifers. Because streams such as the Savannah River and UTR cut through the aquifers of the Floridan Aquifer System, they represent no-flow boundaries. As such, water availability or flow patterns on one side of the boundary (stream) will not change appreciably due to water on the other side. In the central part of SRS, where the Gordon confining unit is breached by faulting, recharge to the Gordon aquifer is locally increased (WSRC 2000a).

Most of the Gordon aquifer is under confined conditions, except along the fringes of UTR (i.e., near the updip limit of the Gordon Confining Unit) and the Savannah River. The potentiometric surface map of the aquifer (Figure 3-36) shows that the natural discharge areas of the Gordon aquifer at SRS are the swamps and marshes along UTR and the Savannah River. These streams dissect the Floridan Aquifer System, resulting in unconfined conditions in the stream valleys and probably in semi-confined (leaky) conditions near the valley walls. Reduced head near UTR induces upward flow from the Crouch Branch aquifer and develops the "head reversal" that is an important aspect of the SRS hydrogeological system (Figure 3-37).

The northeast-southwest oriented hydraulic gradient in this aquifer (Figure 3-36) across SRS is consistent and averages 0.9 m/km. The northeastward deflection of the contours along the UTR indicates incisement of the sediments that constitute the Gordon aquifer by the creek (WSRC 2000a).

3.2.10.1.2 Floridan-Midville Aquifer System

Northwest of UTR, the permeable beds that correspond to the Floridan and Dublin-Midville Aquifer Systems are often in hydrologic communication owing to the thin and laterally discontinuous character of the intervening clay and silty clay beds, to faulting that breaches the confining beds, and to erosion by the local stream systems that dissect the interval. Here, the Floridan and Dublin-Midville Aquifer Systems coalesce to form the Floridan–Midville Aquifer System (Figure 3-13 and Figure 3-34). The Floridan-Midville Aquifer System is divided into three aquifers: in descending order, the Steed Pond aquifer, the Crouch Branch aquifer, and the McQueen Branch aquifer, separated by the Crouch Branch and McQueen Branch confining units (Figure 3-33). Both the Crouch Branch and the McQueen Branch aquifers extend northwestward from the southern part of SRS. The Steed Pond aquifer is the updip hydrostratigraphic equivalent of the Gordon and UTR aquifer units (Figure 3-13 and Figure 3-33).

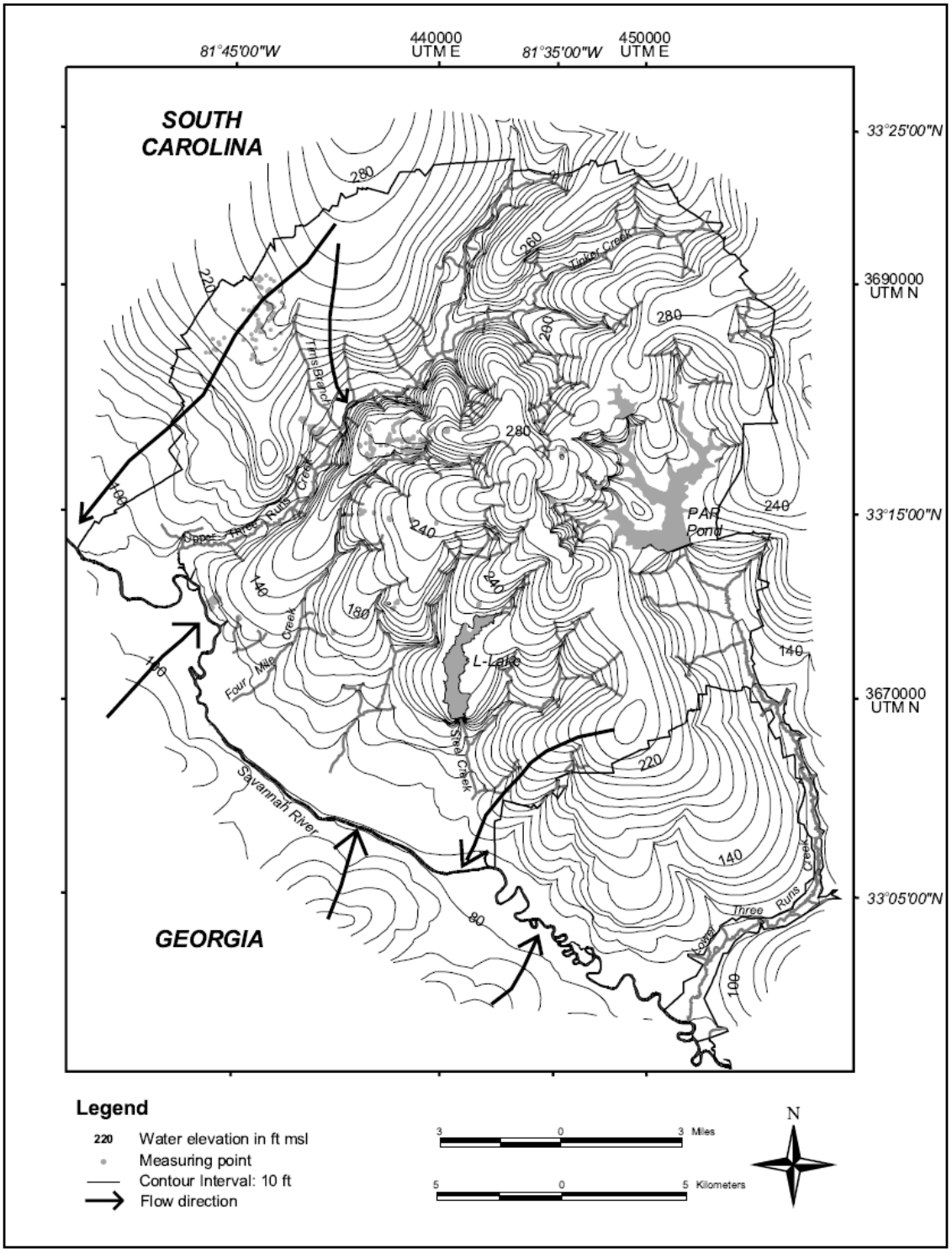


Figure 3-36. Potentiometric Surface of the Gordon Aquifer at the SRS with Elevations in ft above msl (from Mamatey 2007)

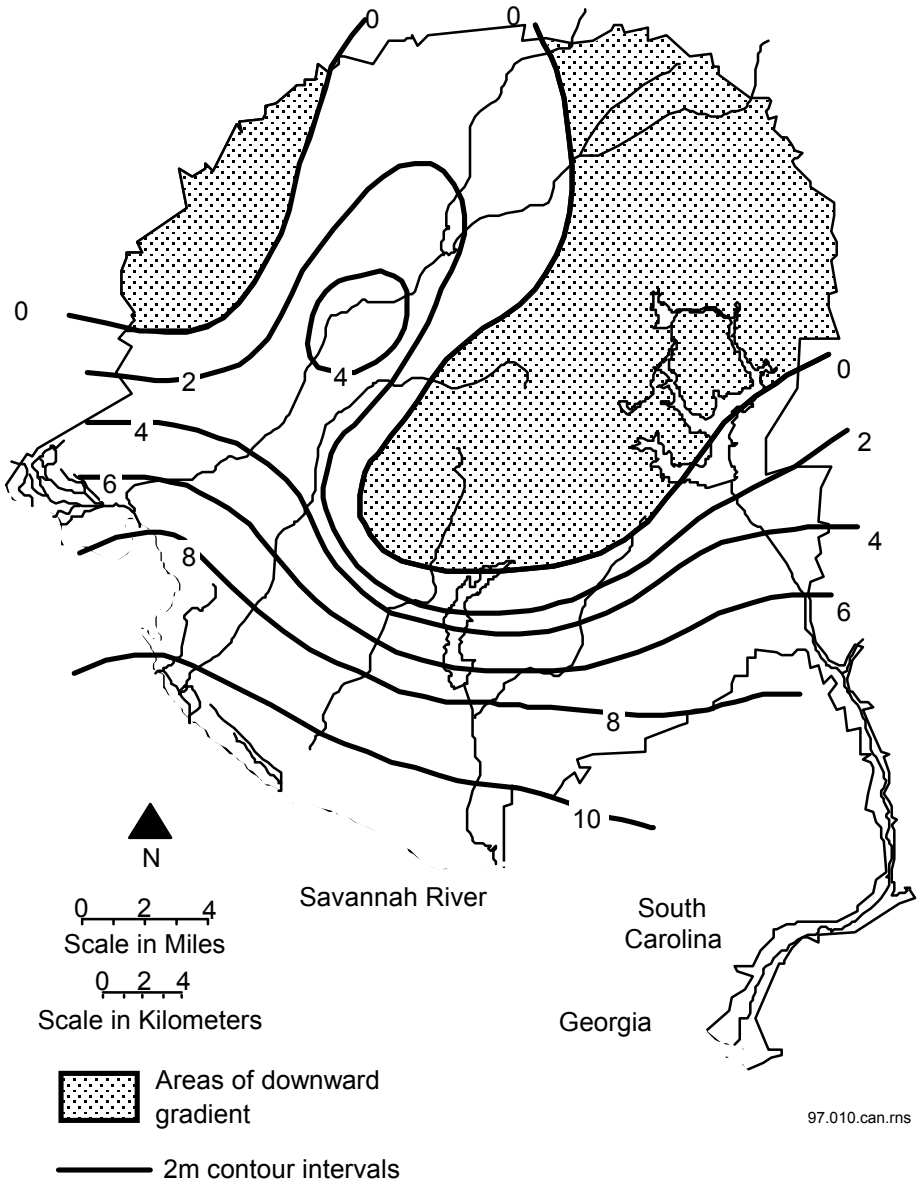


Figure 3-37. Hydraulic Head Difference between the Crouch Branch Aquifer and the Overlying Confined Aquifer (difference in m; positive values indicate areas of upward flow from the Crouch Branch into the overlying aquifers – from Denham 1999)

The Floridan-Midville Aquifer System is defined by the hydrogeologic properties of the sediments characterized in the A/M-Area (i.e., northwest corner of SRS). Near the A/M-Area, the system is 170 m thick and includes all sediments from the water table to the top of the Appleton confining system. The Steed Pond aquifer is 30 m thick in this area; the underlying Crouch Branch confining unit is 25 m thick; the Crouch Branch aquifer is 51 m thick; the McQueen Branch confining unit is 13 m thick; and the McQueen Branch aquifer is 52 m thick (WSRC 2000a).

3.2.10.1.2.1 Steed Pond Aquifer Unit

North of UTR where the Floridan - Midville Aquifer System is defined, the permeable beds that correspond to the Gordon and UTR aquifers of the Floridan Aquifer System are only locally separated, owing to the thin and intermittent character of the intervening clay beds of the Gordon confining unit (Warley Hill Formation) and to erosion by the local stream systems that dissect the interval (Figure 3-13 and Figure 3-33). Here, the aquifers coalesce to form the Steed Pond aquifer of the Floridan-Midville Aquifer System (WSRC 2000a).

The Steed Pond aquifer is defined by hydrogeologic characteristics of sediments located in A/M Area in the northwest corner of SRS. The aquifer is about 30 m thick. Locally, the Steed Pond aquifer can be divided into zones. In A/M Area three zones are delineated, the "Lost Lake" zone, and the overlying "M Area" aquifer zones, separated by clay and clayey sand beds of the "green clay" confining zone (WSRC 2000a).

In A/M Area, water enters the subsurface through precipitation, and recharge into the "M-Area" aquifer zone occurs at the water table by infiltration downward from the land surface. A groundwater divide exists in the A/M Area in which lateral groundwater flow is to the southeast towards Tims Branch and southwest towards UTR and the Savannah River floodplain. Groundwater also migrates downward and leaks through the "green clay" confining zone into the "Lost Lake" aquifer. The "green clay" confining zone that underlies the "M-Area" aquifer zone is correlative with the Gordon confining unit south of UTR (WSRC 2000a).

3.2.10.1.3 <u>Meyers Branch Confining System</u>

The Meyers Branch confining system separates the Floridan Aquifer System from the underlying Dublin and Dublin-Midville Aquifer Systems (Figure 3-13 and Figure 3-33). North of the updip limit of the confining system (Figure 3-34), the Floridan and Dublin-Midville Aquifer Systems are in hydraulic communication and the aquifer systems coalesce to form the Floridan-Midville Aquifer System (Figure 3-34) (WSRC 2000a).

3.2.10.1.3.1 Crouch Branch Confining Unit

In the SRS area, the Meyers Branch confining system consists of a single hydrostratigraphic unit, the Crouch Branch confining unit, which includes several thick and relatively continuous (over several km) clay beds. The Crouch Branch confining unit extends north of the updip limit of the Meyers Branch confining system where the clay thins and is locally absent, and faulting observed in the region locally breaches the unit. Here, the Crouch Branch confining unit separates the Steed Pond aquifer unit from the underlying Crouch Branch aquifer unit (Figure 3-13 and Figure 3-33). The total thickness of the Crouch Branch confining unit where it constitutes the Meyers Branch confining system ranges from about 17 to 56 m. Updip of the confining system, the thickness of the Crouch Branch confining unit ranges from < 1 to 32 m (WSRC 2000a).

3.2.10.1.4 Dublin Aquifer System

The Dublin Aquifer System is present in the southeastern half of SRS and consists of one aquifer, the Crouch Branch aquifer. It is underlain by the Allendale confining system and is overlain by the Meyers Branch confining system (see Figure 3-13 and Figure 3-33). The updip limit of the Dublin Aquifer System beneath the SRS corresponds to the updip limit of the Allendale confining system. North of this line, the Dublin-Midville Aquifer System is defined (Figure 3-34). The thickness of the Dublin Aquifer System generally increases toward the south and ranges from approximately 53 to 88 m. The top of the unit dips 3.79 m/km (20 ft/mi) to the southeast. The Dublin Aquifer System is 65 m thick at well P-22, in the southeast portion of the site (Figure 3-31) (WSRC 2000a).

3.2.10.1.5 Dublin-Midville Aquifer System

The Dublin-Midville Aquifer System underlies the central part of SRS. The system is overlain by the Meyers Branch confining system and underlain by the Appleton confining system (Figure 3-13 and Figure 3-33). The updip limit of the system is established in the region where the overlying Meyers Branch confining system pinches out (Figure 3-34). The downdip limit of the Dublin-Midville is where the Allendale becomes an effective confining system (Figure 3-34). The Dublin-Midville and the updip Floridan-Midville Aquifer Systems were referred to as the Tuscaloosa aquifer by Siple (1967) (WSRC 2000a).

The thickness of the Dublin-Midville Aquifer System ranges from approximately 76 to 168 m. The dip of the upper surface of the system is about 3.8 m/km (20 ft/mi) to the southeast. Near the downdip limit of the system, thicknesses are variable and probably reflect the effects of movement along the Pen Branch Fault (WSRC 2000a).

The Dublin-Midville Aquifer System includes two aquifer units, the McQueen Branch aquifer, and the Crouch Branch aquifer, separated by the McQueen Branch confining unit. The two aquifers can be traced northward, where they continue to be an integral part of the Floridan-Midville Aquifer System and southward where they constitute the aquifer units of the Midville and Dublin Aquifer Systems, respectively (Figure 3-33 and Figure 3-34).

The Dublin-Midville Aquifer System is defined at reference well P-27B, near the GSA, where it is 153 m thick (Figure 3-31). A regional potentiometric surface map prepared by Siple (1967) for this system indicates that the Savannah River is a regional discharge area for the Dublin-Midville Aquifer System (as well as the Floridan-Midville, Dublin and Midville Aquifer Systems), and thus the river represents a no-flow boundary preventing the groundwater in these aquifer systems from flowing southward into Georgia (WSRC 2000a).

3.2.10.1.5.1 Crouch Branch Aquifer Unit

The Crouch Branch aquifer constitutes the Dublin Aquifer System in the southern part of the SRS (Figure 3-13 and Figure 3-33). In the central part of the site, the Crouch Branch aquifer is the uppermost of the two aquifers that constitute the Dublin-Midville Aquifer System. Farther north in the northwestern part of SRS and north of the site, the Crouch Branch aquifer is the middle aquifer of the three aquifers that constitute the Floridan-Midville Aquifer System (WSRC 2000a).

The Crouch Branch aquifer is overlain by the Crouch Branch confining unit and is underlain by the McQueen Branch confining unit. It persists throughout the site, but in northern regions coalesces with the Steed Pond aquifer in areas where the Crouch Branch confining unit ceases to be effective (Figure 3-33) (WSRC 2000a).

The Crouch Branch aquifer ranges in thickness from about 30 to 107 m in the SRS region. Thickness of the unit is variable near the updip limit of the Dublin Aquifer System where sedimentation was affected by movement along the Pen Branch Fault (WSRC 2000a). A relatively high permeability zone exists in a northeast-southwest trending region connecting D-Area, N-Area, and R-Area, paralleling the trace of the Pen Branch Fault (Figure 3-15). A potentiometric surface map for the Crouch Branch Aquifer is presented in Figure 3-38 (Mamatey 2007).

3.2.10.1.6 Allendale Confining System

The Allendale confining system is present in the southeastern third of the SRS and separates the Midville Aquifer System from the overlying Dublin Aquifer System (Figure 3-13 and Figure 3-33). On the site, the Allendale confining system consists of a single unit, the McQueen Branch confining unit. The system dips approximately 6.7 m/km (27 ft/mi) to the southeast and thickens uniformly from about 15 m at the updip limit to about 61 m to the east of the SRS. Near the GSA, the McQueen Branch confining unit is 17 m thick. The rate of thickening is greater in the east than in the west. The updip limit of this confining system is established where pronounced thinning occurs parallel to the Pen Branch Fault (Figure 3-15) (WSRC 2000a).

3.2.10.1.7 Midville Aquifer System

The Midville Aquifer System is present in the southern third of the SRS (Figure 3-34); it overlies the Appleton confining system and is overlain by the Allendale confining system. On the SRS, the Midville Aquifer System consists of one aquifer, the McQueen Branch aquifer unit (WSRC 2000a). The upper surface of the aquifer system dips approximately 4.73 m/km (25 ft/mi) to the southeast across the SRS.

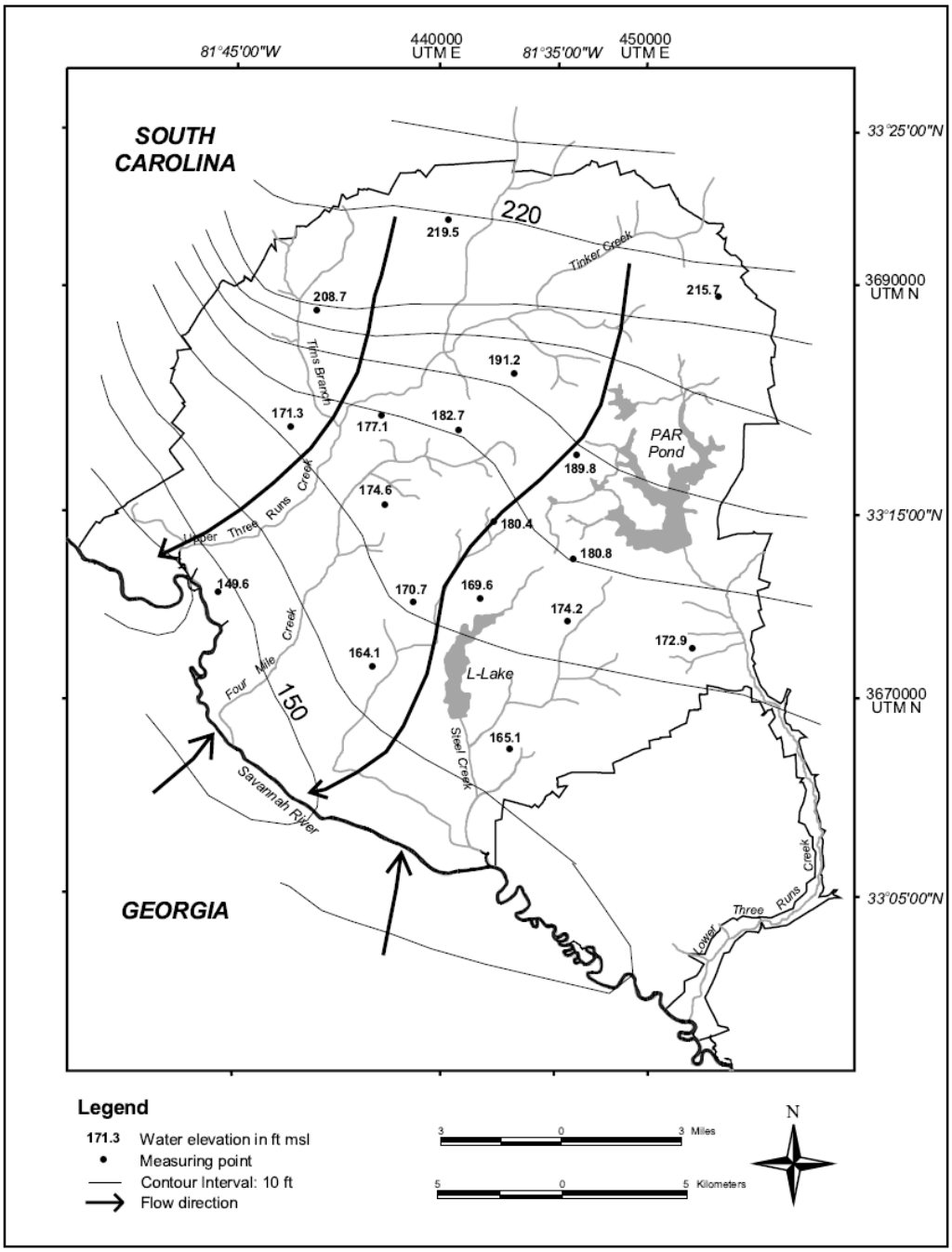


Figure 3-38. Potentiometric Surface of the Crouch Branch Aquifer with Elevations in ft above msl (from Mamatey 2007)

3.2.10.1.7.1 McQueen Branch Aquifer Unit

The McQueen Branch aquifer unit occurs beneath the entire SRS. It thickens from the northwest to the southeast and ranges from 36 m just north of the site (well AIK-858, Figure 3-31) to 103 m south of the SRS. Locally, thicknesses are greater along the trace of the Pen Branch Fault (Figure 3-15) because of the absence and/or thinning of clay beds that compose the overlying McQueen Branch confining unit. A potentiometric map for the McQueen Branch aquifer is presented in Figure 3-39 (Mamatey 2007). The upper surface of the McQueen Branch dips approximately 4.7 m/km (25 ft/mi) to the southeast. The McQueen Branch aquifer unit is defined for the hydrogeologic properties of sediments penetrated by well P-27B near the GSA, where it is 62 m thick (Figure 3-31) (WSRC 2000a).

3.2.10.1.8 Appleton Confining System

The Appleton confining system is the lowermost confining system of the Southeastern Coastal Plain hydrogeologic province and separates the province from the underlying Piedmont hydrogeologic province (Figure 3-13 and Figure 3-33). The confining system is essentially saprolite of the Paleozoic and Mesozoic basement rocks and indurated, silty and sandy clay beds, silty clayey sand and sand beds of the Cretaceous Cape Fear Formation. In the northern part of the SRS region, the Cape Fear Formation pinches out, and the Appleton consists solely of saprolite (WSRC 2000a).

The Appleton confining system dips at about 5.9 m/km (31 ft/mi) to the southeast and thickens from 4.6 m in well C-2 to the north of the SRS (Figure 3-31) to 22 m to the south of the SRS. Sediments of the confining system do not crop out on the SRS (WSRC 2000a).

3.2.10.1.9 The Savannah River as a Boundary to Groundwater Flow from the Savannah River Site

Since 1988, it has been speculated that tritiated groundwater from SRS could flow under the river and find its way into Georgia wells. Considerable effort has been directed at assessing the likelihood of transriver flow. The USGS and the Georgia Department of Natural Resources have drilled 44 wells in support of a special study of transriver flow potential (USGS 2006, Mamatey 2007).

The USGS developed a numerical model to assess the possibility for transriver flow to occur. An extensive effort was undertaken to calibrate the model to water-level measurements obtained from wells on both sides of the Savannah River and screened in each of the hydrostratigraphic units, including the Crouch Branch and McQueen Branch aquifers (Clarke and West 1997, Mamatey 2007). Once the model was calibrated, the USGS employed particle-track analysis to delineate areas of potential transriver flow, which could potentially occur in either an eastward (from Georgia to the SRS) or westward (from the SRS to Georgia) direction (USGS 2006, Mamatey 2007).

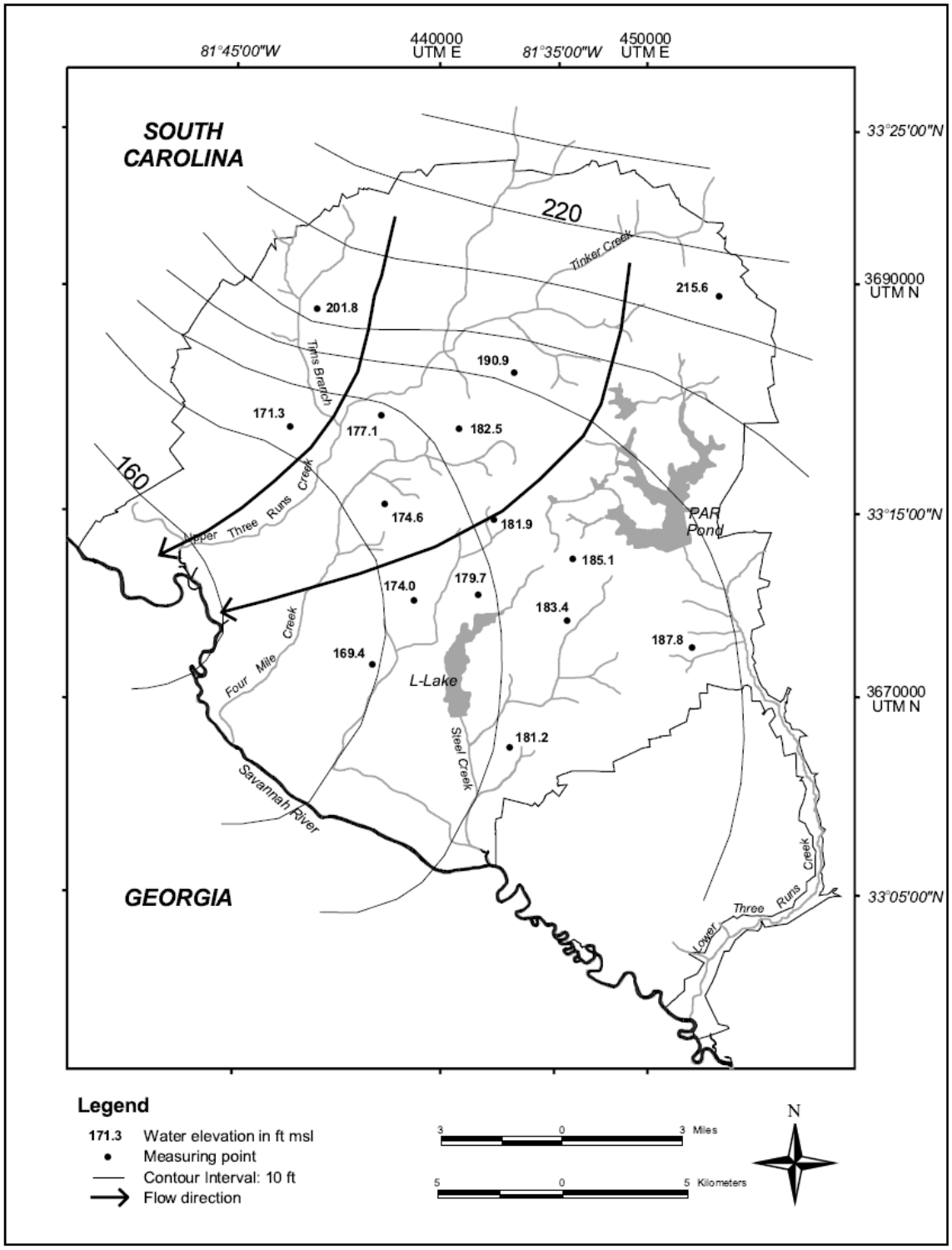


Figure 3-39. Potentiometric Surface of the McQueen Branch Aquifer beneath the SRS with Elevations in ft above msl (from Mamatey 2007)

The model indicated that all locations of transriver flow are restricted to the Savannah River's floodplain, where groundwater passes immediately prior to discharging into the river. The direction depends on the position of the Savannah River as it meanders back and forth within the floodplain. Westward transriver flow primarily occurs in locations south of the SRS, and within the deeper aquifers (Crouch Branch and McQueen Branch). Particle-tracking analysis indicates that such westward flow originates as recharge in upland areas well to the east and south of the SRS (USGS 2006, Mamatey 2007).

The model did identify one area (less than one square mi) of westward transriver flow that has a recharge area located within the SRS. This area occurs immediately adjacent to the Savannah River, where groundwater within the Gordon Aquifer flows immediately prior to discharging into the river. The SRS recharge zones associated with the less- than-one-square-mile area of transriver flow are located in upland areas between D-Area and K-Area, according to the particle tracking results, where there is no known subsurface contamination (USGS 2006, Mamatey 2007). Median groundwater particle transport times between the recharge zones and the region of transriver flow were estimated to range from 366 to 507 years, thus allowing considerable decay of tritium before reaching the area of transriver flow. Consideration of the possibility of enhanced westward flow due to the presence of groundwater extraction wells was given (USGS 2006, Mamatey 2007).

3.2.10.1.10 Summary of Groundwater Discharge Locations for the Savannah River Site

The DOE *End State Vision* report for the SRS discusses the hazards associated with each of the six onsite watersheds as a result of contamination present at the site (DOE 2005). Figure 3-40 is a reproduction of the general sitewide conceptual model for this hazards evaluation, showing the surface water discharge locations for surface runoff and groundwater for many of the SRS areas. In this figure, the term "IOU," or Integrator Operable Unit, refers to the surface water bodies draining all six SRS watersheds. That is, IOUs are the integrators, or collection points, of potential contamination discharged to surface water or groundwater, and thus represent possible paths of contamination.

3.2.10.2 General Separations Area

The GSA, located in the central SRS, includes E-Area, F-Area, H-Area, S-Area and Z-Area Figure 3-30. The area has low-to-moderate topographic relief, and is drained by several unnamed perennial streams. The important streams to which groundwater is discharged in the vicinity of the GSA are UTR to the north, at an average elevation of 150 ft above msl, Fourmile Branch to the south, at an average elevation of 200 ft above msl, and McQueen Branch to the northeast, with elevations ranging from 160 to 250 ft above msl (Smits et al. 1997).

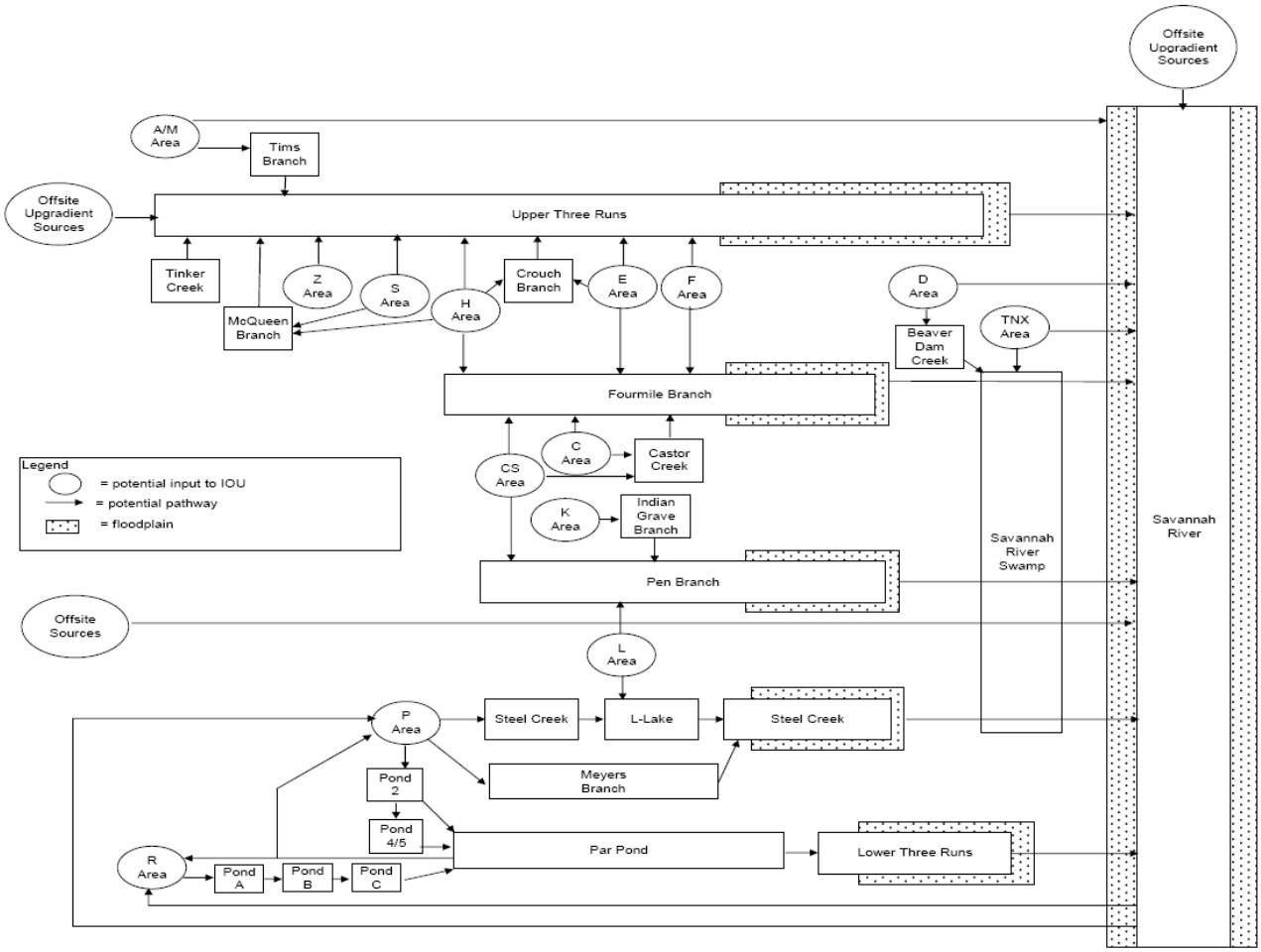
The E-Area LLW Burial Grounds is located on a water table divide. The shallower groundwater from the ORWBG (643-E) flows southwest towards a small tributary of Fourmile Branch. Groundwater from the northeastern parts of the LLRWDF (643-7E) and the MWMF (643-28E) flows toward the north-northwest, toward UTR. Groundwater from the southwestern portions of the BGE (643-7E) and the MWMF flows toward the west-southwest. Groundwater under the northwestern parts of the BGE (643-7E) and the MWMF flows toward the west, and groundwater under the eastern portions of the BGE (643-7E) and the MWMF flows toward the east-southeast (WSRC 2000a).

F-Area is on a near-surface groundwater divide between UTR and an unnamed tributary of Fourmile Branch. The shallower groundwater from the southern part of F-Area discharges to an unnamed tributary of Fourmile Branch, about 600 m to the south. The near-surface groundwater from the northern part of F-Area discharges to one of many tributaries of UTR, about 460 m to the north.

H-Area is located near a water table divide between UTR and Fourmile Branch. The more shallow groundwater from the southern part of H-Area discharges to an unnamed tributary of Fourmile Branch, approximately 300 m south of H-Area. Near-surface groundwater from the northern part of H-Area discharges to one of two tributaries of UTR, which are approximately 460 and 1,200 m north of H-Area, respectively.

The shallower groundwater from S-Area flows toward McQueen Branch, approximately 1,200 m to the northeast. Near-surface groundwater under Z-Area flows generally in a northern direction, with UTR about 1,200 m to the northwest, and McQueen Branch about 150 m to the northeast.

Figure 3-40 summarizes these potential pathways to surface water for contaminants in the GSA. As indicated, contaminants under E-Area are potentially discharged to UTR and Fourmile Branch, as well as to Crouch Branch, according to the *SRS End State Vision* hazards analysis (DOE 2005). F-Area contaminants potentially discharge to UTR and Fourmile Branch. H-Area contaminants are potentially discharged to four streams: UTR, Fourmile Branch, Crouch Branch, and McQueen Branch. S-Area contaminants are potentially discharged to McQueen Branch and UTR, while Z-Area contaminants are potentially discharged to UTR.



Note: CS (i.e., Central Shops) is now known as N-Area

Figure 3-40. General Site-Wide Conceptual Model of Surface Water Discharge Locations for SRS Groundwater (DOE 2005)

3.2.10.2.1 Local Hydrostratigraphy in the General Separations Area

In the GSA, the water table aquifer is the UTR aquifer unit, which is underlain by the Gordon confining unit (Section 3.2.10.1). At well P-27B, in the center of the SRS near the GSA, the water table is approximately 3 m below the ground surface (WSRC 2000a), but ranges between 8 and 18 m at Z-Area (Figure 3-31). The Gordon confining unit is from 0.6 to 9 m thick in the GSA (Aadland et al. 1995). Recent studies indicate the unit is composed of several lenses of green and gray clays and silty sands that thicken, thin, and pinch out abruptly. Extensive carbonate sediments associated with areas of thin or truncated clay beds are present in the GSA.

The UTR aquifer unit is approximately 40 m thick in the center of the SRS (Aadland et al. 1995), near the GSA, and is divided into three zones: two aquifer zones and a tan clay confining zone. In the GSA, the "upper" aquifer zone consists of all saturated strata that lie between the water table and the "tan clay" confining zone. The “lower aquifer" zone of the UTR aquifer occurs between the “tan clay" confining zone and the Gordon confining unit (Aadland et al. 1995, WSRC 2000a). The water table occurs within the “upper” aquifer zone for most of the GSA; but near UTR, where these three zones are incised, the water table is found within the “lower” aquifer zone.

The confining beds of the "tan clay" zone impede the vertical movement of water and often support a local hydraulic head difference, although this is a leaky confining zone. Total thickness of the "tan clay" confining zone, based on measurements at 46 wells distributed throughout the GSA, ranges from 0 to10 m and averages 3.4 m (Aadland et al. 1995, WSRC 2000a).

Recharge to the UTR aquifer occurs at the water table by infiltration downward from the land surface. In the “upper" aquifer zone, part of this groundwater moves laterally toward the bounding streams while part moves vertically downward. The generally low vertical hydraulic conductivities of the “upper" aquifer zone and the intermittent occurrence of the “tan clay" confining zone retard the downward flow of water, producing vertical hydraulic-head gradients in the "upper" aquifer zone and across the "tan clay” confining zone. Downward hydraulic-head differences in the “upper” aquifer zone vary from 1.4 to 1.6 m, and head differences across the “tan clay” are as much as 6.5 m in H-Area. At other locations in the GSA, the head difference across the “tan clay” confining zone is only 0.3 to 1 m, essentially what might be expected due simply to low vertical flow in a clayey sand aquifer. Therefore, the ability of the "tan clay” confining zone to impede water flow varies greatly over this region (WSRC 2000a).

3.2.10.2.2 Characterization Data for the General Separations Area

The GSA has been the focus of numerous geological and hydrogeologic investigations. Monitoring wells were installed as early as the 1950s. Hydrogeologic characterization data for the GSA have been used to: 1) define the various hydrostratigraphic units in the GSA (visual core descriptions, geophysical logs, and cone penetration testing); 2) evaluate hydraulic conductivity (pump tests, slug tests); 3) characterize potentiometric surfaces (water levels and stream flow measurements); 4) characterize vadose zone flow (laboratory measurements); 5) provide data for estimating infiltration at the surface (lysimeter infiltration measurements, watershed water budgets, water content and matrix potential, and weather data); and 6) characterize contaminant sorption potential and monitor contaminant plumes (water quality).

Hydrogeologic characterization data have been collected from the locations denoted in Figure 3-41. Hydrostratigraphic data have been collected from over 100 locations within a 500-meter area surrounding E-Area with over 900 wells providing water level data.

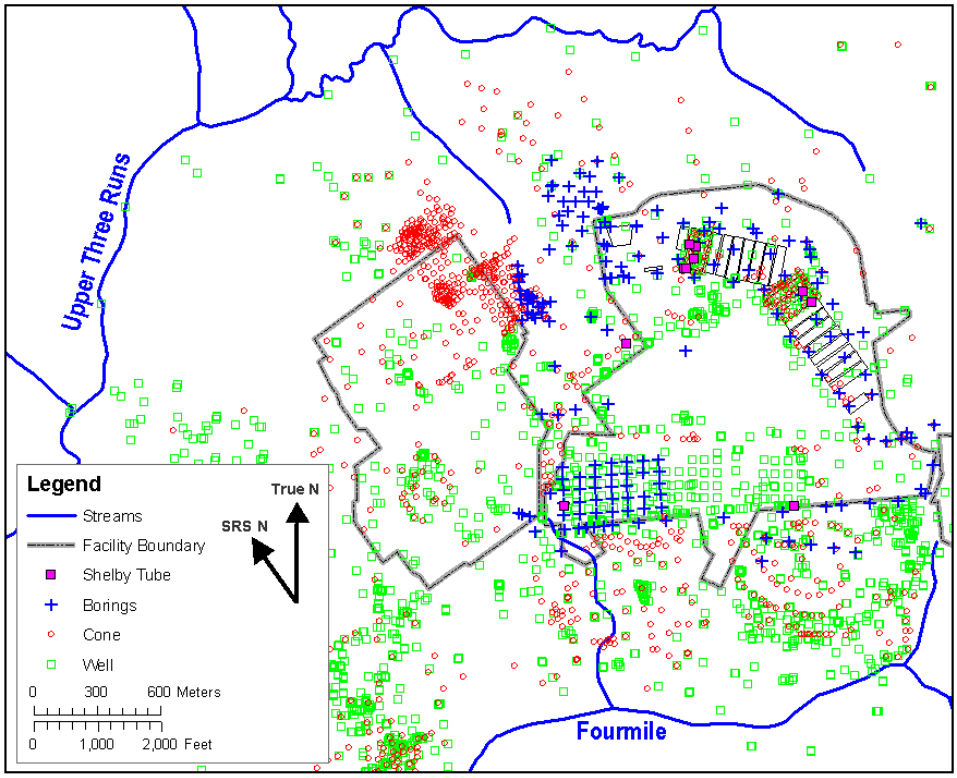


Figure 3-41. Characterization and Monitoring Locations in the GSA Region of the SRS (from the SRS Landmark database)

From characterization data collected, geologic cross-sections have been constructed to provide a two-dimensional vertical view of the subsurface. With numerous such cross-sections, a three-dimensional view begins to emerge. The locations of three such cross-sections constructed for the GSA are identified in Figure 3-42 (Smits et al. 1997). Figure 3-43 shows an example of one two-dimensional view of the subsurface that has been constructed, for the north-south cross-section B-B' (Smits et al. 1997).

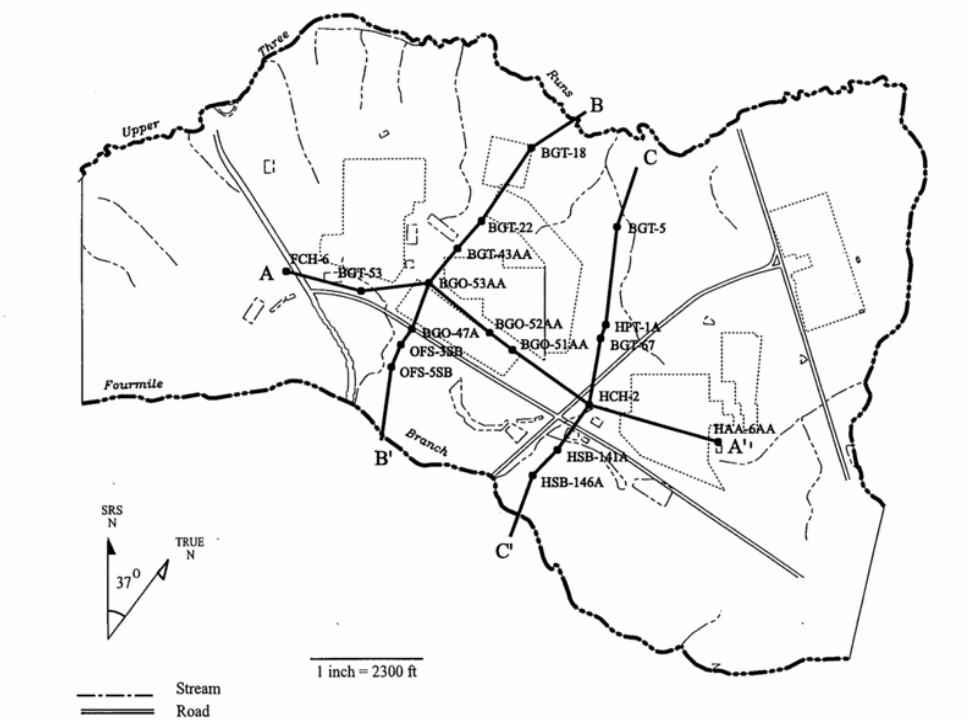


Figure 3-42. Hydrogeologic Cross Sections Constructed for the GSA (from Smits et al. 1997)

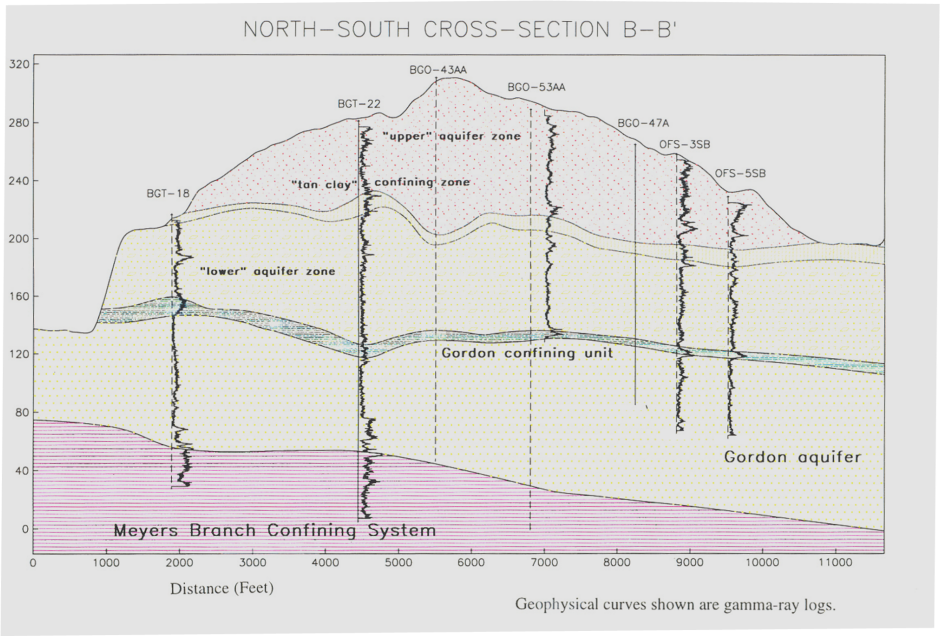


Figure 3-43. Hydrogeologic Cross Section B-B' (refer to Figure 3-42 for location; geophysical curves shown are gamma-ray logs) (from Smits et al. 1997)

3.2.10.2.2.1 Potentiometric Surface Maps for the General Separations Area

Characterization data collected is also interpreted through mapping of hydraulic head contours of water levels measured in wells completed in the aquifers of interest. The potentiometric surface of the UTR (water table) aquifer unit in the GSA, constructed from wells located in the GSA, is illustrated in Figure 3-44. The water table surface depicted in Figure 3-44 is based on the long-term mean elevations measured (Hiergesell 1998). The general configuration shows a groundwater divide area running from west to east with a second divide extending north between McQueen Branch and Crouch Branch to UTR. North of the main divide, the water table slopes steeply toward UTR but is relatively flat in the flood plain of the creek. South of the divide, the water table slopes less steeply toward Fourmile Branch. The highest water table elevations occur just southeast of H-Area, where the 280-foot contour is observed. The lowest elevation is slightly less than 140 feet, at the point where UTR exits the view to the left side of the figure (Hiergesell 1998).

The potentiometric surface of the Gordon aquifer unit in the GSA is illustrated in Figure 3-45. The solid circles indicate locations of measurements in this aquifer unit. McQueen Branch is shown on the right-hand side of the figure. The hydraulic gradient in the Gordon aquifer across the SRS is generally from northeast to southwest, averaging 0.9 m/km, towards the Savannah River. However, the potentiometric surface indicates considerable deflection of the contours due to incisement of aquifer sediments by UTR, such that flow from the GSA is westerly.

Figure 3-45 indicates this incisement of the Gordon aquifer, and shows little or no direct influence on flow in that aquifer by Fourmile Branch and McQueen Branch. This is supported by the hydrogeologic cross section in Figure 3-43. Reduced head near UTR induces upward flow from the underlying Crouch Branch aquifer, such that the GSA lies in the region of head reversal (Figure 3-37). The magnitude of the upward gradient is shown in Figure 3-37, where the head differential contours are plotted. Positive values in the contours indicate upward flow from the Crouch Branch into the overlying aquifer. The magnitude of the upward gradient varies from 2 to 5 m in the GSA, but the low transmissivity of the Meyers Branch confining system results in a low water flux into the Gordon aquifer. Thus, the GSA is hydraulically isolated from the lower aquifers (Crouch Branch and McQueen aquifers) beneath the SRS.

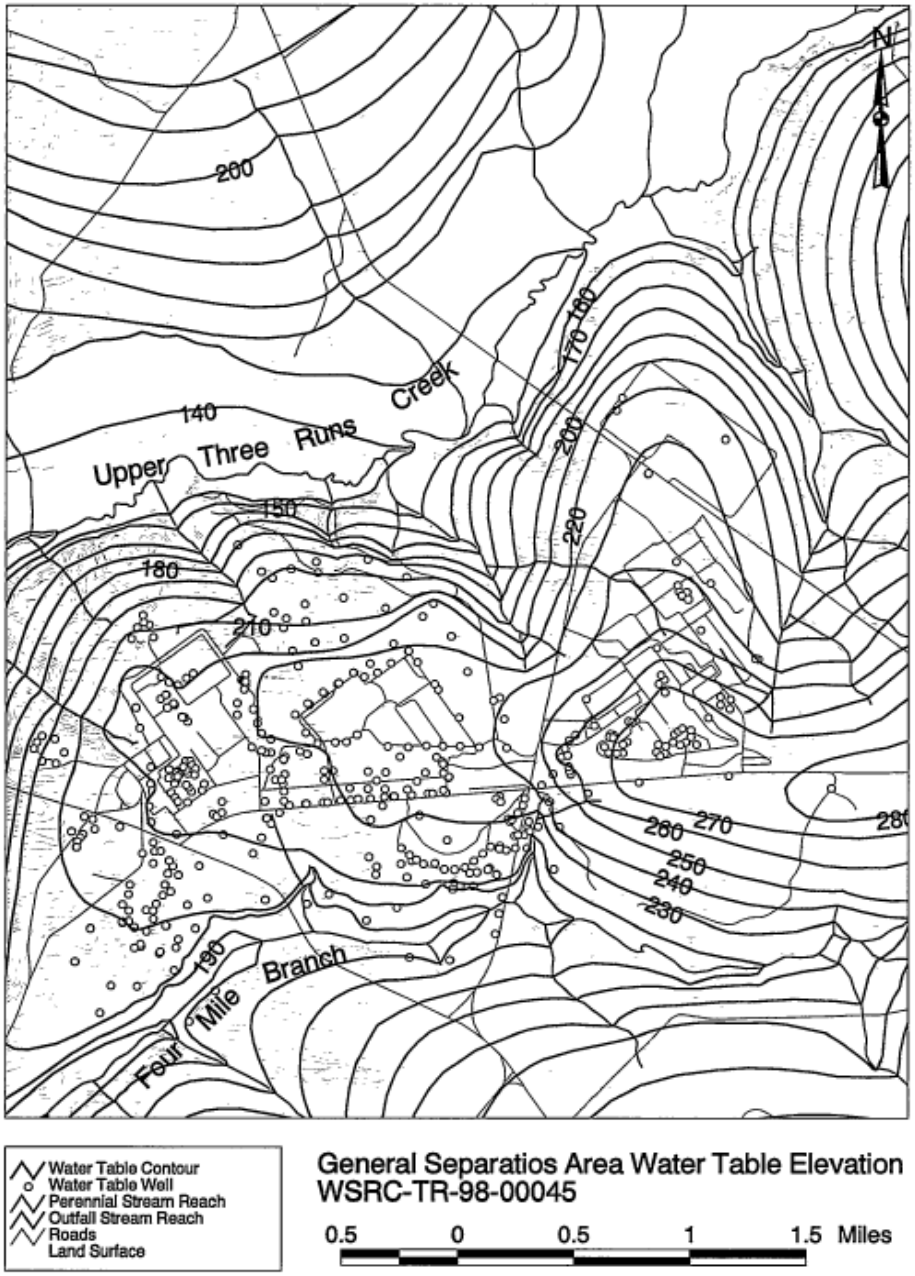


Figure 3-44. Potentiometric Surface of the UTR Aquifer (Water Table Aquifer) for the GSA with Elevations in ft above msl (from Hiergesell 1998)

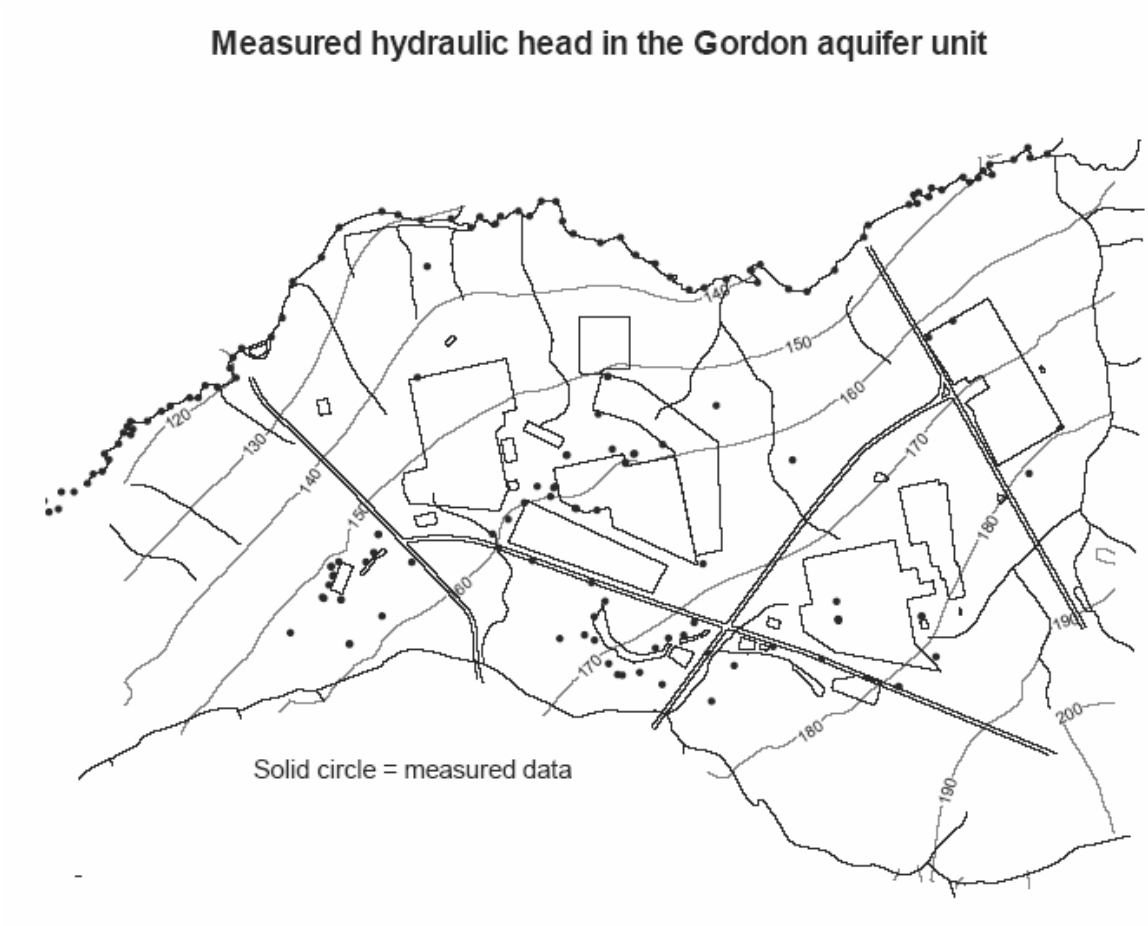


Figure 3-45. Potentiometric Surface of the Gordon Aquifer for the GSA with Elevations in ft above msl

3.2.10.2.2.2 Hydraulic Conductivities in the General Separations Area

According to Aadland et al. (1995) and Denham (1999), the hydraulic conductivity of the "upper" aquifer zone in the GSA is on the order of 4.5 x 10^{-5} m/s based on the more reliable pumping test data (Table 3-18). Hydraulic conductivity of the "lower" zone has been estimated for the GSA by several methods including slug tests, pumping tests, and minipermeameter tests. Average values for the various methods range from 3 x 10^{-6} m/s to 1 x 10^{-4} m/s (Aadland et al. 1995). As reported by Aadland et al. (1995) and Denham (1999), the hydraulic conductivity of the "lower" aquifer zone in the GSA is on the order of 3.5 x 10^{-5} m/s based on long-duration pumping test data. Laboratory analyses of undisturbed samples of the "tan clay" confining zone yielded a range of vertical hydraulic conductivities from 1.2 x 10^{-11} to 4.2 x 10^{-7} m/s (Aadland et al. 1995).

Laboratory and model-derived vertical hydraulic conductivities ranging from 4.0 x 10^{-12} to 9.5 x 10^{-9} m/s have been reported for the Gordon confining unit for the GSA (Aadland et al. 1991; Flach and Harris 1999; Flach 2004). These values suggest that the Gordon confining unit is an effective aquitard in this region.

Horizontal hydraulic conductivity for the Gordon aquifer reported from aquifer tests and modeling studies of the GSA ranges from 8.5 x 10^{-5} to 1.4 x 10^{-4} m/s (Aadland et al. 1995; Denham 1999; Flach and Harris 1999; Flach 2004). According to Aadland et al. (1995) and Denham (1999), a representative horizontal hydraulic conductivity for the Gordon aquifer in the GSA is 1.2×10^{-4} m/s based on long-duration pumping test data.

Table 3-18. Hydraulic Conductivity Values Measured in GSA for Hydrologic Units

Hydrologic Unit	Representative Hydraulic Conductivity Values (m/s)	Literature Source and Notes
"upper" aquifer zone of UTR aquifer unit [a]	4.5x10^{-5} m/s	from long-duration multiple-well pumping test, Aadland et al. (1995) and Denham (1999)
"tan clay" confining zone of UTR aquifer unit [b]	1.2x10^{-11} to 4.2x10^{-7} m/s	based on laboratory analyses of undisturbed samples in GSA, summarized by Aadland et al. (1995)
"lower" aquifer zone of UTR aquifer unit [a]	3.5x10^{-5} m/s	from long-duration multiple-well pumping test, Aadland et al. (1995) and Denham (1999)
Gordon confining unit [b]	4.0x10^{-12} to 9.5x10^{-9} m/s	based on laboratory- and model-derived vertical hydraulic conductivities for the GSA as reported by Aadland et al. (1991), Flach and Harris (1999), and Flach (2004)
Gordon aquifer unit [a]	1.2x10^{-4} m/s	from long-duration pumping tests, Aadland et al (1995) and Denham (1999)

[a] value reflects horizontal hydraulic conductivity for the aquifer
[b] values reflect vertical hydraulic conductivity for the confining zone

3.2.10.2.2.3 Plume Maps for the General Separations Area

The SRS has maintained an environmental restoration effort for many years, managing contaminated groundwater cleanup associated with RCRA HWMFs or Federal Facility Agreement (FFA) units. Figure 3-46 illustrates the major facilities at SRS and the primary contaminants associated with these sites.

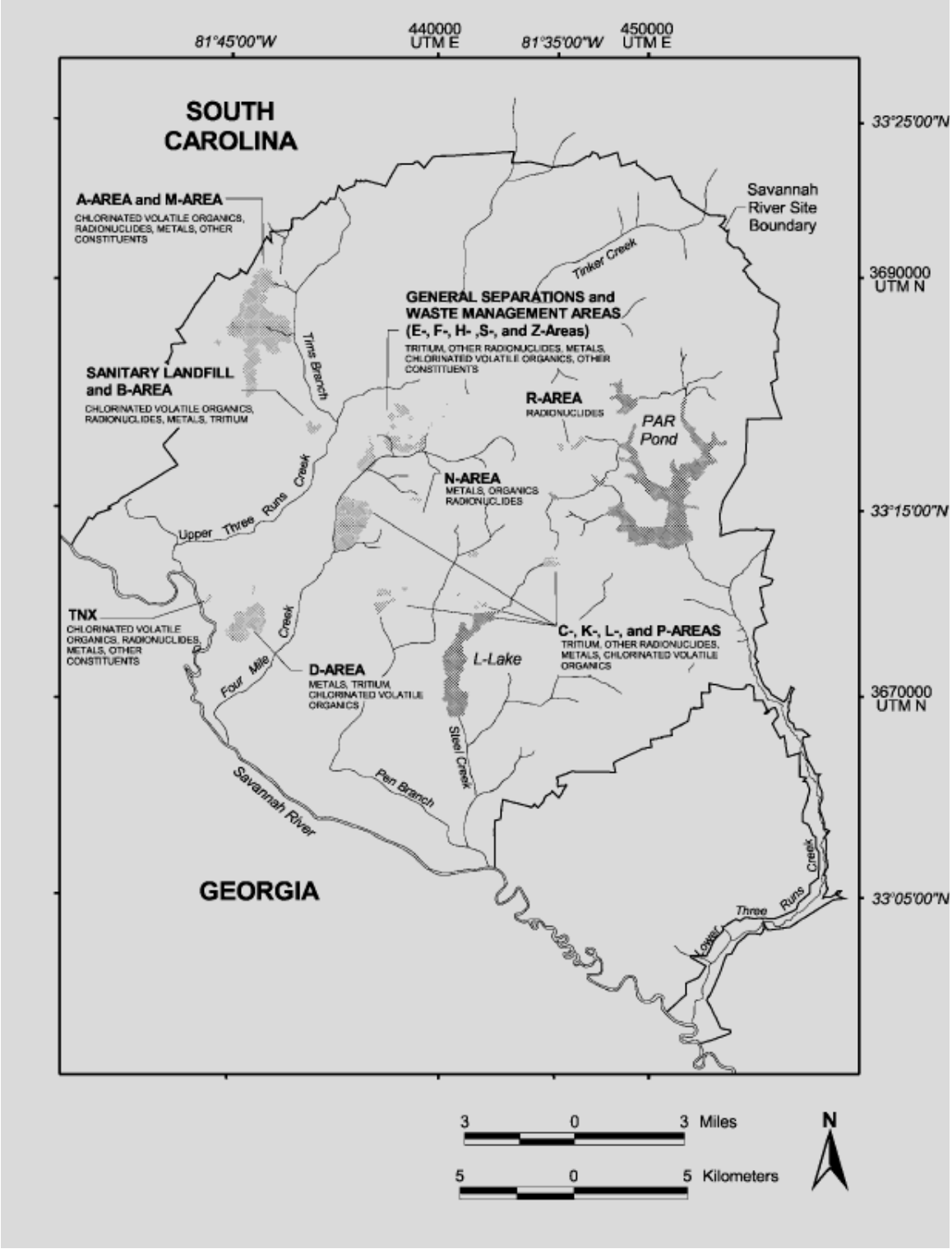


Figure 3-46. Primary Groundwater Contaminants Associated with the Major SRS Operational Areas (from Mamatey 2003)

At the GSA, a number of facilities have contributed to the local contamination of groundwater, including the ORWBG, and the F- and H-Area Seepage Basins. In the course of environmental restoration activities, maps of contaminant plumes are regularly updated. These plume maps are useful in assessing the efficacy of restoration activities, in interpreting other subsurface hydrologic data, and in development of groundwater models that will help predict future contaminant movement.

Below, a sampling of tritium plume maps in the GSA is provided as an example of the types of information derived from monitoring activities. Figure 3-47 depicts tritium plumes in the upper aquifer zone of the UTR aquifer unit under the GSA for the first quarter of 2006 (WSRC 2006c), showing their movement to the north and south of the groundwater divide. Based on the plume concentration contours, it is evident that shallow groundwater beneath the ORWBG flows southerly, toward Fourmile Branch. Shallow groundwater beneath the ELLWF and MWMF, which are located near the groundwater divide, flows northerly, toward UTR. Figure 3-48 shows corresponding tritium plumes in the lower aquifer zone for the first quarter of 2006. The corresponding tritium plumes in the Gordon aquifer are shown in Figure 3-49, from which it can be inferred that most of the tritium in the UTR (water table) aquifer unit discharges to UTR and Fourmile Branch and/or decays before reaching the underlying aquifer. Cross-sectional views have also been constructed, that allow visualization of the depth of the plumes, providing further evidence that contaminants do not travel beneath the Gordon aquifer unit (WSRC 2007c).

3.2.10.2.2.4 Particle-Track Analysis for the General Separations Area

The USGS particle-track analysis, described briefly in Section 3.2.10.1.9, indicates the primary discharge areas for groundwater in the SRS model zone in which the GSA resides (Zone 2 in the study) are UTR to the west and Pen Branch to the south, with discharge also occurring along Fourmile Branch. In the GSA, and east of this area, groundwater flowpaths are dominated by pathlines toward UTR (USGS 2006).

The USGS study also indicates downward migration into the Dublin aquifer system (Crouch Branch Aquifer) east of the GSA in the northern part of Zone 2. This observation is consistent with the head difference map (Figure 3-37) between the Crouch Branch aquifer and overlying Gordon aquifer, showing that a downward gradient re-establishes east of the GSA (USGS 2006).

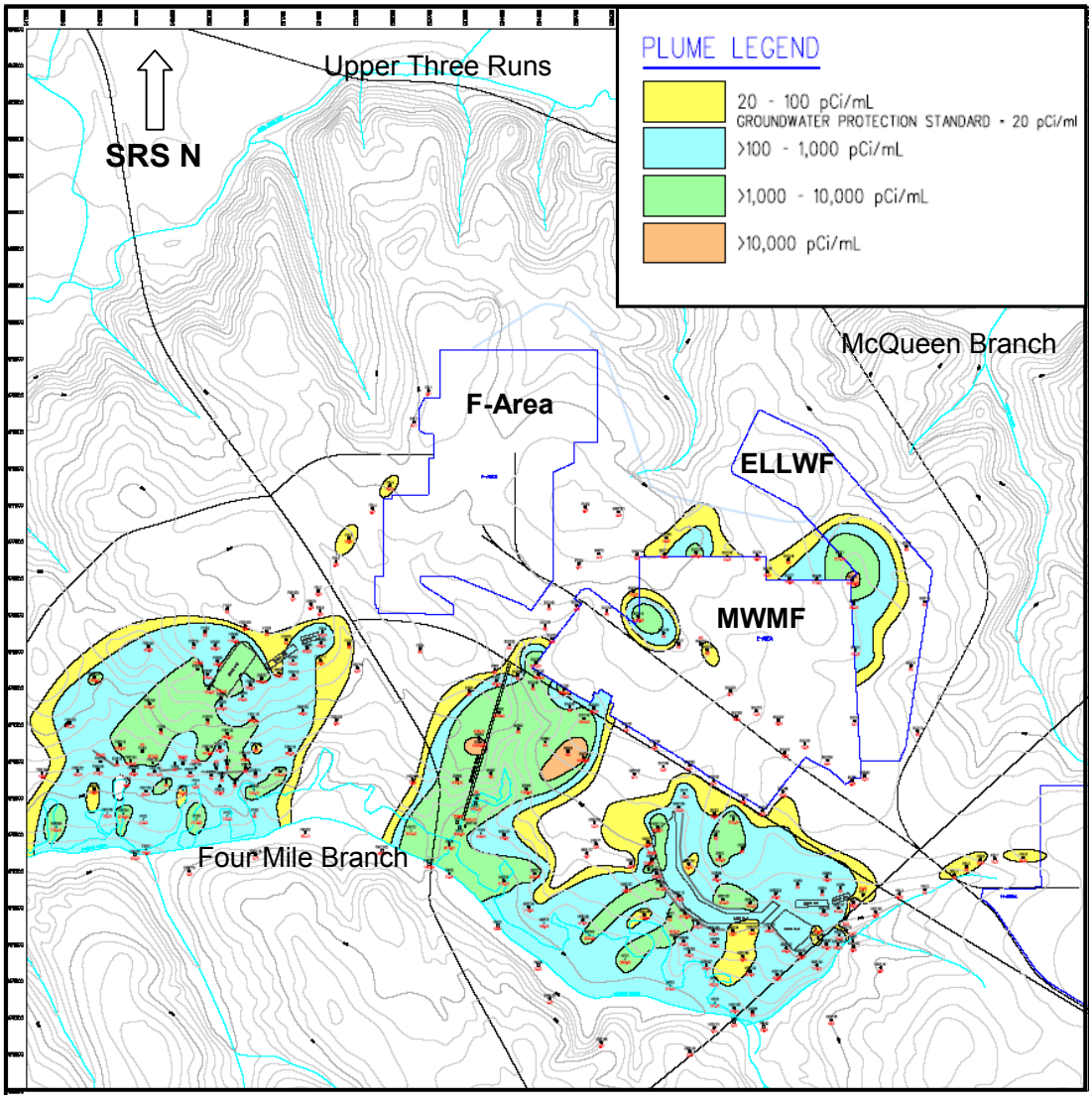


Figure 3-47. GSA Tritium Plumes (first quarter 2006) Moving to North and South Sides of the Groundwater Divide in the Upper Aquifer Zone of the UTR Aquifer (WSRC 2006c)

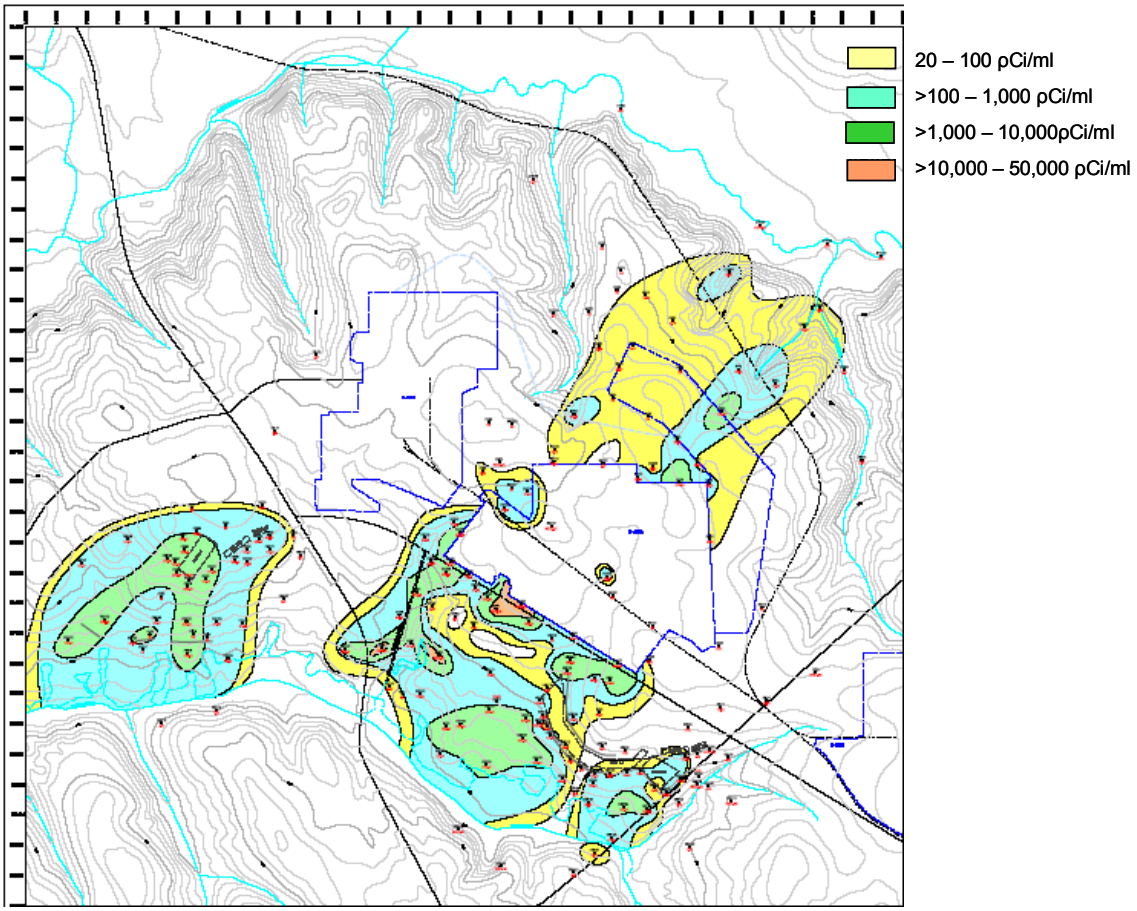


Figure 3-48. GSA Tritium Plumes (first quarter 2006) Moving to North and South Sides of Groundwater Divide in the Lower Aquifer Zone of the UTR Aquifer (WSRC 2006c)

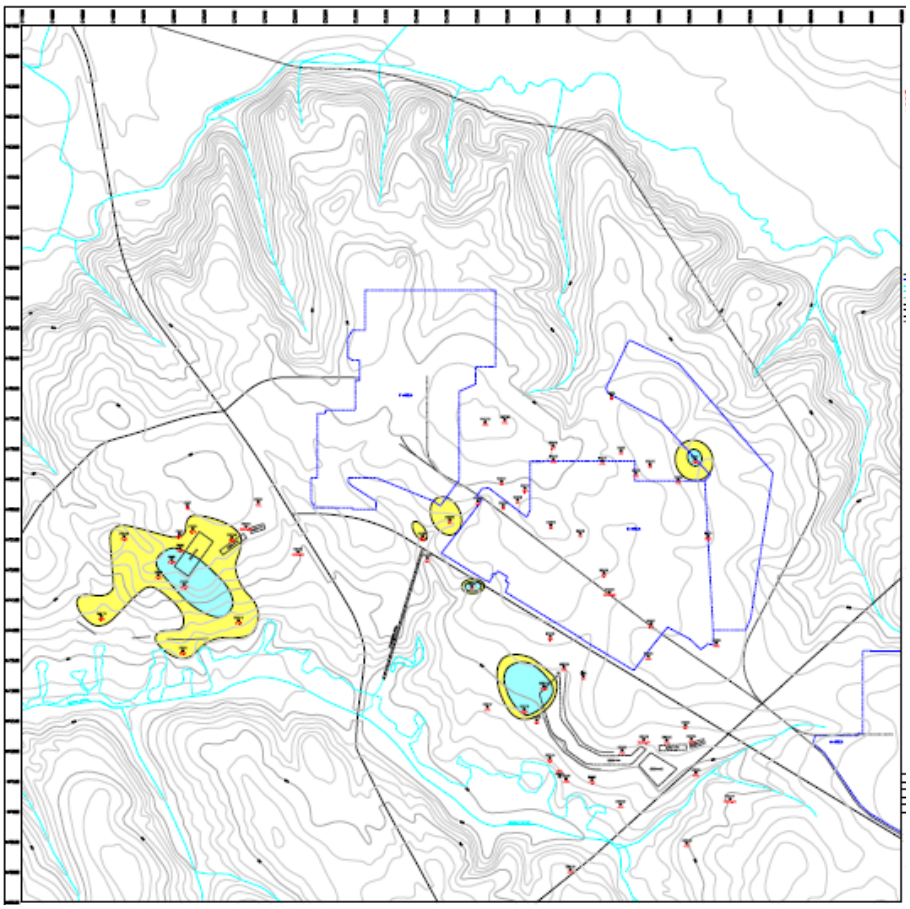


Figure 3-49. GSA Tritium Plumes (first quarter 2006) in the Gordon Aquifer (WSRC 2006c)

3.2.10.3 A/M-Area

A/M-Area, which covers about 10 mi^2 in the northwestern part of the SRS, is situated atop a wide, flat ridge, in the drainage basin of the Savannah River and its tributaries, UTR and Tims Branch (Figure 3-11) (Lewis and Aadland 1994). Elevations range from about 110 to 120 m above msl.

A groundwater divide exists in A/M-Area, such that some of the shallower groundwater flows laterally to the southeast towards Tims Branch, about 0.75 mi from M-Area. However, flow under most of the area is to the south and southwest towards UTR and the Savannah River floodplain (Aadland et al. 1992, Hiergesell 1998, WSRC 2000a). A significant portion of shallower groundwater also migrates vertically downward, leaking through the "green clay" confining zone of the water table aquifer unit into the "Lost Lake" aquifer zone (Jackson et al. 1997, WSRC 2000a).

Figure 3-40 summarizes the potential pathways to surface water for contaminants in A/M-Area. As indicated, contaminants under this area are potentially discharged to Tims Branch, which is a tributary of UTR, or to the Savannah River, according to the *SRS End State Vision* hazards analysis (DOE 2005).

The general groundwater direction of the Crouch Branch aquifer unit, which underlies the water table unit, is to the southwest. In addition to recharge from the overlying water table aquifer, the Crouch Branch aquifer is recharged from an outcrop area approximately 2.5 mi north of A/M-Area. Discharge from the Crouch Branch Aquifer occurs to the southwest along the Savannah River (Aadland et al 1992, Jackson et al. 1997).

3.2.10.3.1 Local Hydrostratigraphy in A/M-Area

North of UTR where the Floridan - Midville Aquifer System is defined, the permeable beds that correspond to the Gordon and UTR aquifers of the Floridan Aquifer System are only locally separated, owing largely to the thin and intermittent character of the intervening clay beds of the Gordon confining unit. Here, the aquifers coalesce to form the Steed Pond aquifer of the Floridan-Midville Aquifer System (WSRC 2000a).

The Steed Pond aquifer is defined by hydrogeologic characteristics of sediments penetrated in well MSB-42 located in A/M Area in the northwest corner of SRS. The aquifer is about 30 m thick. In A/M-Area, the Steed Pond aquifer can be divided into zones. In the southern portion of A/M-Area, three zones are delineated, the "Lost Lake" zone and the overlying "M Area" aquifer zone, separated by clay and clayey sand beds of the "green clay" confining zone (Aadland et al. 1992, Lewis and Aadland 1994, WSRC 2000a). The “green clay” confining zone thins in a northerly direction and becomes absent in the northern portion of A/M-Area. Figure 3-50 illustrates the local hydrostratigraphy of A/M-Area.

The highest water levels in the Steed Pond aquifer unit are on the order of 73 m above msl, in the central A/M-Area (Jackson et al. 1997). The water table in the “M Area” zone in the southern sector of A/M-Area is at a depth of about 37 m below the ground surface (Barnhurst and Noffsinger 2005).

The Crouch Branch aquifer unit, which overlies the McQueen Branch Confining unit, is continuous beneath most of A/M-Area. However, the Crouch Branch confining unit thins northward and eventually pinches out north of the Northern SRS boundary, where the Crouch Branch Aquifer and the overlying Steed Pond aquifer coalesce to form the Hollow Creek Aquifer (Figure 3-50) (Lewis and Aadland 1994). Recent evidence has come to light regarding the lack of competency of the “upper clay” confining zone in the Crouch Branch confining unit in the eastern region towards Tims Branch (Barnhurst et al. 2007).

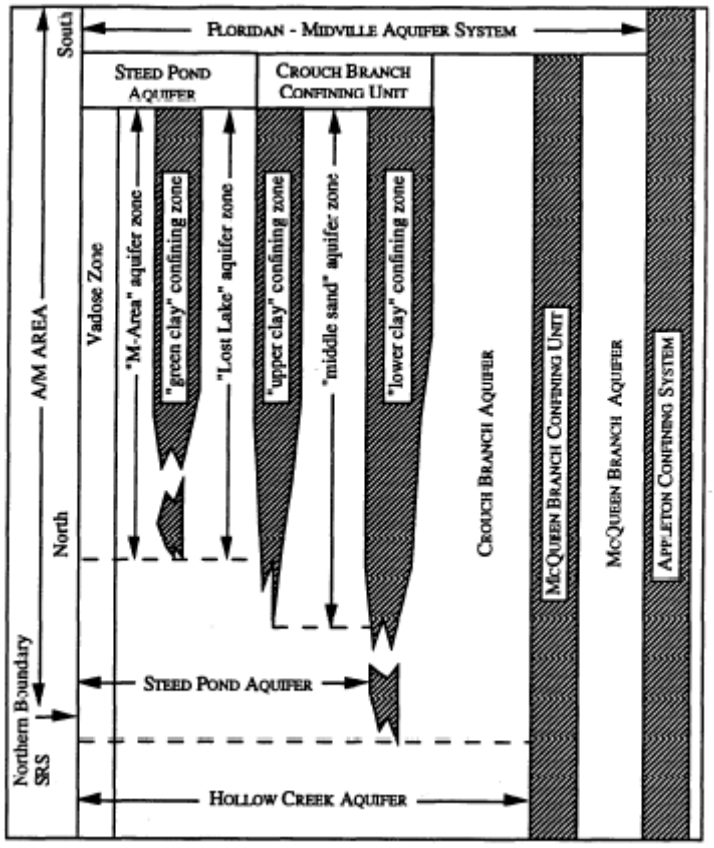


Figure 3-50. Hydrostratigraphic Chart for A/M-Area (from Lewis and Aadland 1994)

3.2.10.3.2 <u>Characterization Data for A/M-Area</u>

Like the GSA, A/M-Area has been the focus of numerous geological and hydrogeologic investigations. A number of monitoring wells have been placed within the area for the purpose of characterizing the subsurface hydrogeologic environment, monitoring existing contamination, and monitoring restoration progress. Figure 3-51 shows the location of monitoring wells in place by 1991 (Fallaw 1991).

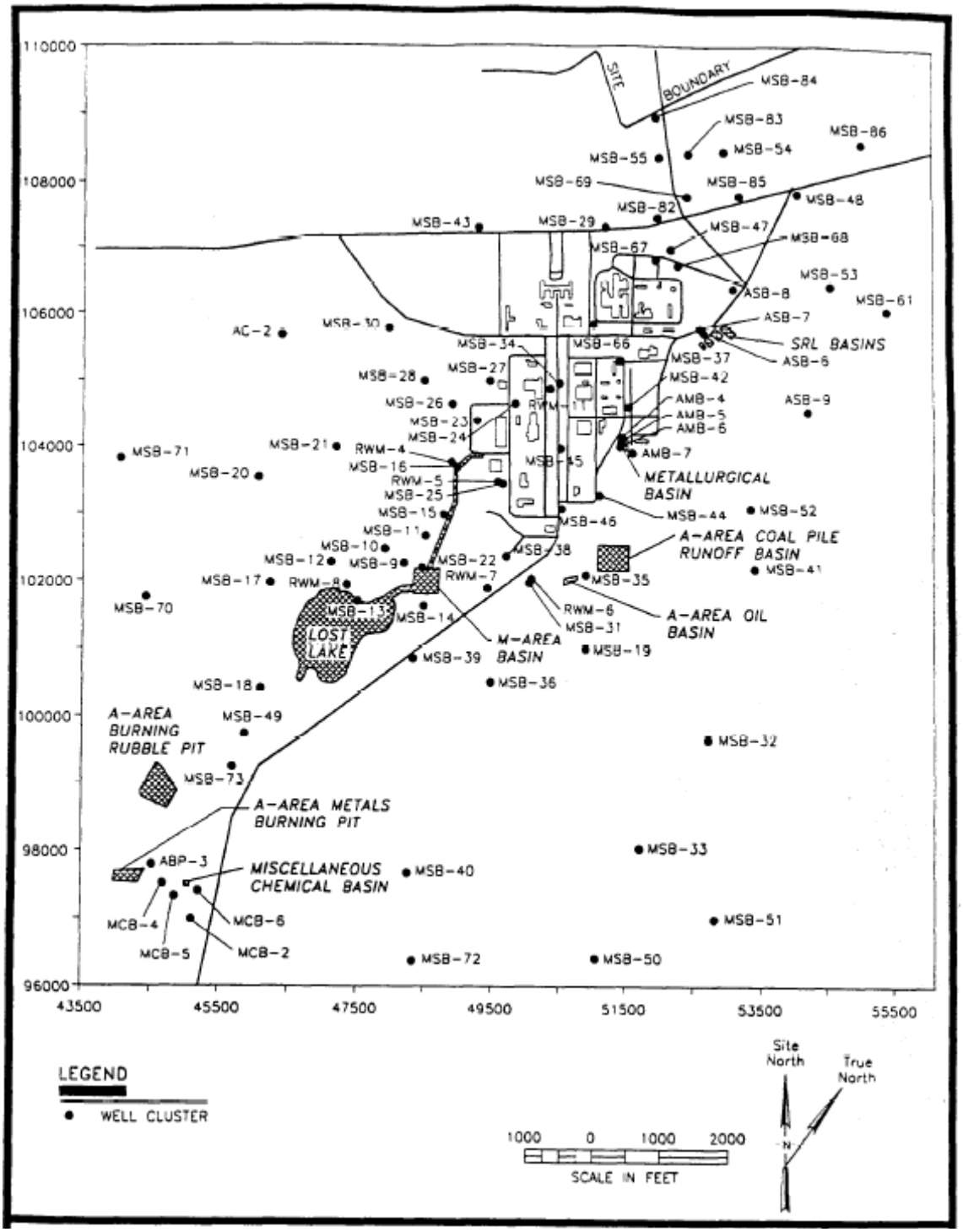


Figure 3-51. Locations of Wells in A/M-Area (from Fallaw 1991)

3.2.10.3.2.1 Potentiometric Surface Maps for A/M-Area

Potentiometric surfaces of the water table unit and the Crouch Branch aquifer unit under A/M-Area are shown in Figure 3-52 and Figure 3-53. Figure 3-52 is a map based on long-term mean elevations measured for the Steed Pond aquifer unit (Hiergesell 1998). The water table contours clearly show the influence of Tims Branch to the southeast of the area, and the groundwater divide.

The potentiometric surface of the Crouch Branch aquifer shown in Figure 3-53 represents the average elevations measured in 1996. Flow in this aquifer is to the southwest, according to this map, and the parallel equipotential lines and their relatively uniform spacing indicate homogeneous conditions in this aquifer unit (Lewis and Aadland 1994).

The potentiometric surface for the McQueen Branch Aquifer in A/M-Area is represented by the regional surface shown in Figure 3-39. Data collected from wells near A/M-Area show a small head difference between wells completed in the Crouch Branch Aquifer and the McQueen Branch aquifer unit (less than 0.3 m in eight of nine well clusters) (Lewis and Aadland 1994).

3.2.10.3.2.2 Hydraulic Conductivities in the A/M-Area

Hydraulic conductivity measurements based on pumping tests for the A/M-Area aquifer units were summarized by Denham (1999), and are provided in Table 3-19.

3.2.10.3.2.3 Plume Maps for A/M-Area

Within A/M-Area are various facilities that are known or suspected sources of groundwater contamination, including the M-Area Settling Basin, sources centered around the SRL (currently called the SRNL), the M-Area Solvent Handling/Storage Area, the Process Sewer Line, the A-001 and A-014 Outfalls, and others (Lewis and Aadland 1994). The primary subsurface contaminants are trichloroethylene (TCE), tetrachloroethylene (PCE) and 1,1,1-trichloroethane (1,1,1-TCA). The first indication of solvent contamination in the Crouch Branch Aquifer occurred in July of 1981 when a water sample from a production well indicated TCE contamination at 10 µg/L. (Jackson et al. 1997).

A number of distinct solvent, mainly TCE, plumes have been mapped for the "M-Area" aquifer zone throughout A/M-Area (Lewis and Aadland 1994). The largest plume in volume and concentration underlies the M-Area solvent Handling/Storage Area, the Process Sewer Line Area, the M-Area Settling Basin, and a portion of the "Lost Lake" (Figure 3-51). In the "Lost Lake" aquifer zone, these plumes cover a substantially larger area of contamination, due to lateral spreading and coalescing of the three major plumes present in the "M-Area" zone. Although not as large in the Crouch Branch aquifer unit, the plume takes on a different shape in this lower unit, reflecting the strong southwestern component of groundwater flow. Figure 3-54 through Figure 3-56 show the TCE plume shapes in the various zones and units for the first half of 2007 (WSRC 2007d).

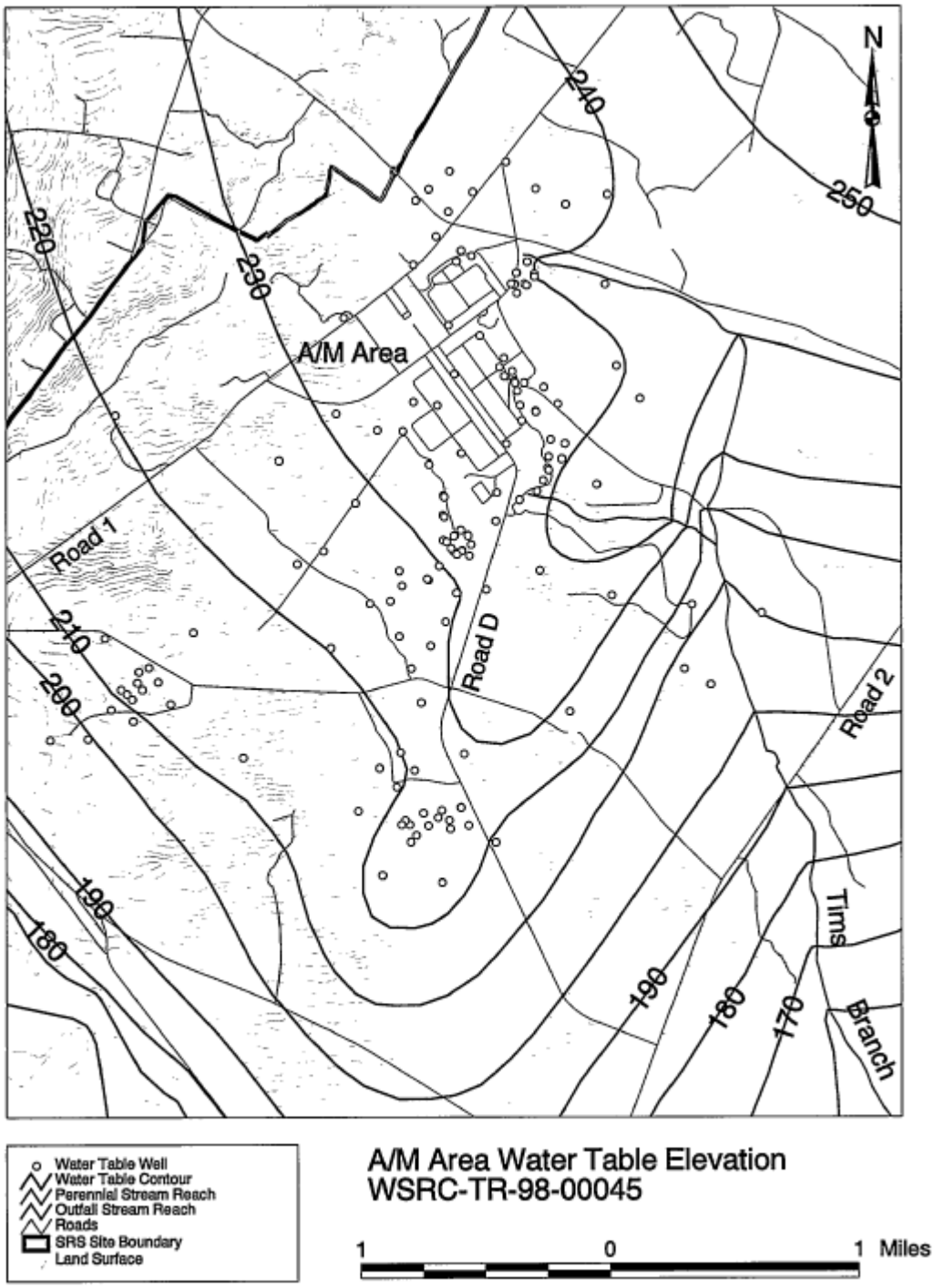


Figure 3-52. Potentiometric Surface of the Steed Pond Aquifer (Water Table Aquifer) for the A/M-Area with Elevations in ft above msl (from Hiergesell 1998)

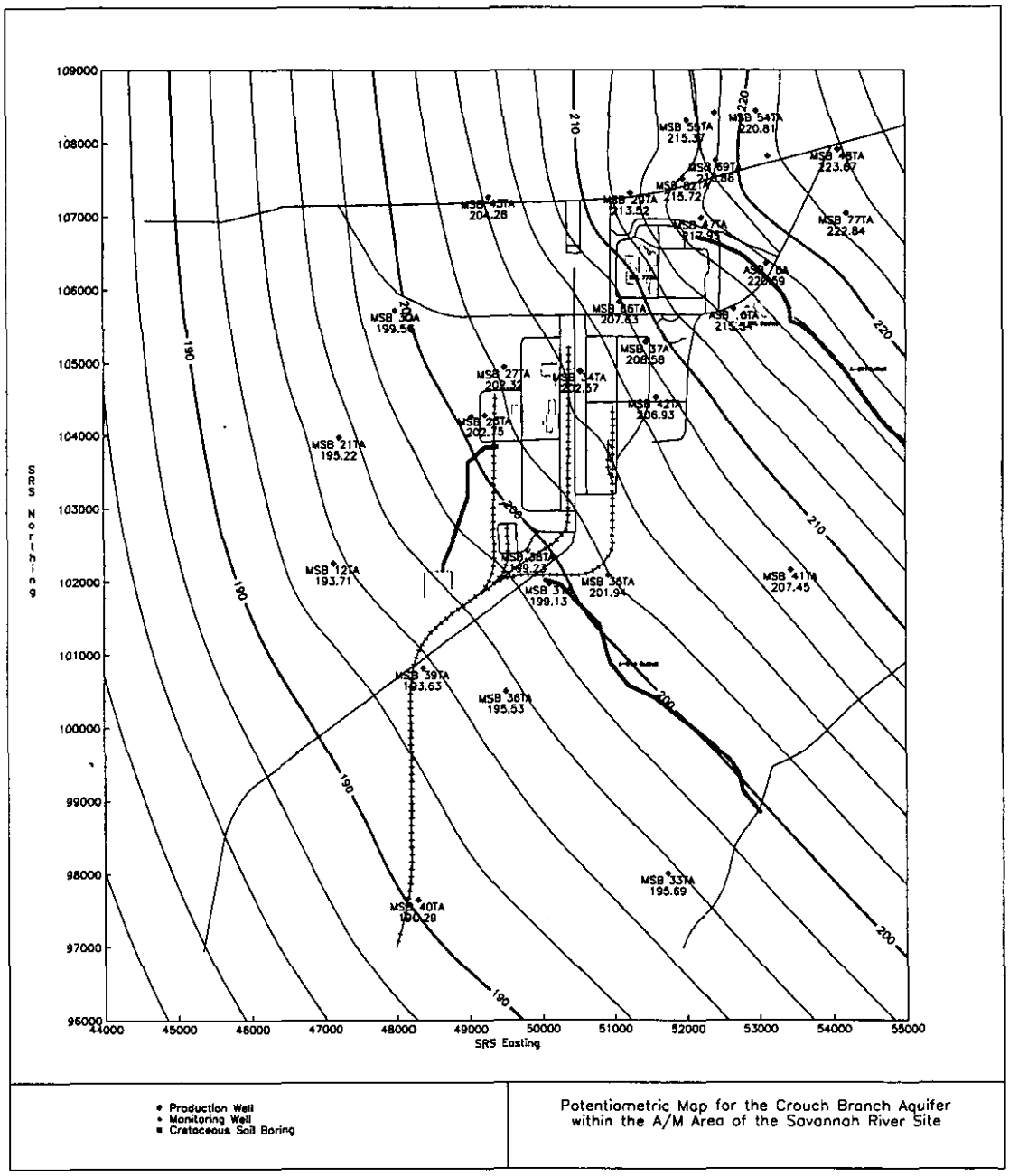


Figure 3-53. Potentiometric Surface of the Crouch Branch Aquifer for the A/M-Area, with elevations in ft above msl (from Jackson et al. 1997)

Table 3-19. Hydraulic Conductivity Values Based on Pumping Tests in A/M-Area for Hydrologic Units (adapted from Denham 1999)

Hydrologic Unit	Mean Hydraulic Conductivity Values	Literature Source and Notes
"M-Area" aquifer zone of Steed Pond aquifer unit [a]	5.7×10^{-5} m/s	Hiergesell and Pemberton (1995)
"Green Clay" confining zone of Steed Pond aquifer unit [b]	$3.5\times10^{-9} – 3.2\times10^{-8}$ m/s	Hiergesell (1993)
"Lost Lake" aquifer zone of Steed Pond aquifer unit [a]	5.0×10^{-5} m/s	Aadland et al (1995), Hiergesell (1993), and Hiergesell and Pemberton (1995)
Upper zone	6.7×10^{-5} m/s	Hiergesell et al. (1994), Hiergesell and Pemberton (1995), and Hiergesell and Novick (1996)
Lower zone	1.2×10^{-4} m/s	Hiergesell (1994), Hiergesell et al. (1994), Hiergesell and Pemberton (1995), and Hiergesell and Novick. (1996)
Crouch Branch aquifer unit [a]	3.0×10^{-5} m/s	Siple (1967)
McQueen Branch aquifer unit [a]	1.2×10^{-4} m/s	Siple (1967)

[a] value reflects horizontal hydraulic conductivity for the aquifer
[b] values reflect vertical hydraulic conductivity for the confining zone

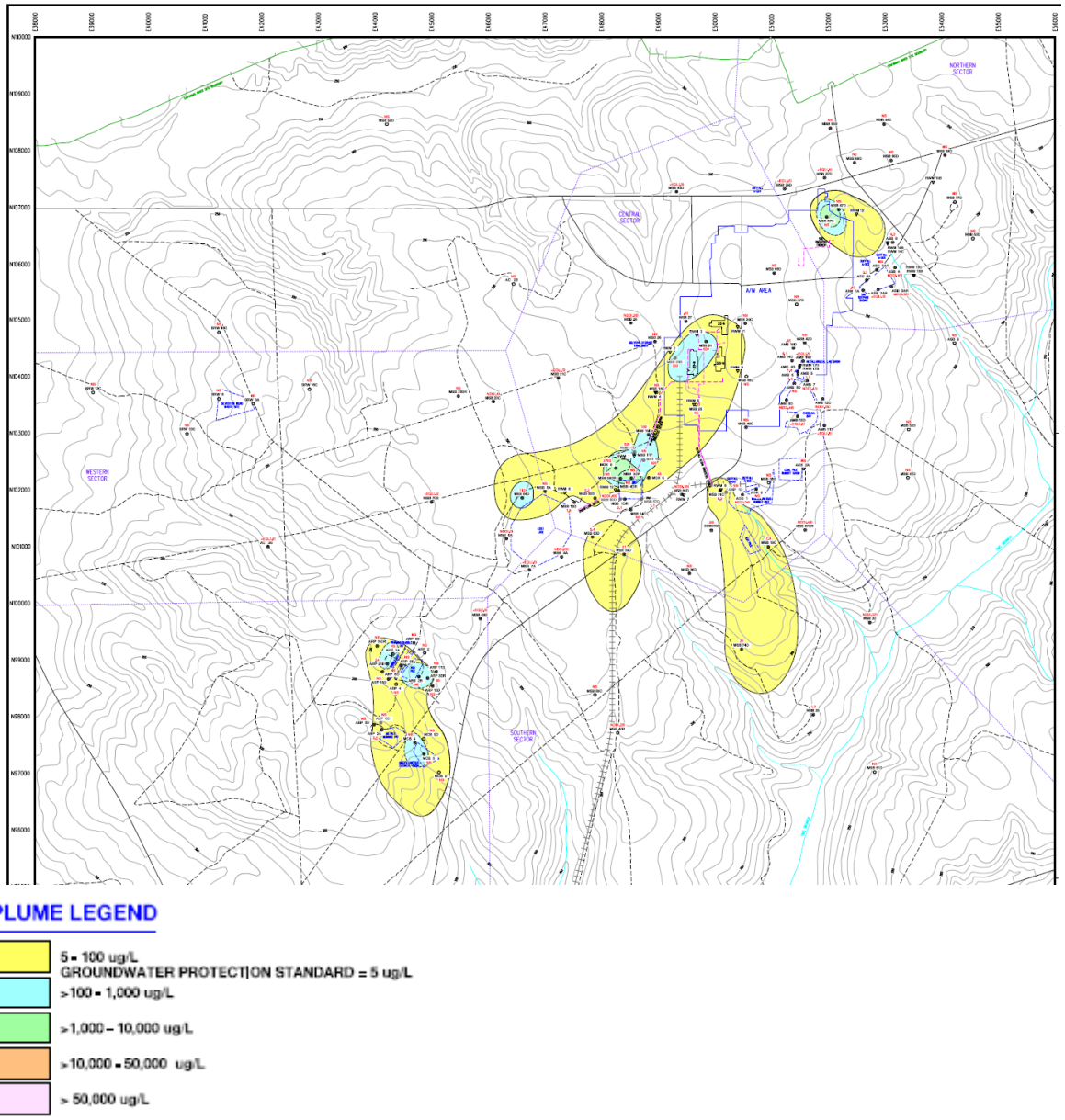


Figure 3-54. TCE Plume Map for the "M-Area" Zone of the Steed Pond (Water Table) Aquifer in the A/M-Area for the First Half of 2007 (adapted from WSRC 2007d)

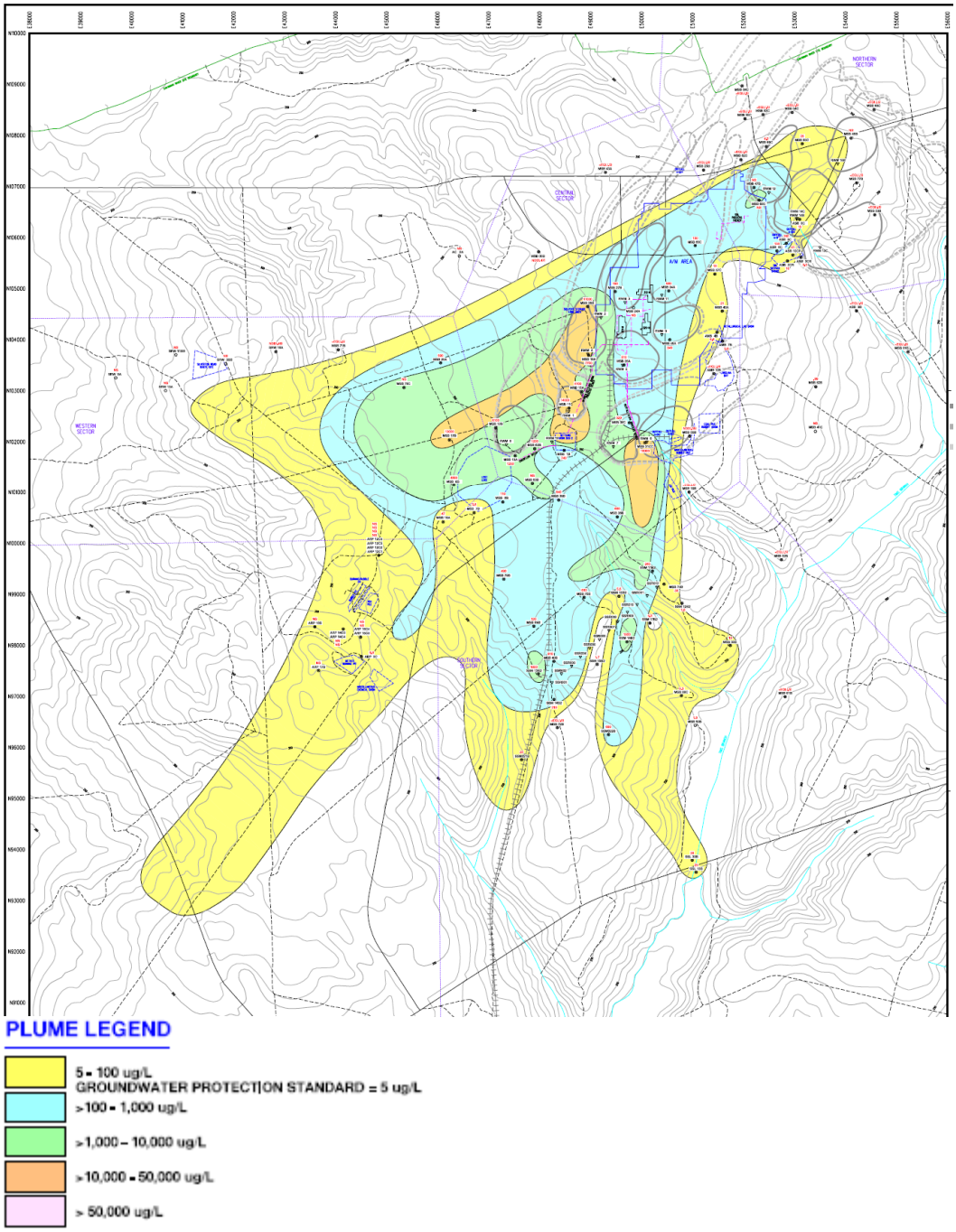


Figure 3-55. TCE Plume Map for the "Lost Lake" Zone of the Steed Pond (Water Table) Aquifer in the A/M-Area for the First Half of 2007 (adapted from WSRC 2007d)

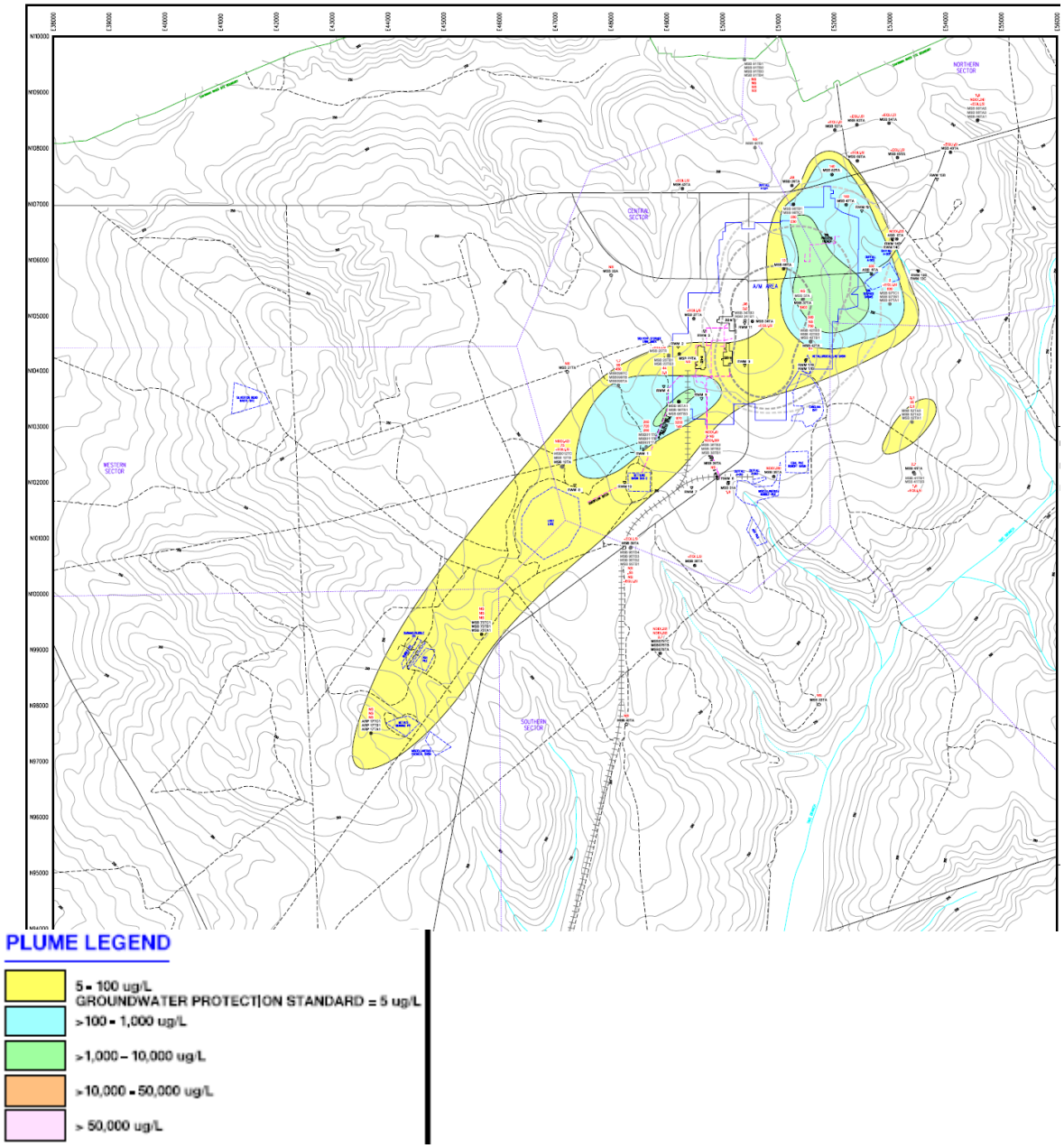


Figure 3-56. TCE Plume Map for the Crouch Branch Aquifer in the A/M-Area for the First Half of 2007 (adapted from WSRC 2007d)

3.2.10.3.2.4 Particle-Track Analysis for A/M Area

The USGS particle-track analysis, described briefly in Section 3.2.10.1.9, indicates the primary discharge areas for groundwater in the SRS model zone in which A/M resides (Zone 1 in the study) are UTR to the south and the alluvial valley of the Savannah River to the southwest (USGS 2006). This study also indicates downward migration into the Dublin aquifer system (of Crouch Branch Aquifer) in the upper half of the Zone 1, which is north of the A/M area. The study also indicates some migration as deep as the upper Midville aquifer system (McQueen Branch aquifer) before moving laterally to UTR. Once in the deeper aquifers, lateral movement is initially southward toward the Savannah River, followed by upward migration to the Gordon aquifer near discharge areas (USGS 2006).

3.2.10.4 D/T-Area

Both D- and T-Areas (also referred to as TNX-Area in various reports) are located within a mile of the Savannah River, on a terrace between UTR to the north, and Fourmile Branch to the south (Figure 3-11). D-Area is relatively flat, with surface elevations ranging from about 36 to 40 m above msl, and is located approximately ½ mi east of the river (WSRC 2001). T-Area, located about ¼ mi east of the river, is at an elevation of 46 m above msl, and also is relatively flat. T-Area slopes toward the east away from the Savannah River. A portion of the Savannah River floodplain lies immediately west of T-Area, at 29 ft above msl. A small levee divides the floodplain, serving as the bank of the river during high stages (Nichols 1993).

In general, groundwater flow beneath D-Area is primarily to the west towards the Savannah River. However, flow in the vicinity of the basins has been altered locally due to recharge from the basins. Groundwater mounds beneath these basins, resulting in radial flow away from the basins in the water table aquifer. Some of this water flows toward the Savannah River and wetlands to the west, but some flows towards the unnamed tributary of Beaver Dam Creek to the east, and a portion to the wetlands area northwest of the ash basins (WSRC 2001).

The Savannah River is also the major discharge point for the groundwater beneath T-Area, largely controlling the direction and rate of lateral and vertical groundwater movement. A ditch constructed in the 1960s to carry discharges from an NPDES outfall to the Savannah River also influences groundwater flow patterns by providing an additional discharge location, as well as a natural seep that exists in the floodplain, which diverts very shallow groundwater to the Savannah River. The New TNX Seepage Basin, located east of T-Area and constructed in 1980, recharges the water table aquifer, causing groundwater flow in the southern part of T-Area to change from a west-southwest direction to a westward direction (Nichols 1993).

Figure 3-40 summarizes the potential pathways to surface water for contaminants in D/T-Area. As indicated, contaminants under D-Area are potentially discharged to Beaver Dam Creek, which flows into the Savannah River Swamp, or to the Savannah River. Contaminants under T-Area (TNX-Area in the figure) are potentially discharged to the Savannah River Swamp, or to the Savannah River, according to the *SRS End State Vision* hazards analysis (DOE 2005).

3.2.10.4.1 Local Hydrostratigraphy in the D/T-Area

In D-Area, the water table ranges from 1.5 to 4.6 m below the ground surface (WSRC 2001). The water table aquifer is about 7.6 m below ground surface throughout much of T-Area (WSRC 1999b).

The water table lies in the UTR aquifer unit at both D- and T-Areas, although the three zones identified in other areas of the SRS (i.e., "upper," "tan clay," and "lower") are not distinguished. In D-Area, the UTR aquifer ranges in thickness from 26 to 29 m (WSRC 2001). In T-Area, it is reported to vary from 11 to 12 m in thickness (Nichols 1993).

The Gordon confining unit is continuous through much of D-Area, but pinches out to the west due to erosion. In T-Area, this unit is reported to be between 1.5- and 3-m thick. Localized zones of weak upward gradient across this unit (from the Gordon aquifer unit below, to the overlying UTR aquifer unit) are present near the Savannah River floodplain (WSRC 2001). A similar upward gradient is observed at T-Area (Nichols 1993).

The Gordon aquifer unit ranges from 11 to 14 m thick at D-Area, and 11 to 12 m thick in T-Area (Nichols 1993, WSRC 2001). Flow in this unit is generally towards the Savannah River in both D- and T-Area. The upward gradient from the underlying Crouch Branch aquifer unit in both areas is quite significant (Figure 3-37), thus hydraulically isolating the lower Dublin-Midville Aquifer System from the UTR and Gordon aquifer units. In D-Area, the average horizontal hydraulic gradient for the Gordon aquifer, towards the Savannah River, is about 0.007 (WSRC 2003a).

3.2.10.4.2 Characterization Data for D/T-Area

In 1984 and 1987, monitoring wells were installed in D-Area, in keeping with the SRS groundwater monitoring program. As part of a 1994 well installation, geophysical data were collected for delineation of subsurface features. Additional lithologic information was obtained in 1997 using CRP technology (WSRC 2001).

The first series of groundwater monitoring wells was installed in T-Area in the early 1980s. Sampling data from these wells indicated seepage from the unlined basins which received wastewater during operations, leakage from the process sewers which carried the wastewater to the basins, as well as leachate from other waste disposal activities in the area. In 1992, the groundwater monitoring system at T-Area consisted of 51 wells (Nichols 1993).

3.2.10.4.2.1 Potentiometric Surface Map for D/T-Area

The potentiometric surface of the water table unit under D/T-Area is shown in Figure 3-57. Figure 3-57 is a map based on long-term mean elevations measured for the UTR aquifer unit (Hiergesell 1998). The water table contours are fairly parallel in both D- and T-Areas, and show the overwhelming influence of the Savannah River discharge point to the southwest.

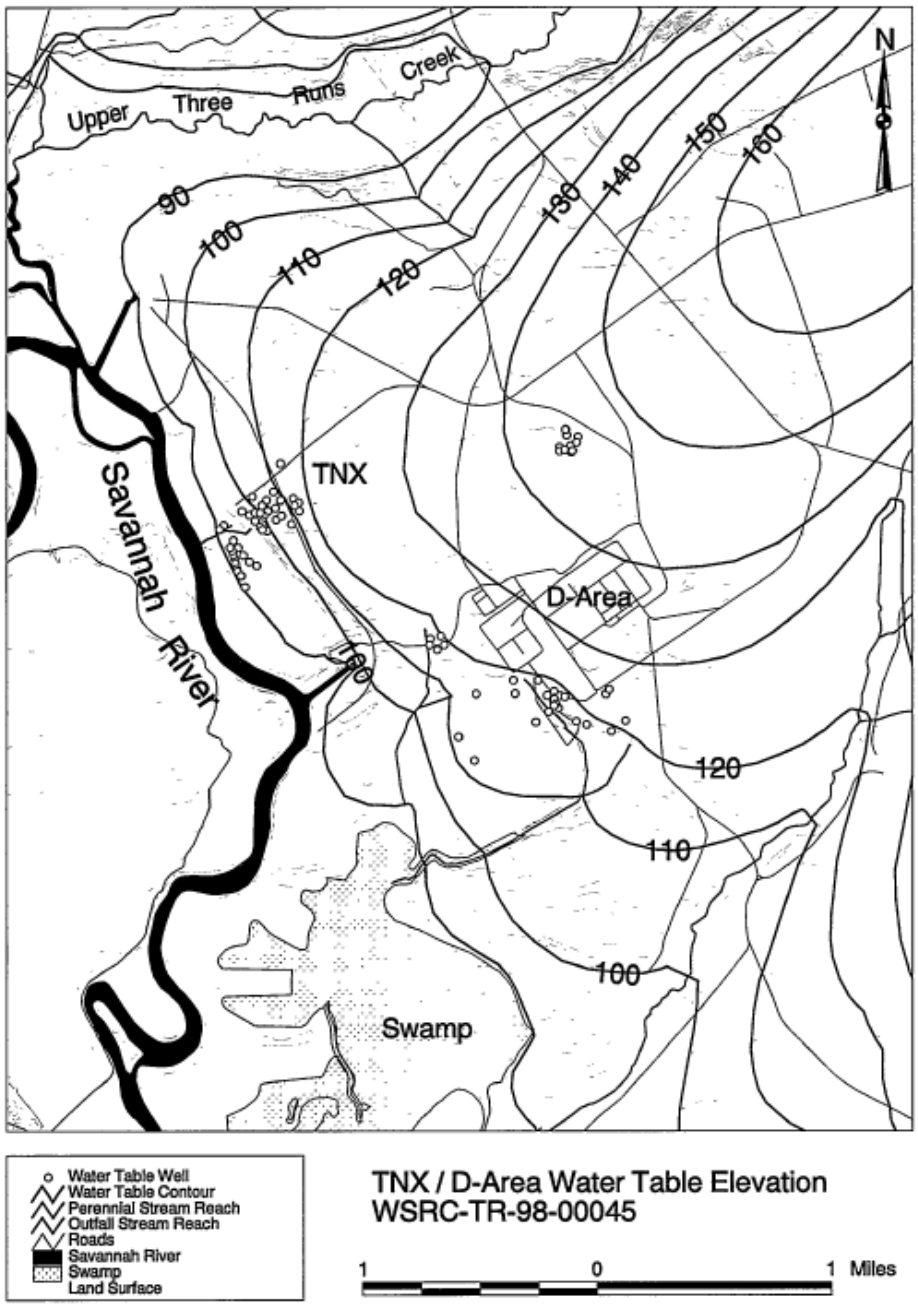


Figure 3-57. Potentiometric Surface of the UTR Aquifer Unit (Water Table Aquifer) for D/T-Area with Elevations in ft above msl (from Hiergesell 1998)

3.2.10.4.2.2 Hydraulic Conductivities in D/T-Area

Hydraulic conductivity measurements based on pumping tests for the D/T-Area aquifer units were summarized by Denham (1999), and are provided in Table 3-20.

Table 3-20. Hydraulic Conductivity Values Based on Pumping Tests in T-Area for Hydrologic Units (adapted from Nichols 1993)

Hydrologic Unit	Mean Hydraulic Conductivity Values [a]	Literature Source and Notes
UTR aquifer unit [b]	7.4×10^{-5} m/s	based on multi-well aquifer performance test (Nichols 1993)
UTR aquifer unit [b]	4.9×10^{-5} m/s	based on multi-well aquifer performance test (Nichols 1993)
Gordon confining unit [c]	3.9×10^{-10} m/s	based on multi-well aquifer performance test (Nichols 1993)
Gordon aquifer unit [b]	4.2×10^{-5} m/s	based on multi-well aquifer performance test (Nichols 1993)

[a] Nichols 1993 did not specify units; the assumption is made they were presented in ft/d, which provides values in general agreement with other pump test results on the SRS (see Table 3-18 and Table 3-19)

[b] values reflect horizontal hydraulic conductivity for the aquifers

[c] value reflects vertical hydraulic conductivity for the confining zone

3.2.10.4.2.3 Plume Maps for D/T-Area

As part of RCRA investigations in D-Area, plumes of groundwater contaminants in the area have been characterized. Figure 3-58 and Figure 3-59 show water table plume maps of TCE and tritium, respectively. As is apparent from these figures, the plumes arise from a number of sources in D-Area. The maximum TCE concentrations in Figure 3-58 are in excess of 200 µg/L. The maximum tritium concentrations in Figure 3-59 are in excess of 1,000 pCi/ml (WSRC 2001).

Figure 3-60 shows a map of a TCE groundwater plume in T-Area. Superimposed on the plume map (actual plume is the shaded areas) is a projected footprint of the plume had remediation not been undertaken, showing the expected southwestern advance (WSRC 1999b). Because of the upward gradient from the Gordon aquifer in T-Area, the plume is confined to the UTR aquifer unit.

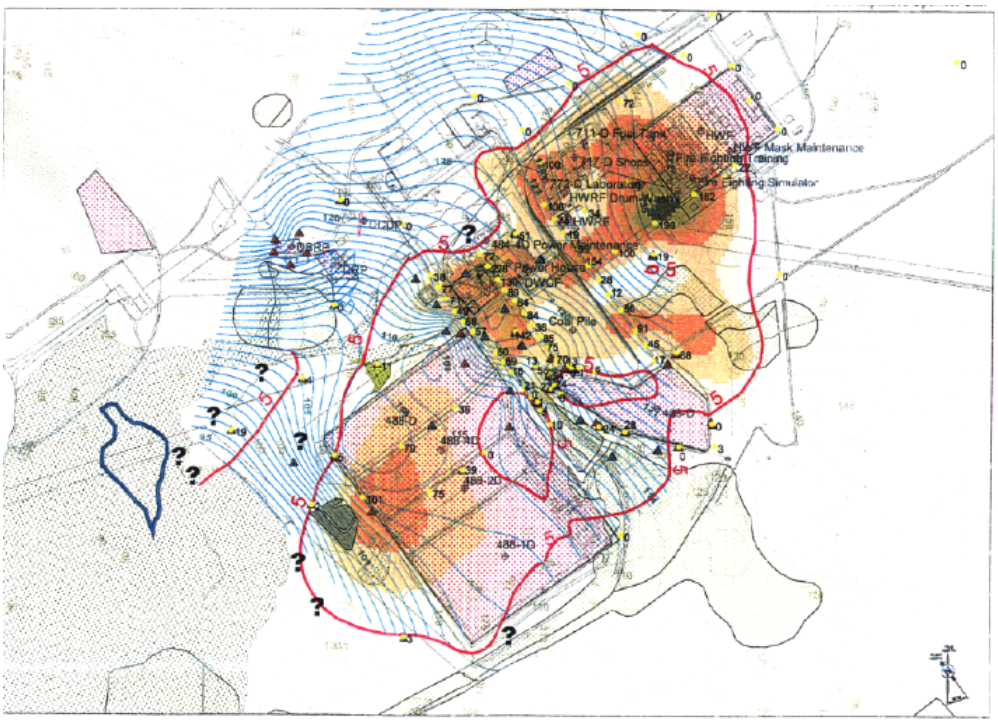


Figure 3-58. TCE Plume in UTR Aquifer Unit (Water Table Aquifer) in D-Area (from WSRC 2001)

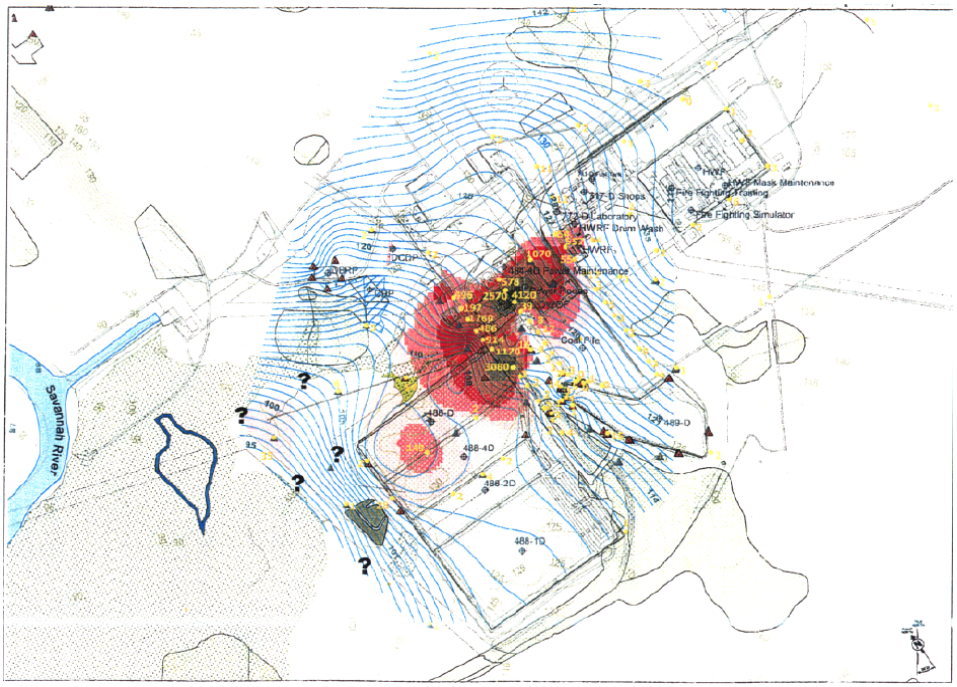


Figure 3-59. Tritium Plume in UTR Aquifer Unit (Water Table Aquifer) under D-Area (from WSRC 2001)

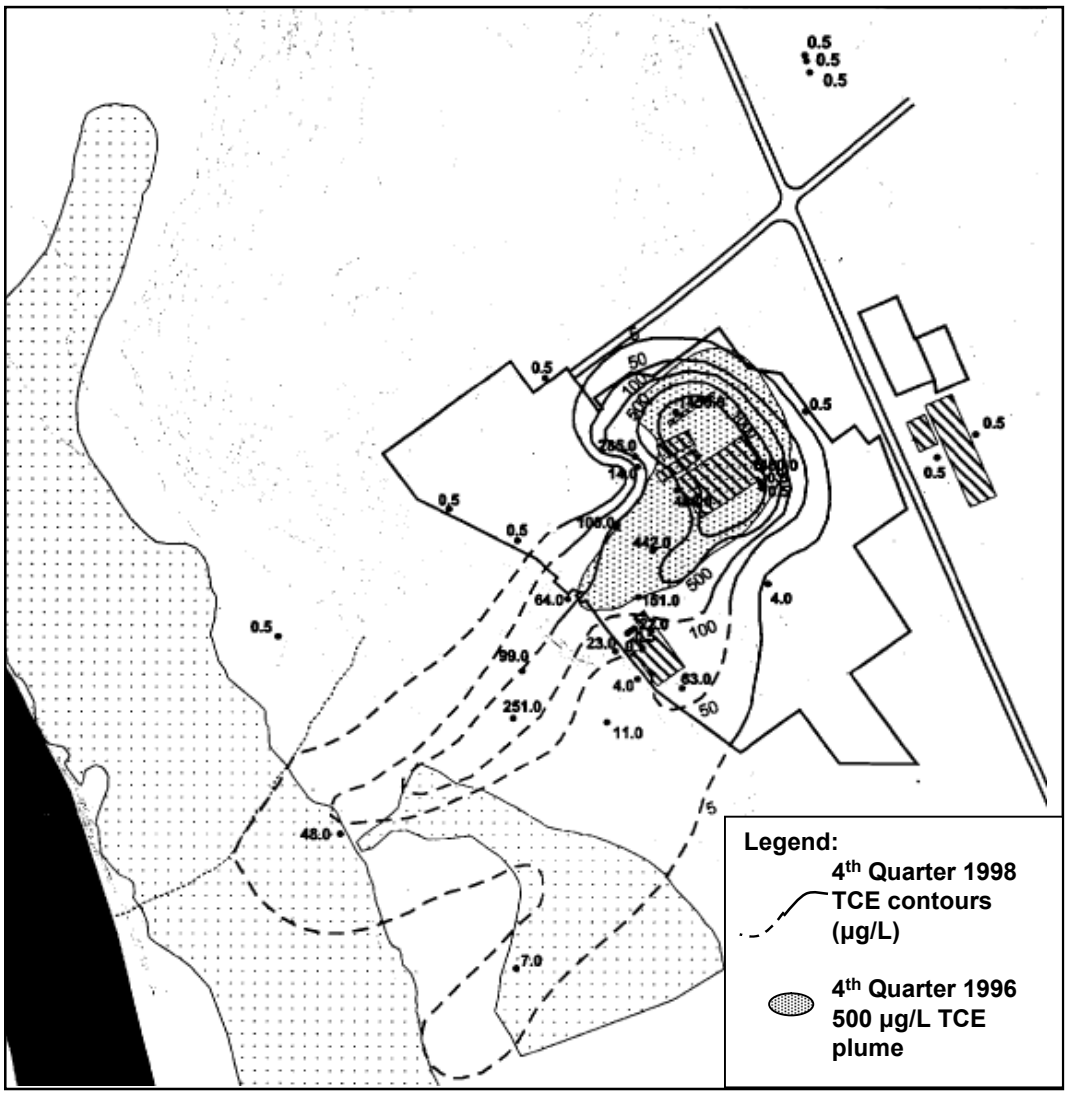


Figure 3-60. TCE Plume Map in T-Area (from WSRC 19

3.2.10.5 C-Area

C-Area, located in the west-central portion of the SRS (Figure 3-1), is at an average ground surface elevation of 87 m above msl. Topographically, it is near the escarpment found along an elevation contour of approximately 48.8m that separates the Aiken Plateau from the Ellenton Plain. The Ellenton Plain is the highest of three step-like surfaces between the Savannah River and the Aiken Plateau (WSRC 2002b). The area is entirely in the Fourmile Branch watershed, and is bounded to the west by Fourmile Branch (approximately one mi from the C Reactor area), to the south by Castor Creek (approximately one mi from the C Reactor area), and to the north by unnamed tributaries of Fourmile Branch (approximately 1/3 mi to the nearest reach).

In the shallower groundwater units, flow is to the northwest toward Fourmile Branch. In general, flow is primarily horizontal, with a small downward gradient in upland areas, and a small upward gradient near discharge boundaries.

Figure 3-40 summarizes the potential pathways to surface water for contaminants in C-Area. As indicated, contaminants under C-Area are potentially discharged to Castor Creek, a tributary of Fourmile Branch, or directly to Fourmile Branch, according to the *SRS End State Vision* hazards analysis (DOE 2005).

3.2.10.5.1 Local Hydrostratigraphy in C-Area

In C-Area, the water table is approximately 21.3 m below the ground surface, and lies in the "upper" aquifer zone of the UTR aquifer unit. The UTR aquifer unit is approximately 30 m thick beneath C-Area and is divided into three zones: two aquifer zones and a tan clay confining zone. In C-Area, the "upper" aquifer zone consists of all saturated strata that lie between the water table and the "tan clay," which is a discontinuous confining zone. The "lower aquifer" zone of the UTR aquifer occurs between the "tan clay" confining zone and the Gordon confining unit (WSRC 2002b).

The Gordon confining unit is from approximately 3.0 m thick near the C-Area (measured about 460 m east of C Reactor), where there is a drop in hydraulic head of about 17 m across the confining unit. The Gordon aquifer is approximately 20 m thick in C-Area, and thickens to the east and southeast.

The Crouch Branch confining unit, which underlies the Gordon Aquifer, is about 53 m thick in the C-Area vicinity. The hydraulic gradient between the Crouch Branch aquifer and the overlying Gordon aquifer is upward in the C-Area vicinity (WSRC 2002b), thereby hydraulically isolating the lower aquifers, and providing an effective barrier to significant contaminant migration into these aquifers.

3.2.10.5.2 Characterization Data for C-Area

Groundwater monitoring in C-Area began prior to 1977. However, as part of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA)/RCRA investigations, many more wells have been installed in the area (Figure 3-61). The C-Area monitoring results indicate that the most prevalent contaminants found in groundwater have been, and continue to be, tritium and TCE (WSRC 2002b).

3.2.10.5.2.1 Potentiometric Surface Map for C-Area

The potentiometric surface of the water table unit under C-Area is shown in Figure 3-62. Figure 3-62 is a map based on long-term mean elevations measured for the UTR aquifer unit. The general configuration shows an east-to-west trending divide, and partitioning of groundwater flow north to Fourmile Branch, and south to Pen Branch. The effect of Fourmile Branch tributaries on the water table elevation contours can be seen in the areas north and south of C-Area (Hiergesell 1998).

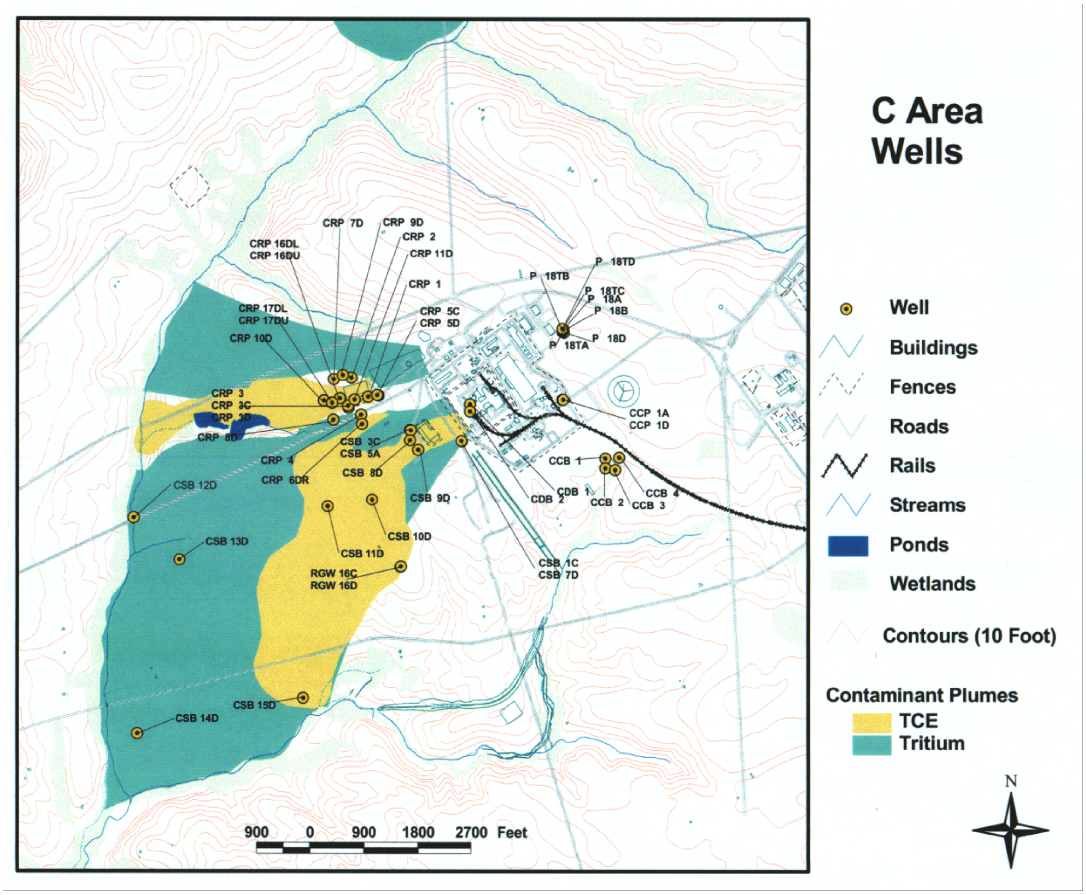


Figure 3-61. CPT and Groundwater Monitoring Well Locations in C-Area circa 2004

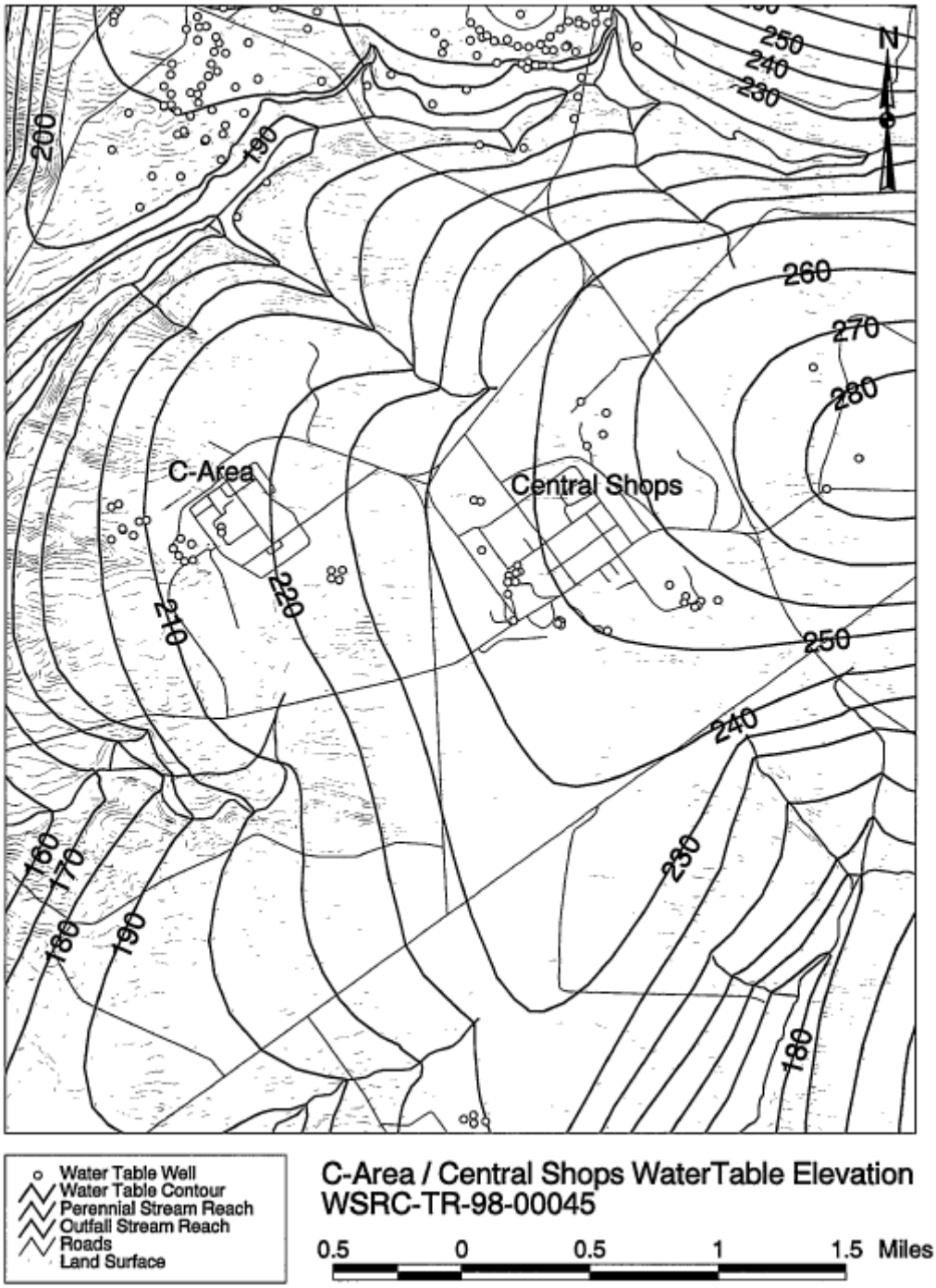


Figure 3-62. Potentiometric Surface of the UTR Aquifer Unit (Water Table Aquifer) for C- and N-Areas (Central Shops) with Elevations in ft above msl (from Hiergesell 1998)

3.2.10.5.2.2 Hydraulic Conductivities in C-Area

It is reasonable to assume that hydraulic conductivity measurements based on pumping tests for the GSA would be applicable to the C-Area aquifer units, since the hydrostratigraphic units are fairly well correlated. These are provided in Table 3-18.

3.2.10.5.2.3 Plume Maps for C-Area

Coincident with the RCRA process, five major groundwater contaminant plumes in C-Area have been identified by CPT investigations: Tritium Plumes 1, 2, 3; the Reactor TCE Plume; and the C-Area Burning/Rubble Pit (CBRP) TCE Plume Figure 3-63. The suspected principal contaminant sources are the C-Area Reactor Seepage Basins, CBRP, C-Area Retention Basin, and facilities, activities and maintenance associated with reactor operations within the C-Reactor area fence line (WSRC 2002b).

Tritium Plume 3, discovered in 1998-1999, originates from the reactor seepage basins and the C-Reactor area, and covers an area of about 1.25 mi^2. The plume is located at the top of the water table, and is approximately 24 m thick at the deepest portion, tapering to about 3 m before discharging into Castor Creek and Fourmile Branch. The plume follows the water table surface, and is not migrating to the other side of the streams to which it discharges (WSRC 2002b).

It has been concluded based upon characterization and groundwater modeling (Fogle 2003) that Tritium Plume 2 originated from various but unspecified depleted sources from within the reactor area. The Reactor TCE Plume originated from a newly discovered vadose zone source outside of the reactor assembly area within the C-Reactor area, and commingles with Tritium Plume 3. Tritium Plume 1 and 3 are indistinguishable downgradient of the reactor seepage basins. These plumes are all located in the upper zone of the UTR aquifer unit (WSRC 2002b).

Tritium Plume 2 is located northwest of C-Area. It flows west toward Fourmile Branch. It is located at the bottom of the “lower” zone of the UTR aquifer unit, and appears to be migrating along the top of the Gordon confining unit. Discharge locations at groundwater seeps along Fourmile Branch have been identified (Fogle 2003). Subsequent characterization has determined that tritium has not migrated below the Gordon confining unit. Additionally, characterization studies on the west side of Fourmile Branch have determined that tritium has not migrated to the far side of Fourmile Branch (Rucker 2000).

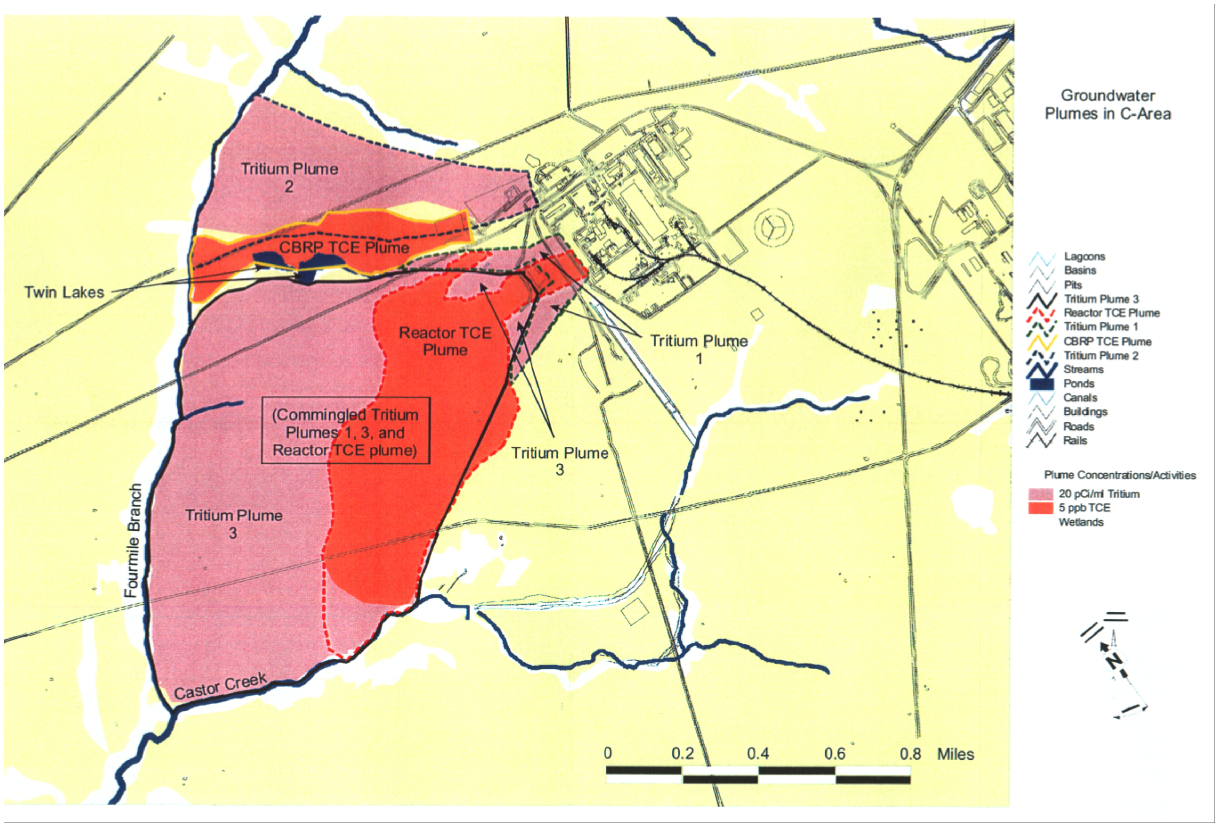


Figure 3-63. Groundwater Plumes in C-Area (from WSRC 2002b)

3.2.10.5.2.4 Particle-Track Analysis for C-Area

The USGS particle-track analysis, described briefly in Section 3.2.10.1.9, indicates that groundwater flowpaths in C-Area move westward through the Gordon aquifer and pass beneath Fourmile Branch to discharge areas along UTR. However, the mean time-of-travel for the particles being tracked is 365 yr. The particle-track analysis, however, does not consider the time of travel downward through the water table aquifer (UTR aquifer) to the Gordon aquifer, which is expected to be several decades near C-Area (USGS 2006). Thus, the present observations, indicating primary discharge areas along Fourmile Branch and its tributary, Castor Creek, are not in conflict with this analysis. The present tritium plumes in C-Area are likely to be intercepted by Fourmile Branch and decay significantly before reaching the Gordon aquifer, and thus may never be observed on the far side of Fourmile Branch.

3.2.10.6 K-Area

K-Area is located in the southwest-central part of the SRS (Figure 3-1) within the Pen Branch Watershed (Figure 3-64). K-Area lies between Indian Grave Branch to the north and west, and Pen Branch, which is approximately one mi to the south (WSRC 1999c). Surface drainage of K-Area is to the southeast, but groundwater flow is to the west-southwest toward Indian Grave Branch, which is less than 0.4 mi from the area (WSRC 1998a).

Figure 3-40 summarizes the potential pathways to surface water for contaminants in K-Area, showing only discharge to Indian Grave Branch (DOE 2005).

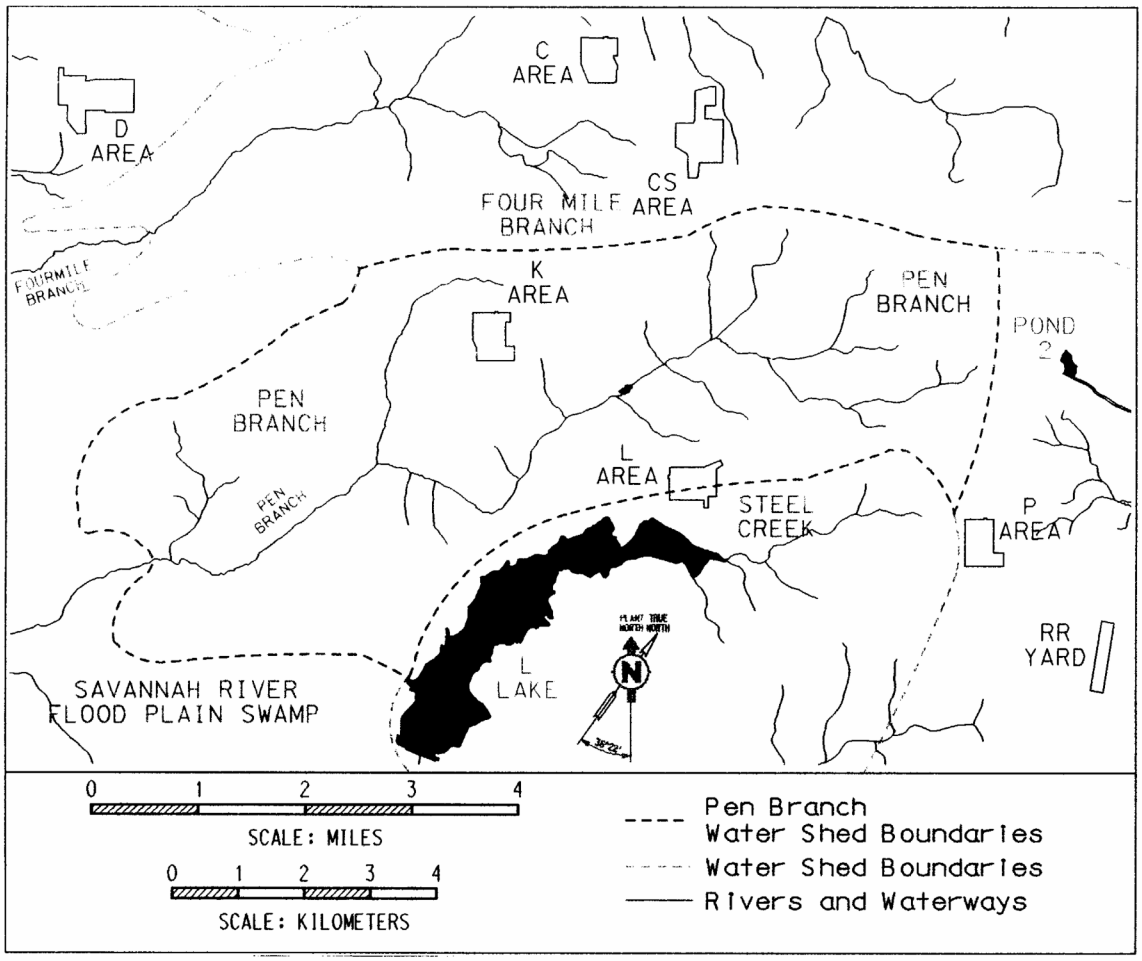


Figure 3-64. Location of K-Area Relative to Watersheds and Streams at SRS (from WSRC 1999c)

3.2.10.6.1 Local Hydrostratigraphy in K-Area

In K-Area, the water table is approximately 15 m below the ground surface, and lies in the UTR aquifer unit (WSRC 1998b). The "tan clay" confining zone of the UTR aquifer unit is 1.6 to 3.1-m thick near K-Area, and is reported to significantly impede downward contaminant migration in the area (WSRC 1998a). The Gordon confining unit (called "Green Clay" in WSRC 1998a) is about 1.24 m thick near K-Area, with a downward change in head of about 7 m across this unit. Data obtained from the SRS P-25 well cluster, approximately 1400 ft southeast of K-Area indicates a weak downward gradient across the Crouch Branch Confining unit (WSRC 1998a). Given this information, and the hydraulic head difference contours shown in Figure 3-37, it is apparent that K-Area is near the boundary between upward and downward gradients between the underlying Crouch Branch aquifer and the Gordon aquifer. If a downward gradient exists in all or parts of K-Area between these two aquifers, it will be very weak.

3.2.10.6.2 Characterization Data for K-Area

A total of nine wells have been installed to monitor groundwater in the vicinity of K-Area. All nine wells are screened in the UTR aquifer unit. Analytical results of groundwater samples collected near K-Area, in the area of the Burning/Rubble Pits and Rubble Pile, indicated the presence of volatile organic compounds, inorganic compounds, and radionuclides (WSRC 1999c). A concentrated tritium plume in groundwater which discharges to Indian Grave Branch has also been investigated in K-Area, and is called the K-Area Tritium Anomaly Plume (WSRC 1993b).

3.2.10.6.2.1 Potentiometric Surface Map for K-Area

The potentiometric surface of the water table unit under K-Area is shown in Figure 3-65. Figure 3-65 is a map based on long-term mean elevations measured for the UTR aquifer unit. The north-south trending groundwater divide is evident for this aquifer unit. Figure 3-65 indicates that groundwater flow is to the south and southwest beneath most of K-Area (Hiergesell 1998).

3.2.10.6.2.2 Hydraulic Conductivities in K-Area

It is reasonable to assume that hydraulic conductivity measurements based on pumping tests for the GSA would be applicable to the K-Area aquifer units, since the hydrostratigraphic units are fairly well correlated. These are provided in Table 3-18. Slug test results reported for the UTR aquifer unit in K-Area averaged somewhat lower (about 7×10^{-6} m/s) (WSRC 1998a).

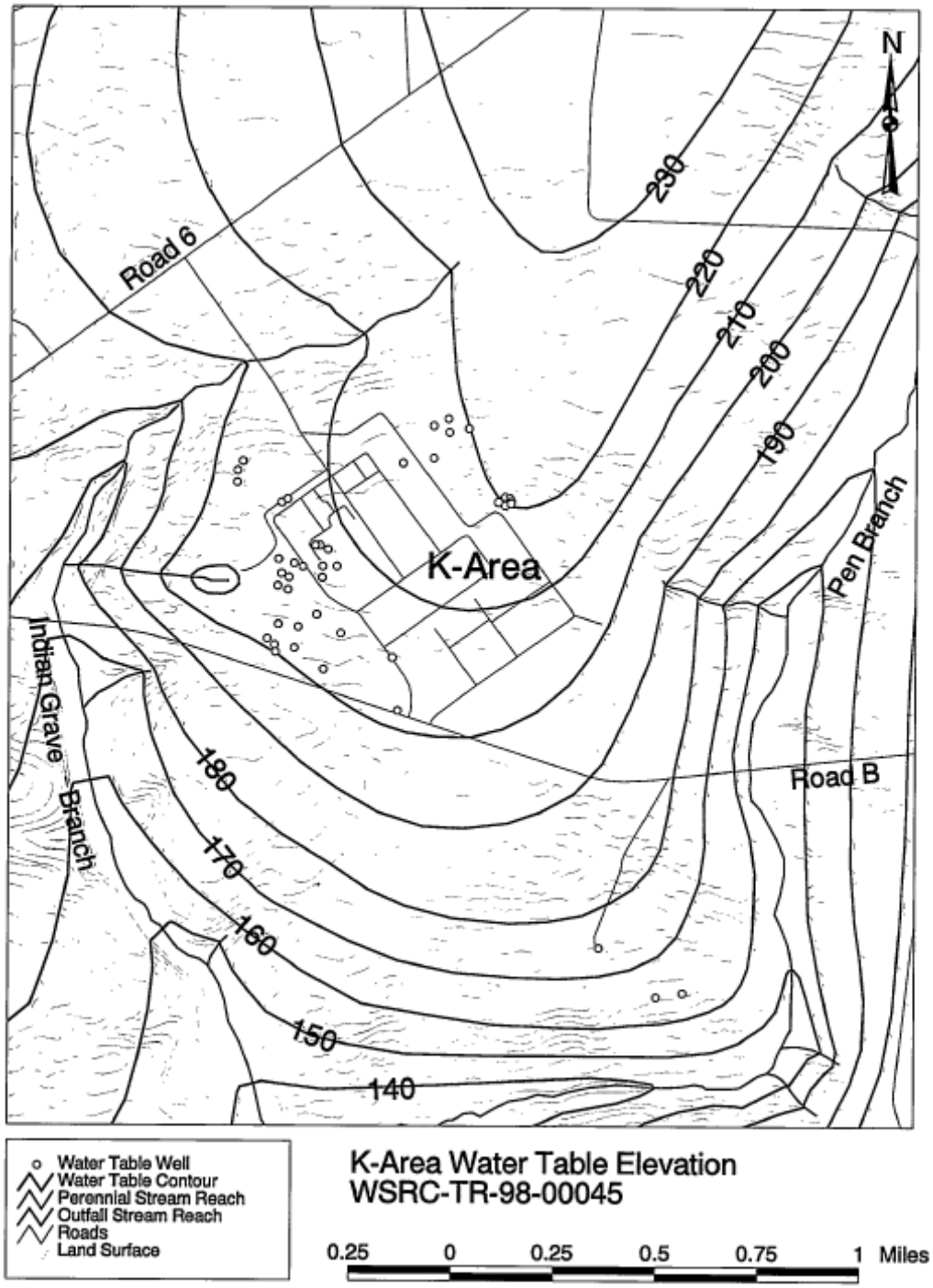


Figure 3-65. Potentiometric Surface of the UTR Aquifer Unit (Water Table Aquifer) for K-Area with Elevations in ft above msl (from Hiergesell 1998)

3.2.10.6.2.3 Plume Maps for K-Area

Following CERCLA/RCRA investigations, several contaminants have been found to be associated with groundwater in the K-Area vicinity, including organic compounds and tritium. Figure 3-66 shows three separate groundwater plumes that have been identified in K-Area. The two plumes on the left side of Figure 3-66 represent tritium plumes, the upper one is associated with a retention basin, and the lower one (the Tritium Anomaly Plume) with an as yet unidentified source. The retention basin received purge water from the reactor disassembly basin. The Anomaly Plume was named as such due to the high tritium concentrations initially detected in groundwater associated with the plume (in excess of 500,000 pCi/ml). The plume on the right side of Figure 3-66 is a VOC plume associated with the K-Burning Rubble Pit. These plume maps confirm that groundwater flow in the UTR aquifer beneath K-Area is toward Indian Grave Branch for the bulk of K-Area. A comprehensive characterization of K-Area groundwater is scheduled in the future.

Figure 3-66. K-Area Groundwater Plumes (tritium plumes on left; VOC plume on right)

3.2.10.6.2.4 Particle-Track Analysis for K-Area

The USGS particle-track analysis, described briefly in Section 3.2.10.1.9, indicates the primary discharge areas for groundwater in the SRS model zone in which K-Area resides (Zone 2 in the study) are UTR to the west and Pen Branch to the south. In K-Area, groundwater flowpaths move southeastward to discharge to areas along Pen Branch (USGS 2006).

3.2.10.7 L-Area

L-Area is located in the southwest-central part of the SRS (Figure 3-1 and Figure 3-64); near the escarpment found along an elevation contour of approximately 48.8 m that separates the Aiken Plateau from the Ellenton Plain. The Ellenton Plain is the highest of three step-like surfaces between the Savannah River and the Aiken Plateau (WSRC 2005a). L-Area is in an area of low to moderate relief, located on the southern flank of a hill situated on the surface water divide between the Steel Creek watershed to the southeast and the Pen Branch watershed to the northwest. The L-Area Southern Groundwater Operable Unit, which encompasses groundwater southeast of the divide to L-Lake, is within the Steel Creek watershed. The topographic elevation ranges from about 76 m above msl near the northeast corner of the area to approximately 58 m above msl at L-Lake (WSRC 2005a).

Surface drainage of L-Area is toward L-Lake, along Steel Creek. The lake level is maintained at 58 m above msl to prevent redistribution of, and exposure to, contaminants that reside in the lake's sediments (WSRC 2005a).

In the shallower groundwater zones, flow is towards the nearest surface water body (Figure 3-11 and Figure 3-64). Pen Branch, Steel Creek, L Lake, and Meyers Branch all receive water from the water table unit. The lower portion of L Lake converts to a losing water body at approximately 58 m above msl, causing subsurface flow from the lake and towards Pen Branch and Meyers Branch. In the saturated zone, groundwater flow is primarily horizontal, with a small downward component in upland areas and a small upward component near discharge boundaries (i.e., the receiving surface water bodies). Flow in the "lower" zone of the water table unit is generally southeast toward L Lake, Pen Branch, and Meyers Branch. Pen Branch comes within 760 m of L Lake – at that point, flow in all aquifer zones of the water table unit is from L Lake to Pen Branch. A small amount of groundwater flows downward through the Gordon confining unit into the Gordon aquifer unit near L-Area. Groundwater in the Gordon aquifer unit flows horizontally to the southeast, towards the Savannah River (GeoTrans 2004).

Figure 3-40 summarizes the potential pathways to surface water for contaminants in L-Area. As indicated in this figure, contaminants under L-Area are potentially discharged to either Pen Branch or L-Lake, according to the *SRS End State Vision* hazards analysis (DOE 2005). This figure does not show the subsequent flow from the lake to Pen Branch and Meyers Branch, nor does it indicate the small amount of groundwater in the Gordon aquifer flowing towards the Savannah River.

3.2.10.7.1 Local Hydrostratigraphy in L-Area

In L-Area, the water table is approximately 5 to 8 m below the ground surface, and lies in the "upper" aquifer zone of the UTR aquifer unit. The UTR aquifer unit is approximately 40-m thick beneath L-Area and is informally divided into three zones: two aquifer zones and a tan clay confining zone. The "upper" aquifer zone consists of all saturated strata that lie between the water table and the "tan clay," which is about 4.6-m thick south of L-Area, and 2.4-m thick north of L-Area. The "lower aquifer" zone of the UTR aquifer occurs between the "tan clay" confining zone and the Gordon confining unit and varies between 6 and 18 m in thickness (WSRC 2005a).

The Gordon confining unit is approximately 6-m thick in the southern part of L-Area, is semi-confining, and may be a leaky confining unit in parts of L-Area. The Gordon aquifer is approximately 12 m thick near L-Area and thickens to the east and southeast (WSRC 2005a).

The Crouch Branch confining unit, which underlies the Gordon Aquifer, contains several thick and relatively continuous clay beds with interbedded sands near L-Area. The unit is about 59 m thick in the region between L-Area and L-Lake. The hydraulic gradient is upward from the underlying Crouch Branch aquifer unit to the overlying Gordon aquifer unit; thus, contaminants are not expected to reach the Crouch Branch aquifer in this area (WSRC 2005a).

3.2.10.7.2 Characterization Data for L-Area

Groundwater monitoring in L-Area has been carried out since the early 1980s, and a more intensive effort has been applied in support of RCRA investigations. As a result of these investigations, plumes of TCE, PCE and tritium were discovered at concentrations in groundwater that warranted action (WSRC 2005a). Locations of monitoring wells in L-Area are depicted in Figure 3-67.

3.2.10.7.2.1 Potentiometric Surface Map for L-Area

The potentiometric surface of the water table unit under L-Area is shown in Figure 3-68. Figure 3-68 is a map based on long-term mean elevations measured for the Steed Pond aquifer unit. The groundwater divide that separates groundwater moving toward Pen Branch to the north and Steel Creek (and L-Lake) to the south is evident. Groundwater movement beneath L-Area is southward toward L-Lake (Hiergesell 1998).

3.2.10.7.2.2 Hydraulic Conductivities in L-Area

It is reasonable to assume that hydraulic conductivity measurements based on pumping tests for the GSA would be applicable to the L-Area aquifer units, since the hydrostratigraphic units are fairly well correlated. These are provided in Table 3-18. Slug test results reported for the UTR aquifer unit in K-Area averaged somewhat lower (about 7×10^{-6} m/s) (WSRC 1998a).

3.2.10.7.2.3 Plume Maps for L-Area

Two distinct commingled plumes of volatile organic compounds and tritium extend from the L Reactor area and discharge in L-Lake (Figure 3-69). Historical primary sources of tritium in these two plumes were the reactor seepage basin, a chemical and oil basin, the L-Area Hot Shop, and the area within the L Reactor fence line. A separate tritium plume exists west of the L Reactor area, which is moving in a southwest direction between Pen Branch and L-Lake. The source of the western plume appears to have been an emergency retention basin west of the reactor facility (WSRC 2005a).

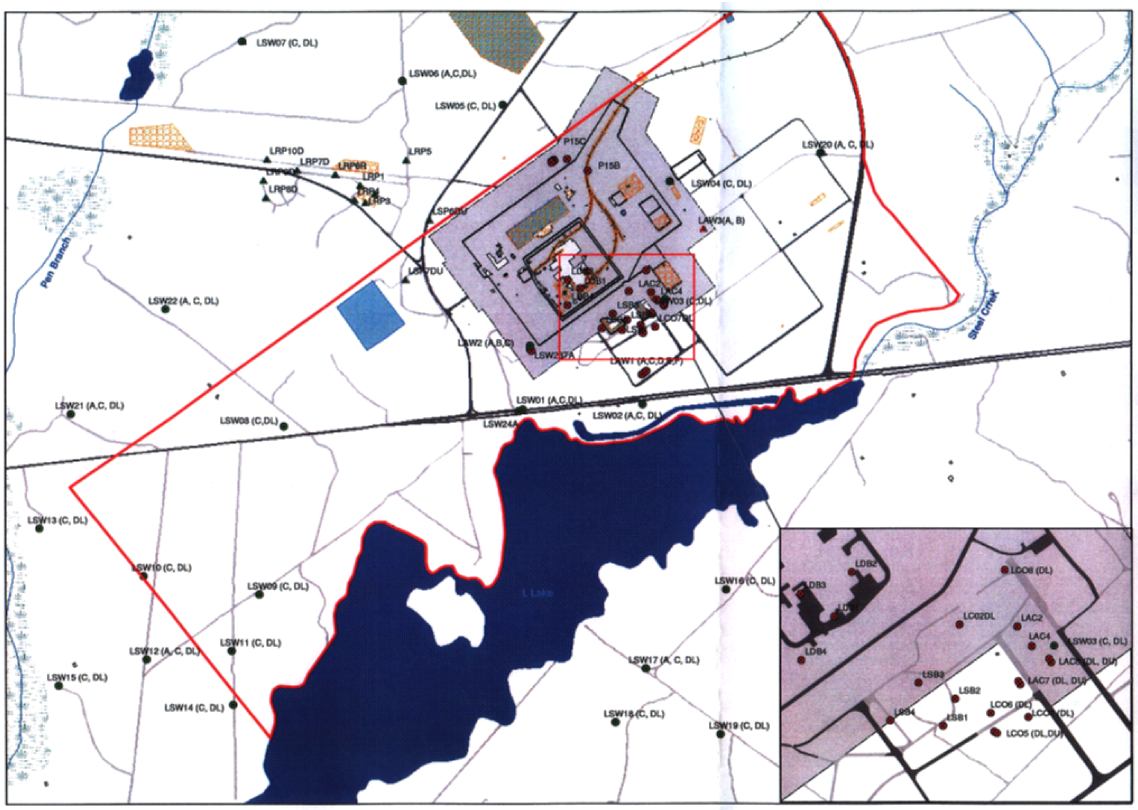


Figure 3-67. Groundwater Monitoring Wells in L-Area, in Support of CERCLA/RCRA Investigations (from WSRC 2005a)

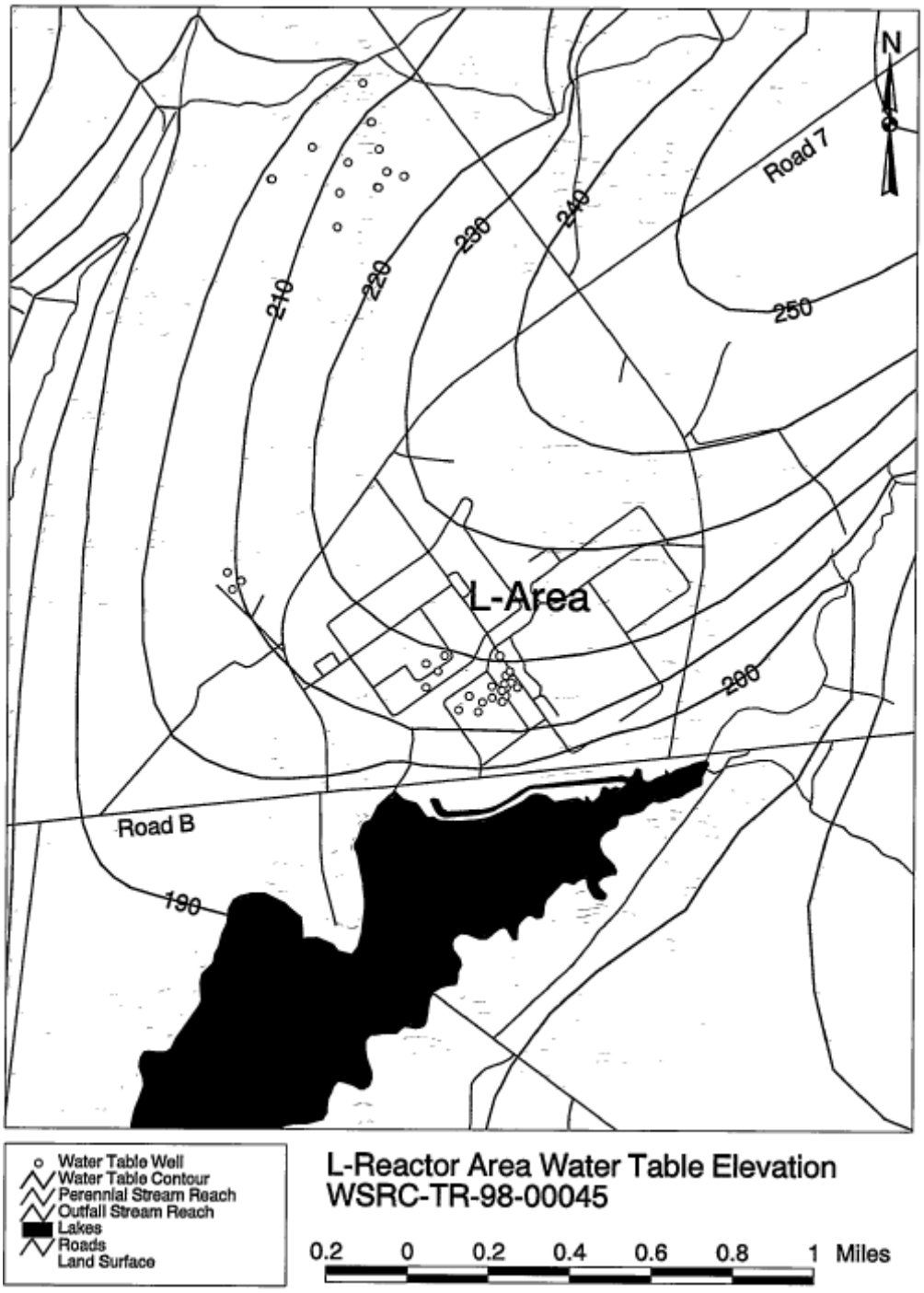


Figure 3-68. Potentiometric Surface of the UTR Aquifer Unit (Water Table Aquifer) for L-Area with Elevations in ft above msl (from Hiergesell 1998)

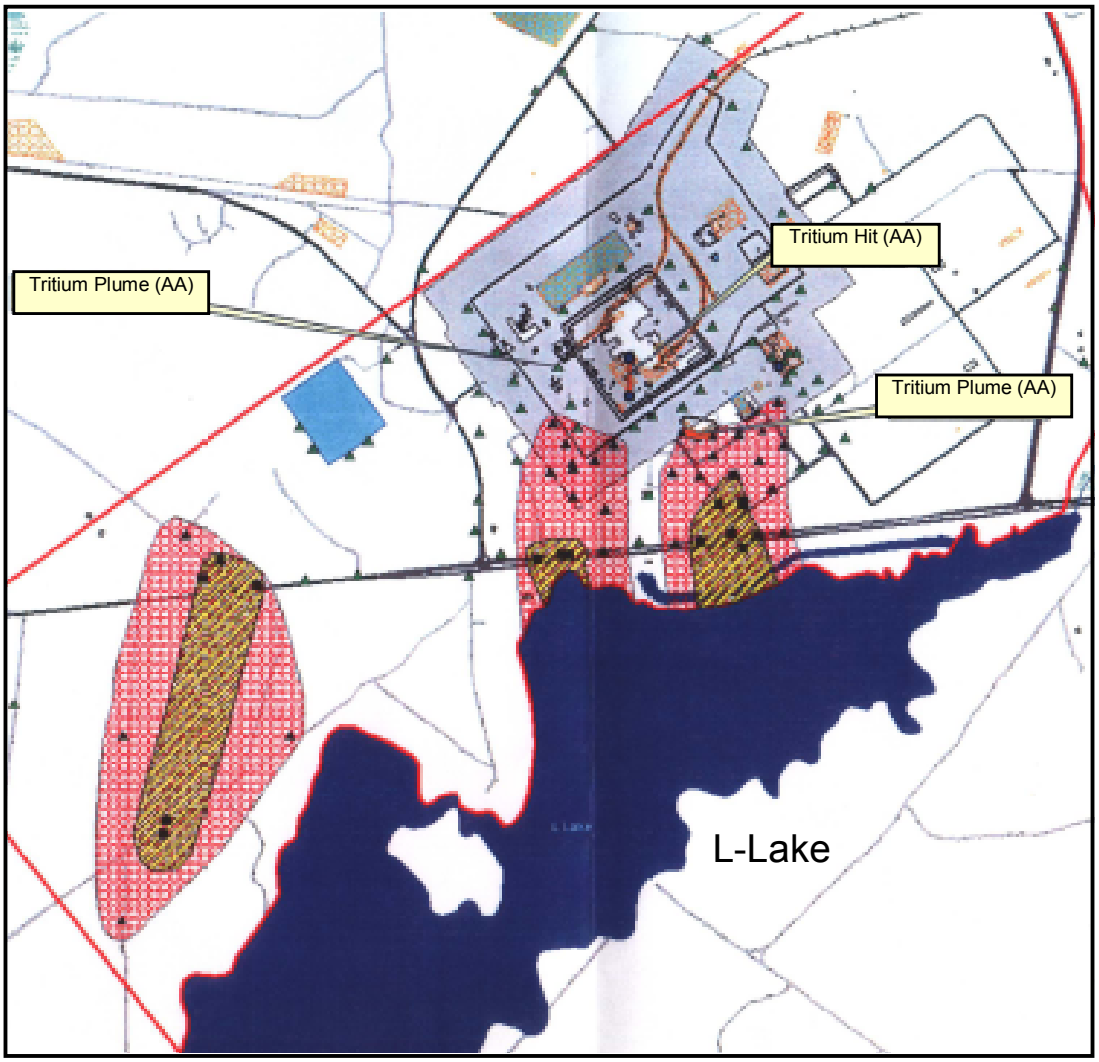


Figure 3-69. Tritium Plumes in the UTR Aquifer Unit of L-Area (from WSRC 2005a)

3.2.10.8 P-Area

P-Area is located in the south-central part of the SRS (Figure 3-1). The reactor area sits on an upland area, at greater than 100 m above msl, which is characterized by flat to gently rolling topography and is located between Steel Creek to the southwest and unnamed tributaries to PAR Pond to the northeast, and Meyers Branch to the southeast (Figure 3-11). It is located in both the Steel Creek and LTR watersheds. The headwaters of Steel Creek, at an elevation of approximately 85 m, originate near P-Area (Rabin et al. 2004, WSRC 2006d).

Generally, groundwater flow directions in the water table unit diverge at P-Area with flows toward Steel Creek to the northwest, PAR Pond to the northeast, and Meyers Branch to the southeast. In the Gordon aquifer unit, flow under P-Area is generally to the southeast toward Meyers Branch (WSRC 2006d).

Figure 3-40 summarizes the potential pathways to surface water for contaminants in P-Area. As indicated in this figure, contaminants under P-Area are potentially discharged to either Steel Creek, Meyers Branch, or "Pond 2," which ultimately ends up in PAR Pond, according to the *SRS End State Vision* hazards analysis (DOE 2005).

3.2.10.8.1 Local Hydrostratigraphy in P-Area

In P-Area, the water table is approximately 15 m below the ground surface, and lies in the "upper" aquifer zone of the UTR aquifer unit (WSRC 2006d). The UTR aquifer unit under P-Area is divided into three zones: two aquifer zones and a tan clay confining zone (WSRC 2006d). The "upper" aquifer zone consists of all saturated strata that lie between the water table and the "tan clay," and is approximately 17-m thick in P-Area. The "tan clay" zone has an average thickness in the area of 7 m (WSRC 2006d). The "lower aquifer" zone of the UTR aquifer occurs between the "tan clay" confining zone and the Gordon confining unit.

3.2.10.8.2 Characterization Data for P-Area

Several groundwater monitoring wells and piezometers are installed in and around P-Area to monitor groundwater quality as a result of previous reactor operations and to obtain water level measurements. Currently, there are 150 wells, of which 82 are designated as piezometers and are specifically used for collecting water-level measurements (WSRC 2006d). Figure 3-70 shows the location of wells around P-Area (WSRC 2006d). Monitoring of the P-Reactor groundwater operating unit, as defined by CERCLA/RCRA, has been conducted since 1986.

3.2.10.8.2.1 Potentiometric Surface Maps for P-Area

The potentiometric surface of the water table unit under P-Area is shown in Figure 3-71. Figure 3-71 is a map based on long-term mean elevations measured for the UTR aquifer unit (Hiergesell 1998). The water table configuration indicates a divide that extends from the northwest to southeast, partitioning flow between Steel Creek to the southwest and tributaries of LTR to the northeast. A groundwater mound area exists beneath P-Area, and flow radiates outward in all directions except to the northwest (Hiergesell 1998). Figure 3-72 shows the potentiometric surface in the Gordon Aquifer, indicating flow under P-Area in the south-southeasterly direction (WSRC 2006d).

3.2.10.8.2.2 Hydraulic Conductivities in P-Area

It is reasonable to assume that hydraulic conductivity measurements based on pumping tests for the GSA would be applicable to the P-Area aquifer units, since the hydrostratigraphic units are fairly well correlated. These are provided in Table 3-18. These values are very consistent with the ranges reported for previous modeling efforts in P-Area (Rabin et al. 2004).

3.2.10.8.2.3 Plume Maps for P-Area

In general, tritium, PCE, and TCE are the most prevalent contaminants found in groundwater local to P-Area. Tritium activities in exceedance of the Maximum Contaminant Level (MCL) (20 pCi/mL) have been exhibited throughout the "upper" aquifer zone, "tan clay" confining zone, and the "lower" zone of the UTR aquifer unit. The highest tritium activities in the "upper," "tan clay" confining, and "lower" zones were 25,000 pCi/mL, 23,600 pCi/mL, and 14,500 pCi/mL, respectively. As shown in Figure 3-73 and Figure 3-74, tritium plumes extend from various locations under P-Area to upper Steel Creek in both the "upper" and "lower" UTR aquifer zones, respectively. There is also a tritium plume east of P Reactor that is moving to the east (WSRC 2006d).

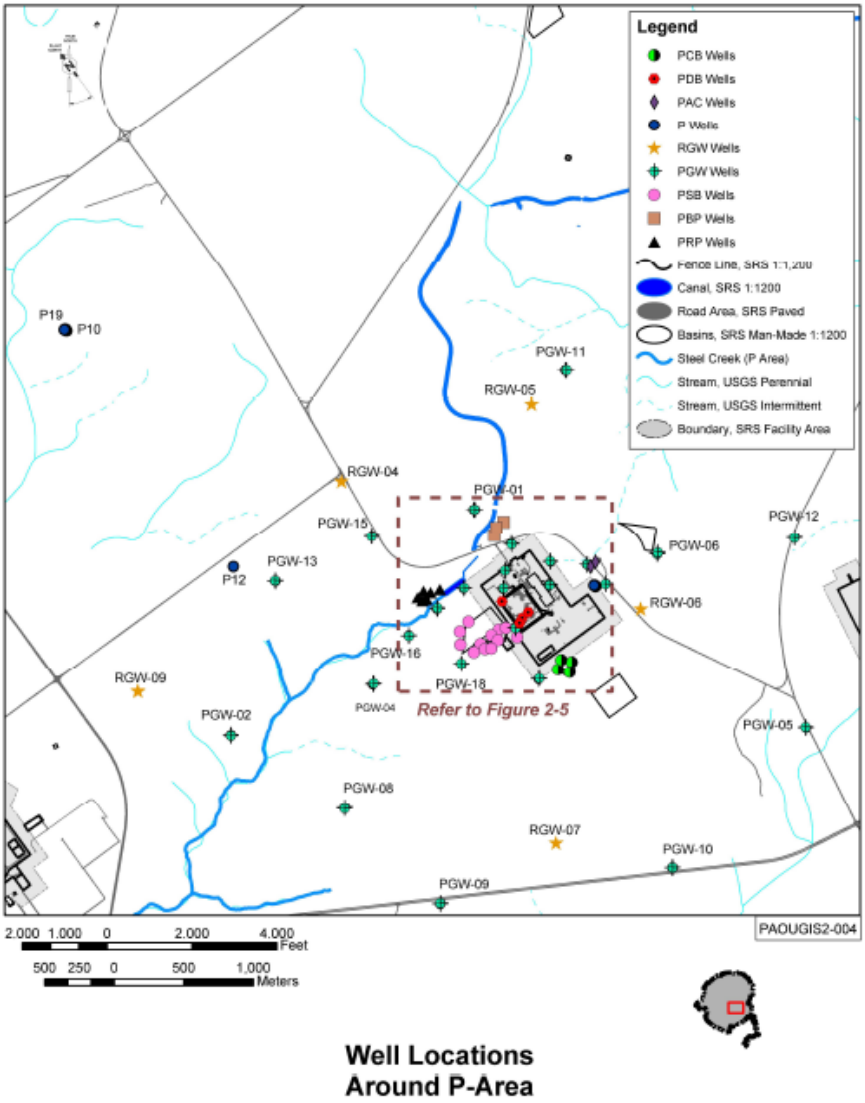


Figure 3-70. P-Area Monitoring Wells and Known Groundwater Plumes (from WSRC 2006d)

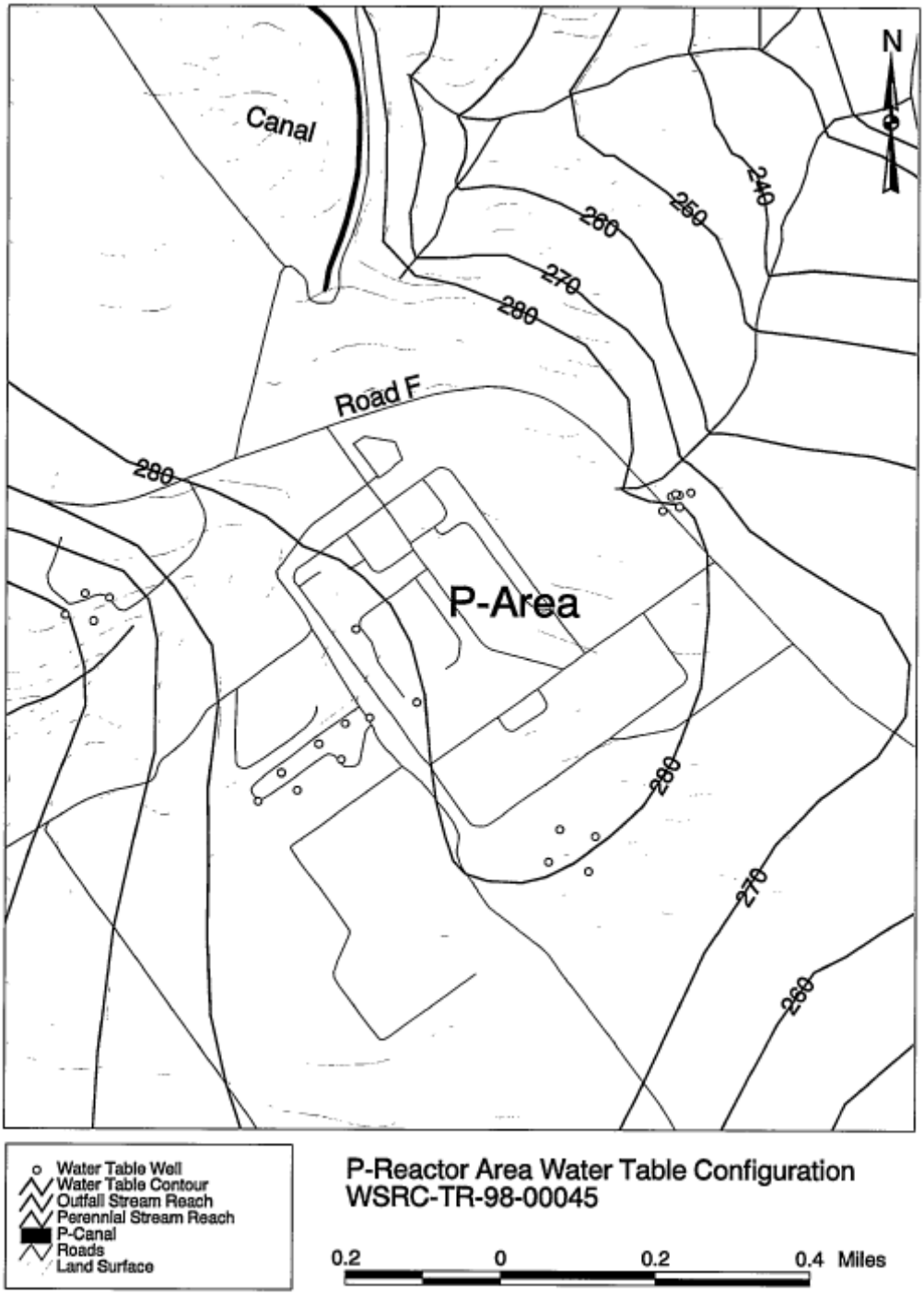


Figure 3-71. Potentiometric Surface of the UTR Aquifer Unit (Water Table Aquifer) for P-Area with Elevations in ft above msl (from Hiergesell 1998)

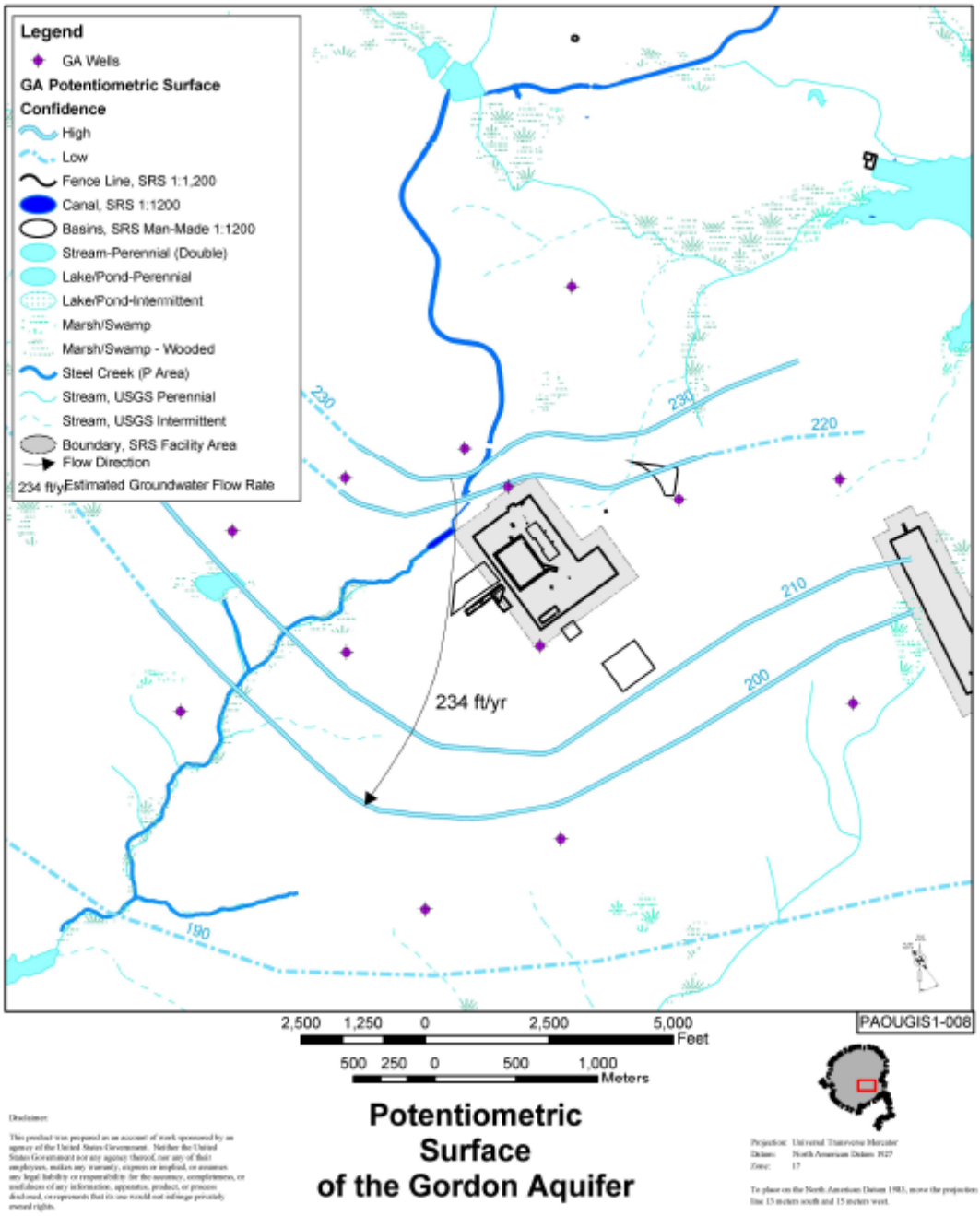


Figure 3-72. Potentiometric Surface of the Gordon Aquifer Unit (Water Table Aquifer) for P-Area with Elevations in ft above msl (from WSRC 2006d)

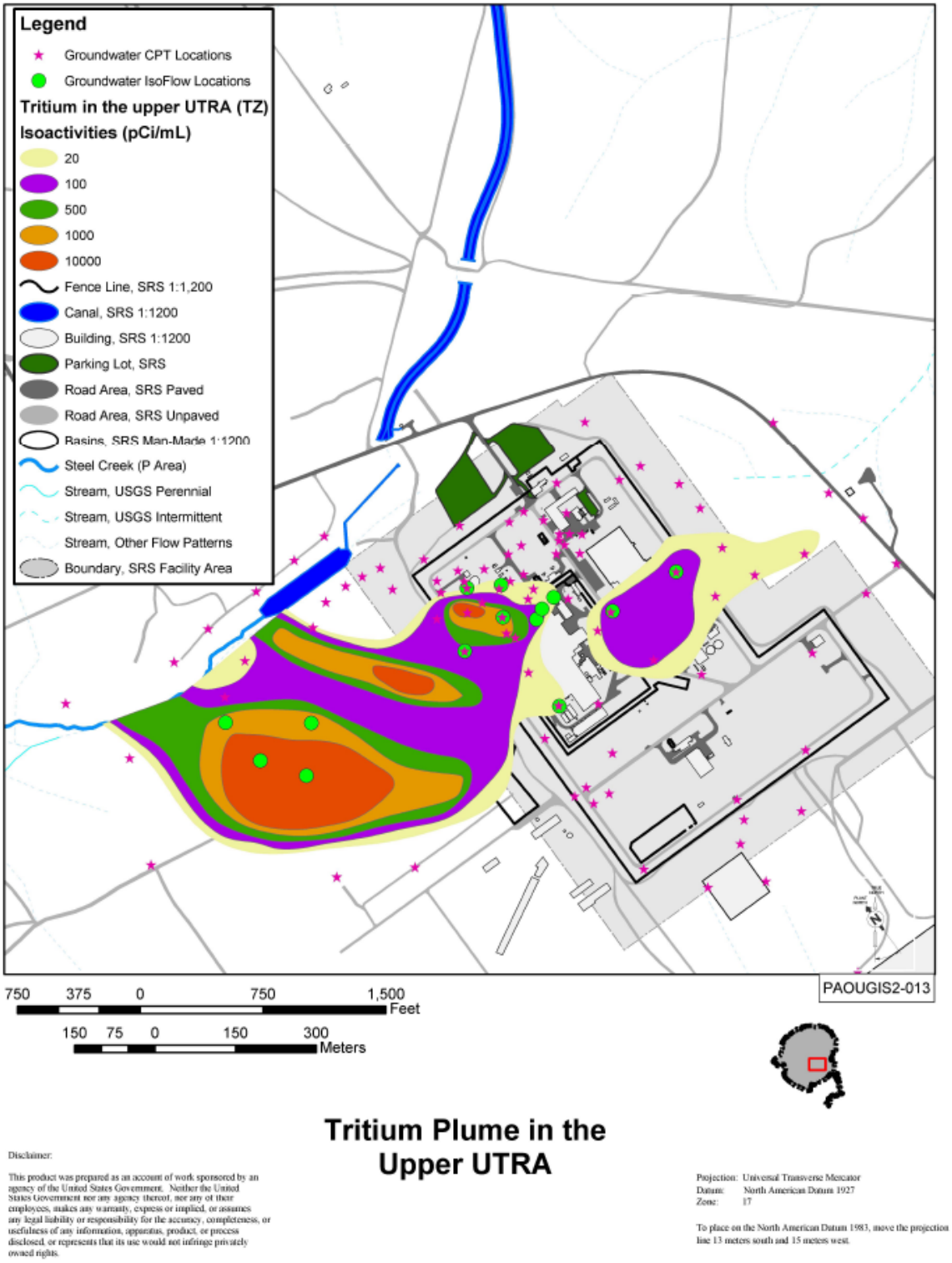


Figure 3-73. Plume Map for Tritium in the "Upper" Zone of the UTR Aquifer Unit under P-Area (from WSRC 2006d)

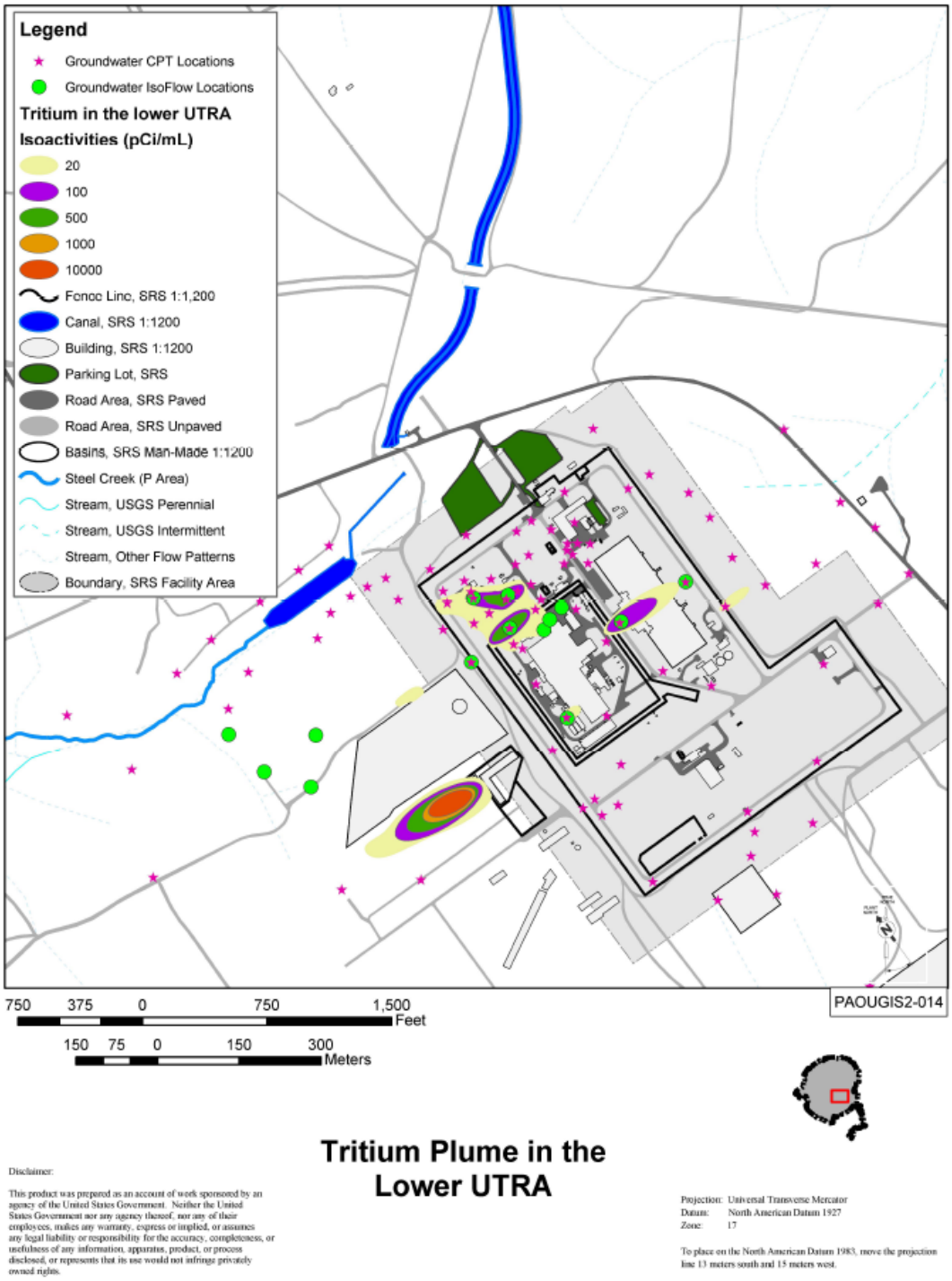


Figure 3-74. Plume Map for Tritium in the "Lower" Zone of the UTR Aquifer Unit under P-Area (from WSRC 2006d)

3.2.10.9 R-Area

R-Area is located in the east-central portion of the SRS (Figure 3-1). The area has flat-to-gently rolling topography, and is approximately 91.4 m above msl (WSRC 2006b). R-Area is located on a groundwater divide. Groundwater under the northern portion of the area flows north towards the UTR watershed. Groundwater flow under the southern and eastern portion is generally eastward towards PAR Pond, its associated cooling ponds, and the canal system. The canal system located north and east of R-Area represents an artificial point of discharge (WSRC 2006b).

Figure 3-40 summarizes the potential pathways to surface water for contaminants in R-Area. As indicated in this figure, contaminants under R-Area are potentially discharged to "Pond A" (which feeds PAR Pond), according to the *SRS End State Vision* hazards analysis (DOE 2005). The summary figure does not indicate potential flow of contaminants toward UTR via Mill Creek, but the conceptual model specific to the portion of R-Area in the UTR watershed (Figure 3-75) indicates flow to UTR via Mill Creek may occur (DOE 2005).

3.2.10.9.1 Local Hydrostratigraphy in R-Area

In R-Area, the water table lies in the "upper" aquifer zone of the UTR aquifer unit. Water table elevations range from approximately 87 m above msl in the vicinity of the seepage basins north of the reactor area, to approximately 88 to 76 m above msl at the R-Area Burning Rubble/Rubble Pits southeast of the reactor area (WSRC 2006b).

The UTR aquifer unit under R-Area is divided into three zones: two aquifer zones and a tan clay confining zone (WSRC 2002c). The "upper" aquifer zone consists of all saturated strata that lie between the water table and the "tan clay," and is approximately 32- to 38-m thick under the seepage basins. The "tan clay" zone occurs at a depth of 18 to 46 m below land surface with a thickness varying from 2 to 17 m. The "lower aquifer" zone of the UTR aquifer occurs between the "tan clay" confining zone and the Gordon confining unit, and is approximately 20.5-m thick in R-Area. The Gordon confining unit is approximately 2.5-m thick near the R-Area (WSRC 2002c).

3.2.10.9.2 Characterization Data for R-Area

Groundwater monitoring in R-Area began prior to 1957. In November 1957, a failed calorimeter experiment resulted in the discharge of large volumes of radioactive wastewater to the effluent canal and Seepage Basin 1. Although R Reactor was shut down in 1964, and decontamination activities were conducted, approximately 5 million gallons of wastewater were discharged to a total of six seepage basins (Brewer and Fogle 2004).

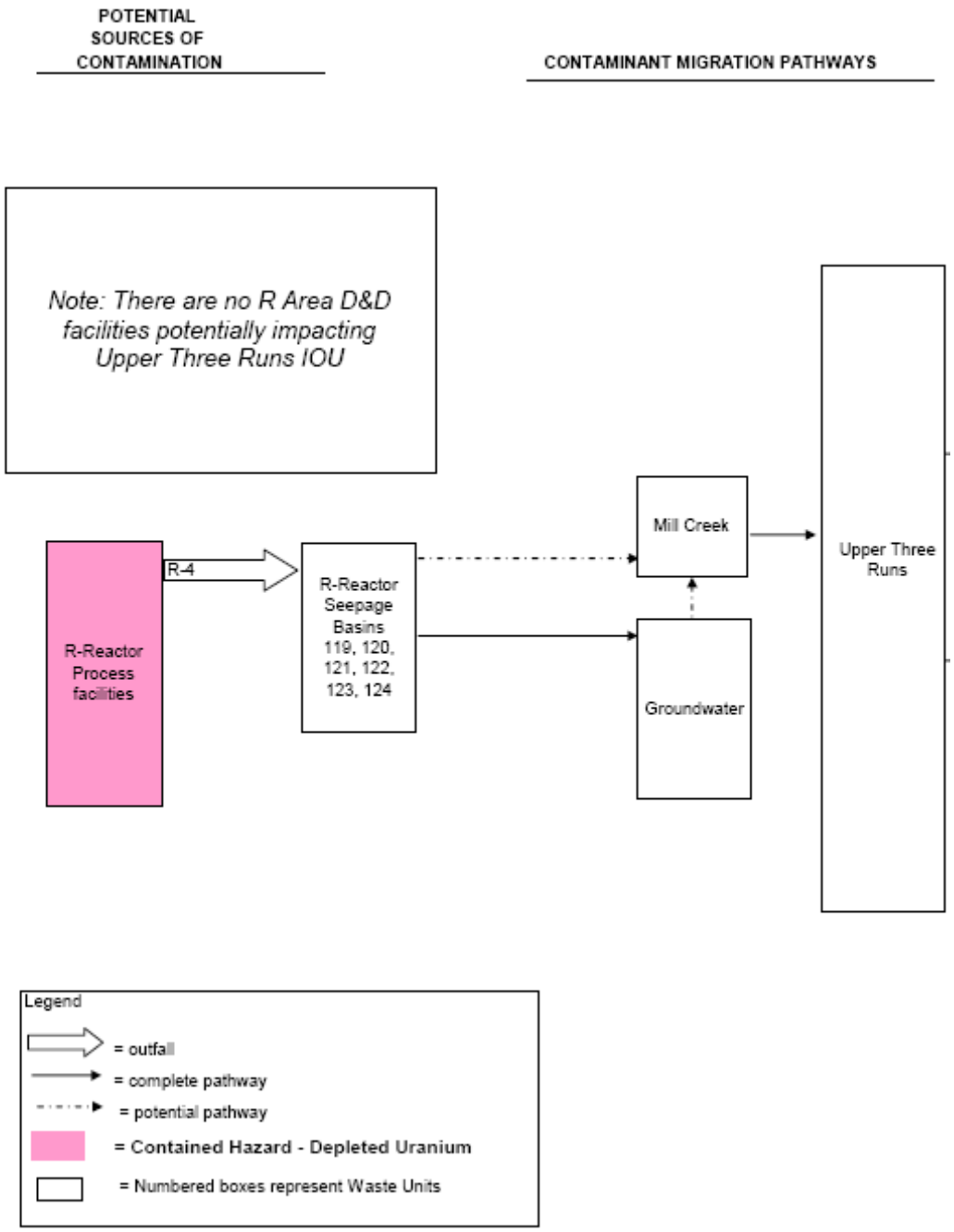


Figure 3-75. Migration Pathways for R-Area Contaminants in Upper Three Runs Watershed (DOE 2005)

Monitoring well networks were installed in R-Area in 1983, and a well cluster was installed east of R-Area in 1985 as part of the baseline hydrogeologic study. Presently over 200 monitoring wells are located in R-Area, most of which are completed in the "upper" zone of the UTR aquifer unit (WSRC 2006b). Figure 3-76 shows the location of some of these wells in R-Area, as well as the location of known groundwater plumes in the area.

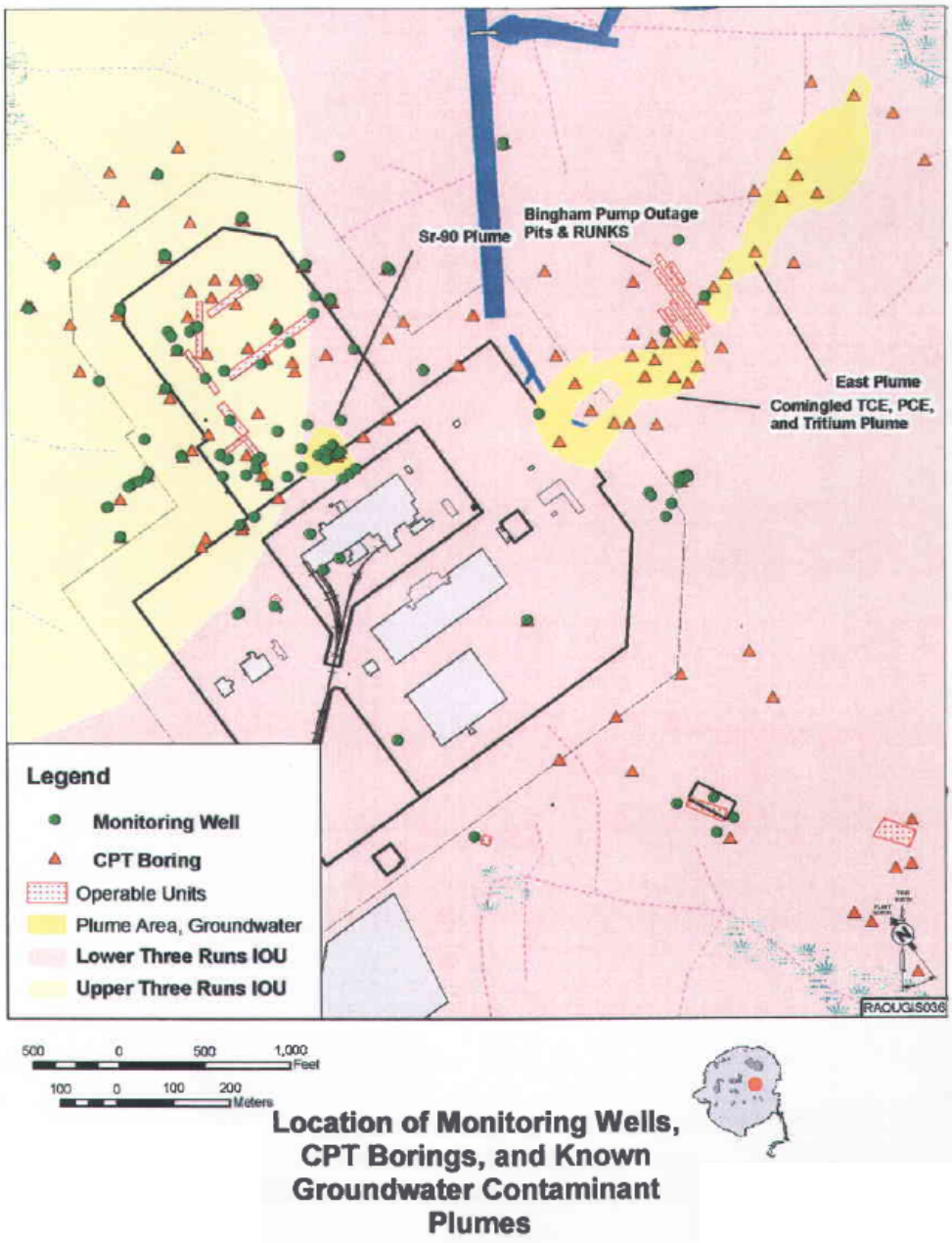


Figure 3-76. R-Area Monitoring Wells and Known Groundwater Plumes (from WSRC 2006b)

3.2.10.9.2.1 Potentiometric Surface Map for R-Area

The potentiometric surface of the water table unit under R-Area is shown in Figure 3-77. Figure 3-77 is a map based on long-term mean elevations measured for the UTR aquifer unit (Hiergesell 1998). The water table configuration indicates a mounded area, with flow to the north toward discharge points along Mill Creek, flow northeast to a stream that runs into PAR Pond, and flow to the southeast toward PAR Pond. The mounded groundwater is suspected of reflecting perched conditions (Hiergesell 1998).

3.2.10.9.2.2 Hydraulic Conductivities in R-Area

It is reasonable to assume that hydraulic conductivity measurements based on pumping tests for the GSA would be applicable to the R-Area aquifer units, since the hydrostratigraphic units are fairly well correlated. These are provided in Table 3-18.

3.2.10.9.2.3 Plume Maps for R-Area

Figure 3-76 shows the location of known groundwater plumes in R-Area. The largest known plume extends to the east of the R Reactor area, and is associated with more than one source. Tritium is commingled with TCE and PCE in the East Plume (WSRC 2006b).

Strontium-90 is the contaminant of primary concern associated with the seepage basin plumes, but investigations have shown that the groundwater contamination has not extended significantly beyond the boundaries of the seepage basin area (Brewer and Fogle 2004). The lateral and vertical extent of the plumes associated with the seepage basins are shown in Figure 3-78.

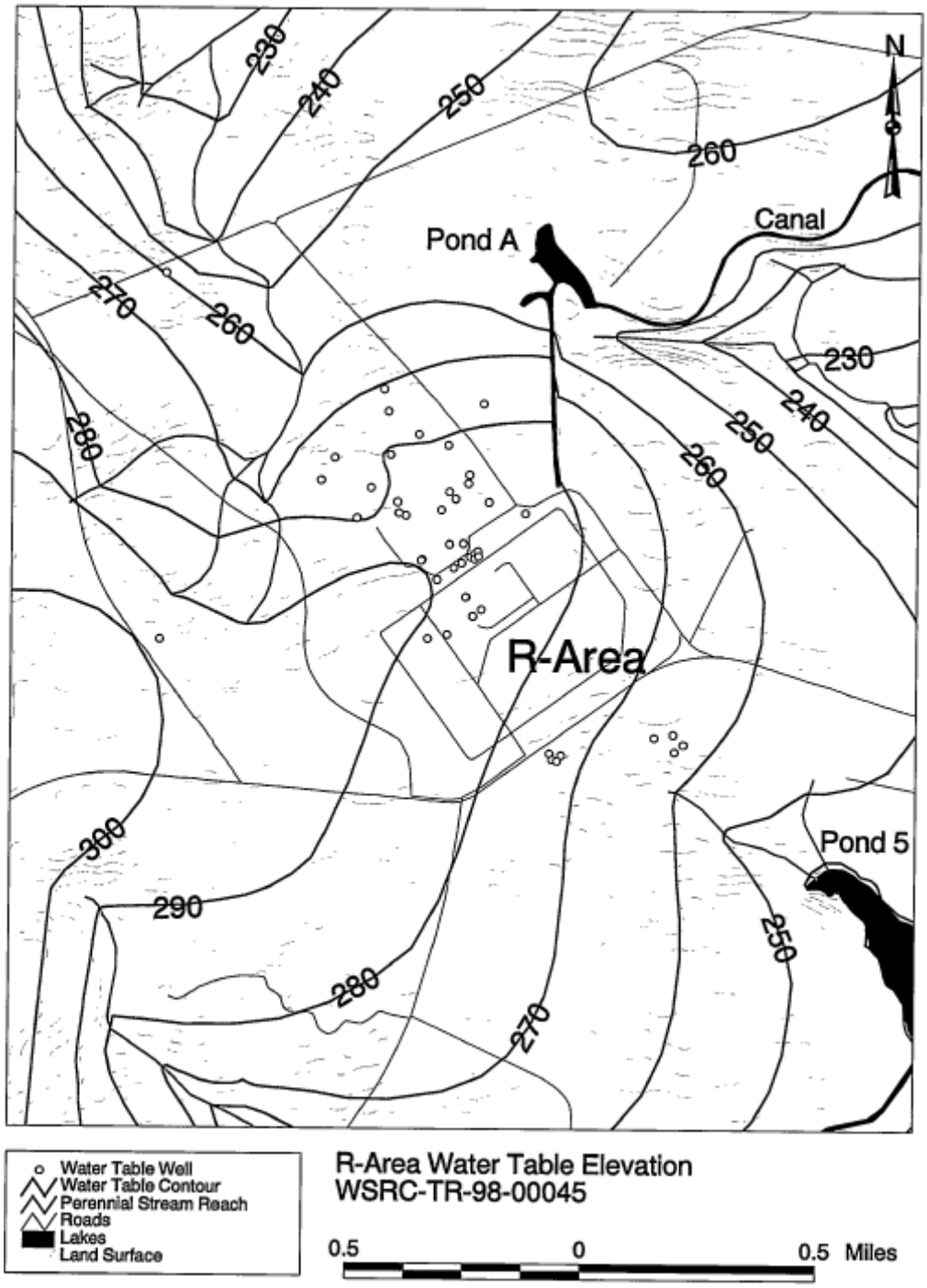


Figure 3-77. Potentiometric Surface of the UTR Aquifer Unit (Water Table Aquifer) for R-Area with Elevations in ft above msl (from Hiergesell 1998)

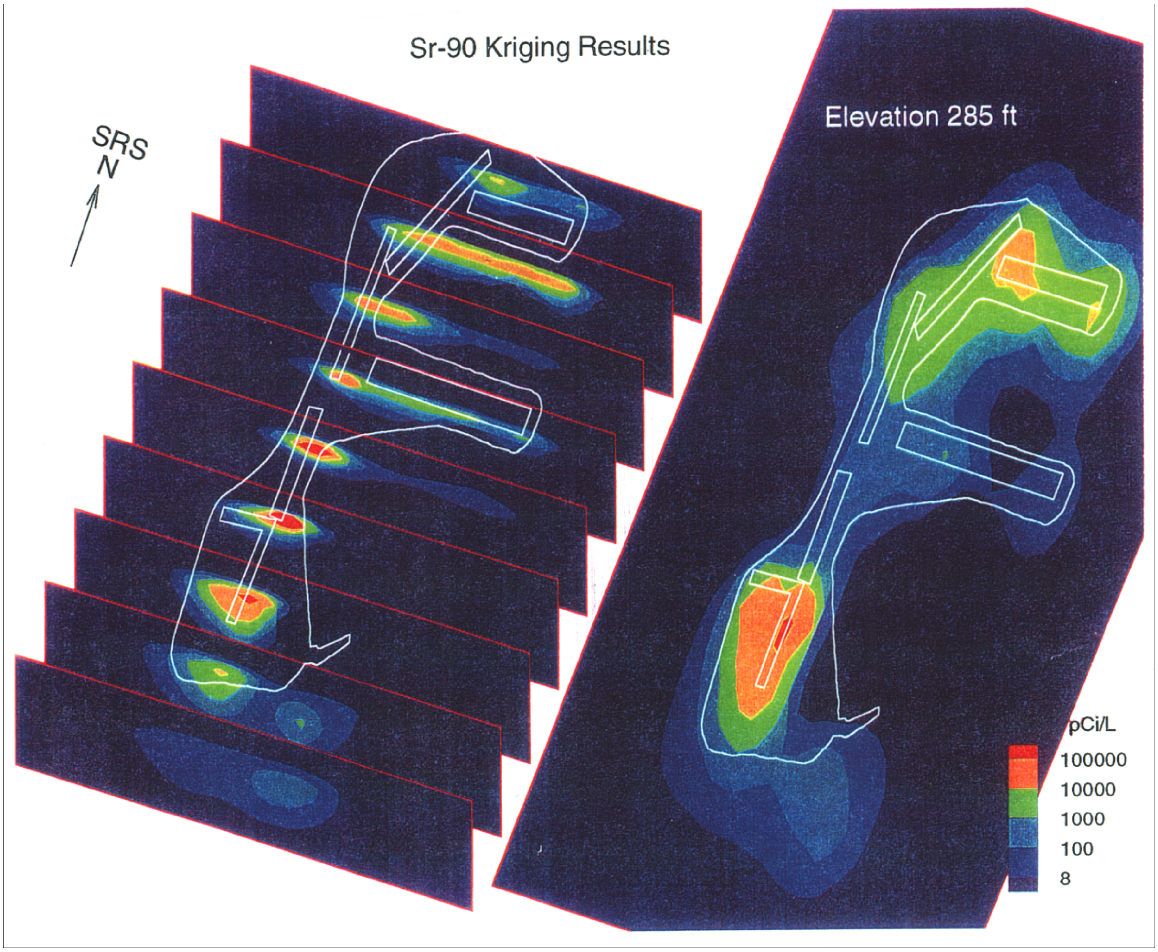


Figure 3-78. Lateral and Vertical Extent of ^{90}Sr under the R-Area Reactor Seepage Basins (from WSRC 2002c)

3.2.10.10 N-Area

N-Area (also known as Central Shops) is located in the central portion of the SRS (Figure 3-1). Portions of N-Area lie within the Fourmile Branch watershed, while others lie within the Pen Branch watershed (WSRC 2003c, WSRC 2000b). Ground surface elevations range from 91 to 96 m above msl. This area of SRS is on a nearly flat interfluvial divide approximately equidistant from the Pen Branch stream valley to the southeast and the Fourmile Branch stream valley to the northwest.

Under the Heavy Equipment Wash Basin and Central Shops Burning Rubble Pit operable Unit (under RCRA) in N-Area, the water table flow direction is generally in a northwesterly direction (WSRC 2003c). However, under the Ford Building Seepage Basin Operable Unit (also in N-Area), water table flow is toward the southwest (WSRC 2000b). These divergent flow regimes are reflective of the groundwater divide present in N-Area. Flow in the underlying Gordon aquifer is not significantly affected by Fourmile Branch or Pen Branch, and thus can be expected to be in a west-to-southwest direction (Figure 3-36).

Figure 3-79 and Figure 3-80 (DOE 2005) show potential migration pathways from groundwater to either Fourmile Branch (Figure 3-79), or Pen Branch (Figure 3-80), in greater detail than previously shown in Figure 3-40.

3.2.10.10.1 Local Hydrostratigraphy in N-Area

Due to the proximity of N-Area to C-Area, it is reasonable to expect that hydrostratigraphy at N-Area would be similar to that at C-Area. At C-Area, the water table is approximately 21.3 m below the ground surface, and lies in the “upper” aquifer zone of the UTR aquifer unit. The UTR aquifer unit is approximately 30 m thick in this region and is divided into three zones: two aquifer zones and a tan clay confining zone. The "upper" aquifer zone consists of all saturated strata that lie between the water table and the "tan clay," which is a discontinuous confining zone. The “lower aquifer" zone of the UTR aquifer occurs between the “tan clay" confining zone and the Gordon confining unit (WSRC 2002b).

The Gordon confining unit is from approximately 3.0 m thick near the C-Area (measured about 460 m east of C Reactor), where there is a drop in hydraulic head of about 17 m across the confining unit. The Gordon aquifer is approximately 20 m thick in this area, and thickens to the east and southeast.

The Crouch Branch confining unit, which underlies the Gordon Aquifer, is about 53 m thick in the C/N-Area vicinity. The hydraulic gradient between the Crouch Branch aquifer and the overlying Gordon aquifer is upward (WSRC 2002b), thereby hydraulically isolating the lower aquifers and providing an effective barrier to significant contaminant migration into these aquifers.

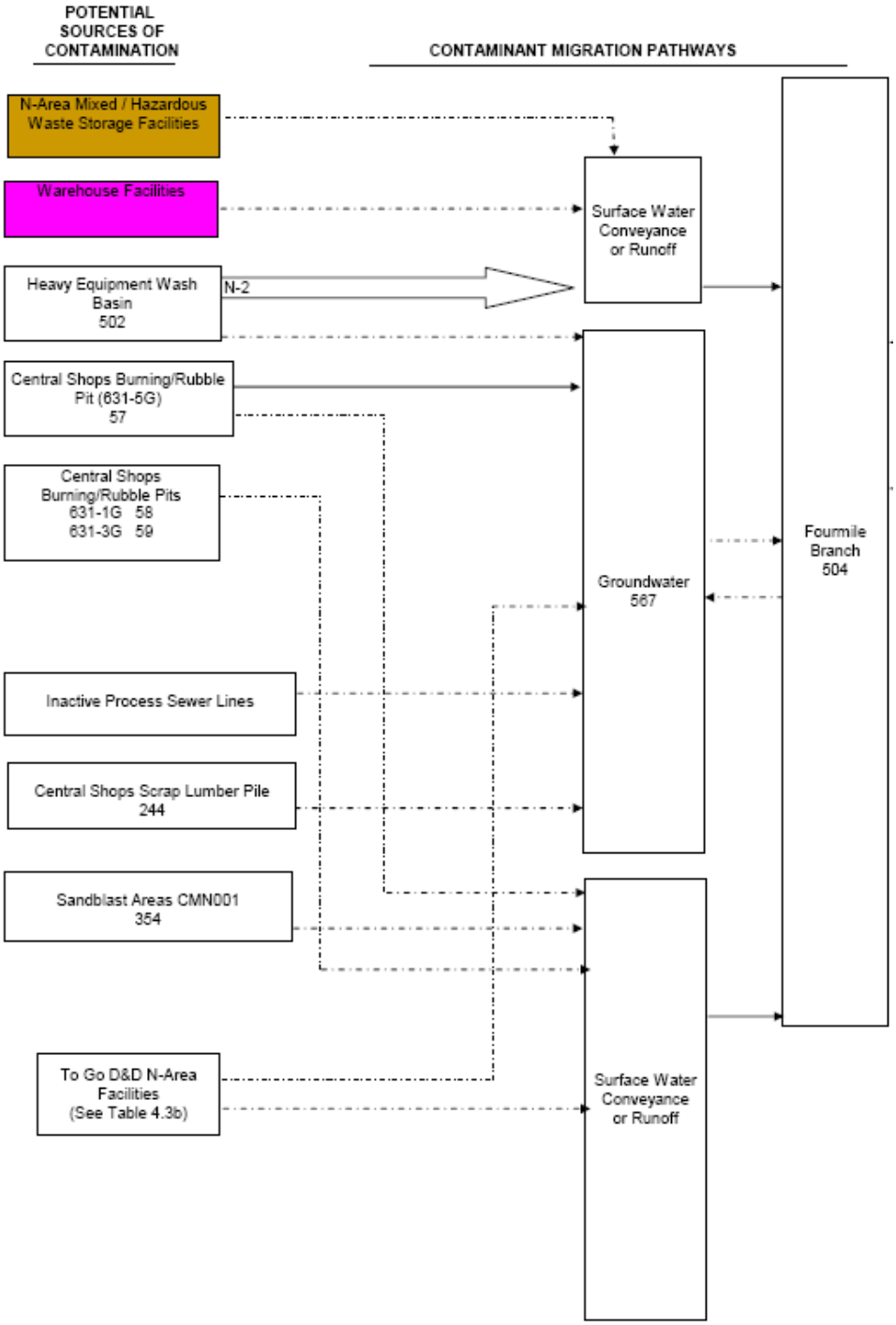


Figure 3-79. Migration Pathways for N-Area Contaminants in the Fourmile Branch Watershed (DOE 2005)

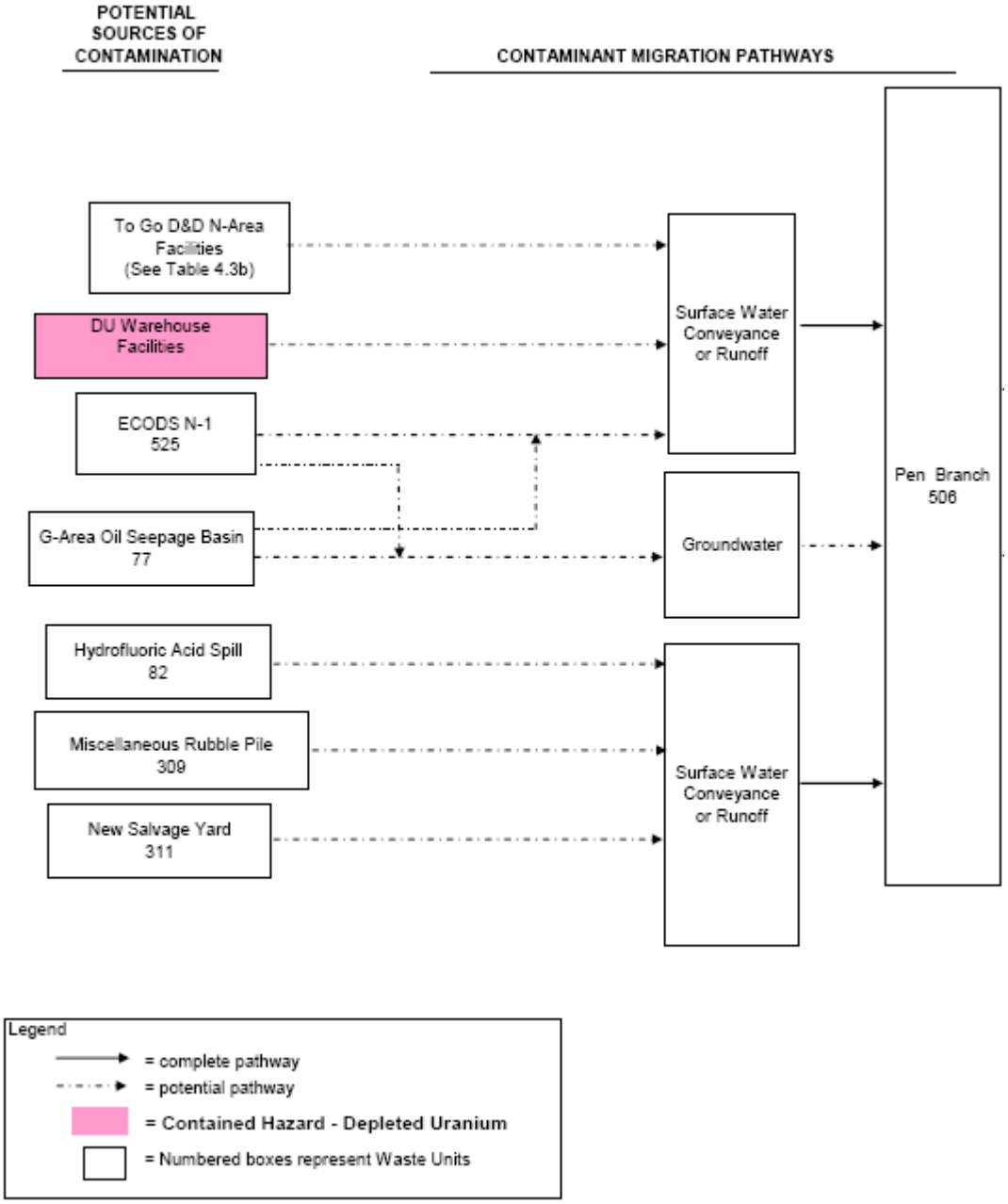


Figure 3-80. Migration Pathways for N-Area Contaminants in the Pen Branch Watershed (DOE 2005)

3.2.10.10.2 Characterization Data for N-Area

Several groundwater monitoring wells are screened in the water table aquifer in N-Area, for the purpose of monitoring contaminants and measuring water levels. Groundwater contamination in N-Area is associated with organic compounds, including TCE, as well as heavy metals, and has only been detected at the SRL oil test site, and at the heavy-equipment wash basin and Central Shops burning/rubble pit (Arnett and Mamatey 2002).

3.2.10.10.2.1 Potentiometric Surface Map for N-Area

Figure 3-62 shows the potentiometric surface of the water table unit under N-Area (called Central Shops in the figure). The potentiometric surface shown is based on long-term mean elevations measured for the UTR aquifer unit. The general configuration shows an east-to-west trending divide, and partitioning of groundwater flow north to Fourmile Branch, and south to Pen Branch. The effect of the Fourmile Branch tributary on the water table elevation contours can be seen in the areas northwest of N-Area (Hiergesell 1998).

3.2.10.10.2.2 Hydraulic Conductivities in N-Area

It is reasonable to assume that hydraulic conductivity measurements based on pumping tests for the GSA would be applicable to the N-Area aquifer units, since the hydrostratigraphic units are fairly well correlated. These are provided in Table 3-18.

3.2.10.10.2.3 Groundwater Plumes in N-Area

It is known that at the SRL oil test site, where the natural biodegradation of petroleum waste or waste oil was evaluated, organic contaminants have been detected in both the upper and lower zones of the underlying UTR aquifer unit. In the vicinity of the heavy-equipment wash basin and Central Shops burning/rubble pits outfall, a TCE plume exists in two “lobes” – one moves downgradient on top of a clay layer in the UTR aquifer; the other has been drawn deeper into the lower zone of this aquifer (Arnett and Mamatey 2002).

3.3 NATURAL RESOURCES

Of the approximately 198,000 acres the SRS occupies in west central South Carolina, about 16,000 acres are used for industrial missions, including the cooling ponds, about 14,000 acres are reserved for nonmanipulative, ecological research, and the remaining approximately 168,000 acres are managed for a variety of natural resources (USDA 2005). The management programs have been, and are, directed toward meeting conservation and restoration objectives (e.g., protecting threatened and endangered species or water resources), providing research and educational opportunities, and generating revenue from the sale of forest products (Blake et al. 2005a). This section focuses on the surface water and groundwater resources on the SRS, and briefly describes some of the other site natural resources.

3.3.1 Water Resources

The South Carolina Department of Health and Environmental Control (SCDHEC) has been delegated authority by the EPA to implement and enforce the requirements of the Clean Water Act for the State of South Carolina. The SCDHEC therefore is responsible for maintaining the chemical and biological integrity of all state waters, including those on federal reservations such as SRS. It does this by enforcing a system of water quality standards and by regulating all point-source discharges through the NPDES program. The SCDHEC is the principal regulatory authority for water quality issues on the SRS.

3.3.1.1 Surface Water

Over 20 percent of the SRS consists of wetlands and aquatic systems, and except for the far northeastern corner (1.3 percent of the SRS) that drains to the Salkehatchie River, all SRS surface waters drain to the Savannah River. The SRS makes up about 3 percent of the Savannah River watershed (Kolka et al. 2005).

Five major tributaries of the Savannah River (UTR, Fourmile Branch, Pen Branch, Steel Creek, and LTR) flow through and drain the SRS (Figure 3-11). The Savannah River serves as an industrial water source for the SRS and both a domestic and an industrial water source for several downstream communities (the cities of Port Wentworth and Savannah in Georgia and Hilton Head Island and Beaufort and Jasper counties in South Carolina). The intakes for these downstream water systems are located at river miles 29 and 39.2, respectively. In addition, the VEGP (see Section 3.2.3.2), located across the river from the SRS, uses the Savannah River for cooling water, withdrawing an average of 92 cfs and returning 25 cfs (CSRACT 2007).

The SRS monitors non-radioactive liquid discharges to surface waters through the NPDES, having NPDES permits in place for discharges to the waters of the United States and South Carolina. These permits establish the specific sites to be monitored, parameters to be tested, and monitoring frequency, as well as analytical, reporting, and collection methods. Continuous surveillance monitoring of site streams occurs downstream of several process areas to detect and quantify levels of radioactivity in effluents transported to the Savannah River (Mamatey 2007).

The SRS streams and the Savannah River are classified by SCDHEC as "Freshwaters," which are defined as surface water suitable for (Mamatey 2007):

- Primary and secondary contact recreation and as a drinking water source after treatment in accordance with SCDHEC requirements
- Fishing and survival and propagation of a balanced indigenous aquatic community of fauna and flora
- Industrial and agricultural uses

Table 3-21 provides characteristics of Savannah River water quality both upstream and downstream of the SRS. Table 3-22 provides characteristics related to water quality in SRS streams of interest downstream of SRS. A detailed discussion of water quality in both the Savannah River in the vicinity of the SRS and in SRS streams can be found in Wike et al. (2006).

Table 3-21. Water Quality in the Savannah River Upstream and Downstream from SRS (Calendar Year 2006)

Parameter	Unit of Measure[a]	MCL[b, c] or DCG[d]	Upstream		Downstream	
			Minimum[e]	Maximum[e]	Minimum[e]	Maximum[e]
Aluminum	mg/L	0.05-0.2[f]	0.056	0.32	0.10	0.43
Cadmium	mg/L	0.005[b, c]	0.0011	0.0011	0.0011	0.0017
Chromium	mg/L	0.1 [b, c]	Non-Detectable (ND)	ND	ND	ND
Copper	mg/L	1.3 [b, g]	ND	ND	ND	ND
Dissolved Oxygen	mg/L	>5.0[h]	6.9	11.6	5.65	11.6
Gross Alpha Radioactivity	pCi/L	15[b]	ND	7.35	ND	1.05
Lead	mg/L	0.015[g]	ND	ND	0.0041	0.0041
Mercury	mg/L	0.002[b, c]	0.00002	0.000026	0.00002	0.000025
Nickel	mg/L	Not Applicable (N/A)	0.0029	0.0029	ND	ND [h]
Nitrate (as N)	mg/L	10[b, c]	0.032	0.31	0.24	0.38
pH	pH units	6.5-8.5[h]	5.9	7.1	6.3	7.6
Phosphate	mg/L	N/A	0.084	0.22	0.078	2.5
Suspended solids	mg/L	N/A	2	10	4.4	14
Temperature	°F	90[h, i]	48.2	77.7	50.4	79.7
Tritium	pCi/L	20,000[b, c]	ND	335	185	1,990
Zinc	mg/L	5[f]	0.0032	0.015	0.0032	0.0049

Notes: Information extracted from WSRC-TR-2007-00008 and accompanying data files. Parameters are those DOE routinely measures as a regulatory requirement or as part of ongoing monitoring programs. Less than (<) indicates concentration below lower limit of detection.

[a] mg/L = milligrams per liter (L); pCi/L = picocuries per L

[b] Maximum Contaminant Level (MCL), EPA National Primary Drinking Water Standards [40 CFR 141].

[c] MCL, SCDHEC State Primary Drinking Water Regulation [SCDHEC R.61.58].

[d] DOE Derived Concentration Guide (DCG) for water, based on committed effective dose of 100 mrem/yr. for consistency with drinking water MCL of 4 mrem/yr [DOE O 5400.5_Chg 2].

[e] Minimum concentrations of samples. The maximum listed concentration is the highest single result found during one sampling event.

[f] Secondary MCL. EPA National Secondary Drinking Water Regulations [40 CFR 143].

[g] EPA action level for lead and copper [40CFR141].

[h] SCDHEC water quality standard [SCDHEC R.61.68].

[i] Shall not exceed weekly average of 90°F after mixing nor rise more than 5°F in 1 week unless appropriate temperature criterion mixing zone has been established.

Table 3-22. Water Quality in Selected SRS Streams (Calendar Year 2006)

	Temperature (^oF)	pH	Dissolved Oxygen (mg/L)	Total Suspended Solids (mg/L)
Sampling Location: Fourmile Branch (Downstream)				
Mean	65	7.1	9.4	3.1
Range	50 - 79	6.6 – 7.4	5.8 – 14	0.5 – 7.2
Sampling Location: Upper Three Runs (Downstream)				
Mean	64	6.7	8.5	7.1
Range	52 - 77	5.7 – 7.4	5.9 – 12	0 – 22

Note: All data extracted from Mamatey (2007) and accompanying data files.

3.3.1.2 Groundwater

Groundwater is an important source of flow in riparian zones, streams, and wetlands of the SRS (Kolka et al. 2005). Within 20 mi of the SRS, there are more than 56 major municipal, industrial, or agricultural groundwater users that consume approximately 36 million gallons of water per day (DOE 1997b). Total SRS groundwater (domestic and process water) use has dropped from about 11 million gallons per day during 1983 – 1986 to 2.7 million gallons per day in 2008 (Mamatey 2009); pumping occurs mainly in A/M Area, F Area, and H Area (Clarke and West 1997). At the SRS, only the deeper aquifers (Crouch Branch and McQueen Branch, Figure 3-13) are used as groundwater sources, providing a more abundant supply of higher quality water (Kolka et al. 2005).

Under most of the SRS, the quality of groundwater is considered to be good. The pH for SRS groundwater ranges from 4.9 to 7.7 and the water is generally soft (DOE 1997b). Concentrations of dissolved and suspended solids are low, but iron concentrations are elevated in some of the aquifers (DOE 1995). At the SRS, approximately 5 to 10 percent of the shallow aquifer system has been contaminated with tritium, industrial solvents, and other chemicals (Arnett et al. 1993). Table 3-23 indicates locations of areal and local groundwater contamination, and the major contaminants in each location, at the SRS (Mamatey 2007). The groundwater represented in Table 3-23 is not being consumed and active remediation projects are in progress to address the groundwater conditions.

The Federal Safe Drinking Water Act (SDWA) was enacted in 1974 to protect public drinking water supplies. SRS domestic water is supplied by 17 separate systems, all of which utilize groundwater sources. The A-Area, D-Area, and K-Area systems are actively regulated by SCDHEC, while the remaining 14 site water systems receive a reduced level of regulatory oversight (Mamatey 2007).

Table 3-23. Summary of Maximum Groundwater Monitoring Results for Major Areas within the SRS, 2005–2006 (from Mamatey 2007)

Location	Major Contaminants	Units	2005 Maximum	MCL[a]	2004 Maximum	Likely Outcrop Point
A-Area/M-Area	TCE PCE	ppb ppb	28,400 127,000	5 5	37,300 142,000	Tims Branch/Upper Three Runs Creek in East; Crackerneck Swamp in West
C-Area	TCE Tritium	ppb pCi/L	1,611 4,851,000	5 20,000	1,611 6,230,000	Tributaries of Fourmile Branch
D-Area	TCE Tritium	ppb pCi/L	490 1,030,000	5 20,000	395 1,030,000	Savannah River Swamp
E-Area	Tritium TCE	pCi/L ppb	45,700,000 570	20,000 5	82,300,000 606	Upper Three Runs/Crouch Branch in North; Fourmile Branch in South
F-Area	TCE Tritium Gross alpha Beta	ppb pCi/L pCi/L pCi/L	55 435,000 103 359	5 20,000 15 4 mrem /yr[b]	25.3 1,190,000 89.4 283	Upper Three Runs/Crouch Branch in North; Fourmile Branch in South
F-Area Seepage Basins	Tritium Gross alpha Beta	pCi/L pCi/L pCi/L	7,660,000 781 3,030	20,000 15 4 mrem /yr[b]	8,599,078 784 3,280	Fourmile Branch
H-Area	Tritium TCE Gross alpha Beta	pCi/L ppb pCi/L pCi/L	54,300 undetected 8.46 81.5	20,000 5 15 4 mrem /yr[b]	93,048 undetected 24.1 146	Upper Three Runs/Crouch Branch in North; Fourmile Branch in South
H-Area Seepage Basins	Tritium Gross alpha Beta	pCi/L pCi/L pCi/L	6,710,000 89 2,630	20,000 15 4 mrem /yr[b]	9,250,000 239 1,540	Fourmile Branch
R-Area	Tritium	pCi/L	111,000	20,000	164,000	Mill Creek in Northwest; tributaries of PAR Pond elsewhere
K-Area	Tritium TCE	pCi/L ppb	26,900,000 17	20,000 5	1,560,971 12.7	Indian Graves Branch
L-Area	Tritium TCE	pCi/L ppb	1,250,000 21	20,000 5	1,250,000 13	L Lake
P-Area	Tritium TCE	pCi/L ppb	18,400,000 13,600	20,000 5	523,000 8,460	Steel Creek in North; Meyer's Branch in South
Sanitary Landfill	TCE Vinyl chloride	ppb ppb	16 30	5 2	14.1 28.4	Upper Three Runs Creek
TNX	TCE	ppb	566	5	680	Savannah River Swamp
CMP Pits	TCE	ppb	1,090	5	713	Pen Branch

[a]MCL= Maximum contaminant level

[b]The activity (pCi/L) equivalent to 4 mrem/yr varies according to which specific beta emitters are present in the sample.

3.3.1.3 Other Resources

About two-thirds of the SRS are managed for the protection of the endangered red-cockaded woodpecker, and for the associated fire-maintained savanna conditions that support a great diversity of species. The species requires large pine trees and an open understory, maintained primarily through prescribed burning. Many of the forest management practices at the SRS are geared toward providing nesting and foraging habitat for this species (Kilgo and Blake 2005).

The Crackerneck Wildlife Management Area and Ecological Reserve, encompassing about 11,000 acres of wetland and mesic land, with predominantly pine forest, bottomland hardwood, and cypress-tupelo swamp habitats, is managed primarily as wildlife habitat to enhance recreational hunting, fishing, and nonconsumptive use. The South Carolina Department of Natural Resources, in conjunction with the DOE, manages this area, and no industrial facilities exist within it (Blake et al. 2005a).

The Savannah River swamp and LTR corridor are natural resources that are managed to protect the wetlands represented, limit access, and minimize sediment movement. Restoration programs constitute many of the management activities (Blake et al. 2005a).

Approximately 14,000 acres are part of the DOE Set-Aside Program, of which activities are restricted to nonmanipulative research and monitoring. These areas include unique ecological areas such as Carolina bays and major stream systems, as well as old experimental sites (Blake et al. 2005a).

Forest products harvesting and silviculture management are conducted to meet security needs, encourage habitat development for threatened and endangered species, to promote forest health, and generate revenue from the sale of forest products. Since establishment of the SRS, the standing volume of pine and hardwood has increased dramatically (USDA 2005). The SRS sells both softwood and hardwood trees, as well as pine straw, using revenues to support mainly restoration and resource management activities (Blake and Bonar 2005). Over the last 10 years, revenue from the sale of forest products has averaged $5.3 million dollars per year (USDA 2005).

One material of significance as a geologic resource in the vicinity of the SRS is kaolin clay. About 90 percent of the U.S. production of kaolin at one time came from a district in Georgia and South Carolina that includes Aiken County. Commercial deposits occur as lenses in the Lang Syne Formation along the Fall Line bordering the northwestern edge of the Coastal Plain (Bates 1969). The Lang Syne Formation is at considerable depth beneath much of the SRS, occurring at greater than 100 m from the ground surface in the central area of the site, making commercial exploration unlikely due to the large amount of overburden that would have to be removed to exploit a deposit. Furthermore, land-use conditions make it unviable. The SRS area has also been considered a potential source of the mineral monazite, although there are no known economically viable mineral deposits on the SRS (Denham 2005).

3.3.2 Natural Background and Anthropogenic Sources of Radiation

All human beings are exposed to sources of ionizing radiation that include naturally occurring and man-made sources. The average dose contribution estimates from various sources to individuals were obtained from the NCRP Report No. 160, *Ionizing Radiation Exposure of the Population of the United States* (NCRP 2009). On average, a person living in the U.S., as well as the CSRA, receives an annual radiation dose of 620 mrem (6.2 mSv). The average dose contributions from the various radiation sources to individuals in the U.S. are shown in Figure 3-81.

The major sources of radiation exposure to an average member of the public shown in Figure 3-81 are reasonable estimates for the CSRA and include:

- **Ubiquitous background exposure.** Often referred to as natural background radiation, this accounts for 50% (311 mrem/yr) of the total. Natural sources of radiation exposure include 1) internal inhalation of radon and thoron (228 mrem/yr), 2) external exposure to cosmic sources of radiation (33 mrem/yr), 3) internal ingestion (29 mrem/yr), and 4) external exposure to terrestrial radiation (21 mrem/yr). The actual amount of dose any one individual receives depends largely on their geographical location.
- **Medical exposure.** This includes diagnostic and therapeutic exposures and accounts for about 48% (300 mrem/yr) of the total. The four major subcategories of medical exposures are 1) computed tomography (147 mrem/yr), 2) nuclear medicine (77 mrem/yr), 3) interventional fluoroscopy (43 mrem/yr), and 4) conventional radiography and fluoroscopy (33 mrem/yr). These are annual average doses, the actual amount of medical dose received is obviously individual specific.
- **Consumer product exposure.** This accounts for about 2% (13 mrem/yr) of the total and includes 1) cigarette smoking (4.55 mrem/yr), 2) building materials (3.51 mrem/yr), 3) commercial air travel (3.38 mrem/yr), and 4) other man-made products (1.56 mrem/yr).
- **Occupational and industrial exposures.** These account for much less than 1% (0.5 and 0.3 mrem/yr, respectively) of the total annual average dose. The industrial dose includes offsite public exposure to nuclear facilities such as SRS. In 2008, the potential dose to an offsite maximally exposed individual from SRS operations was 0.12 mrem (0.04 mrem from the air pathway and 0.08 mrem from the liquid pathway) (Mamatey 2009).

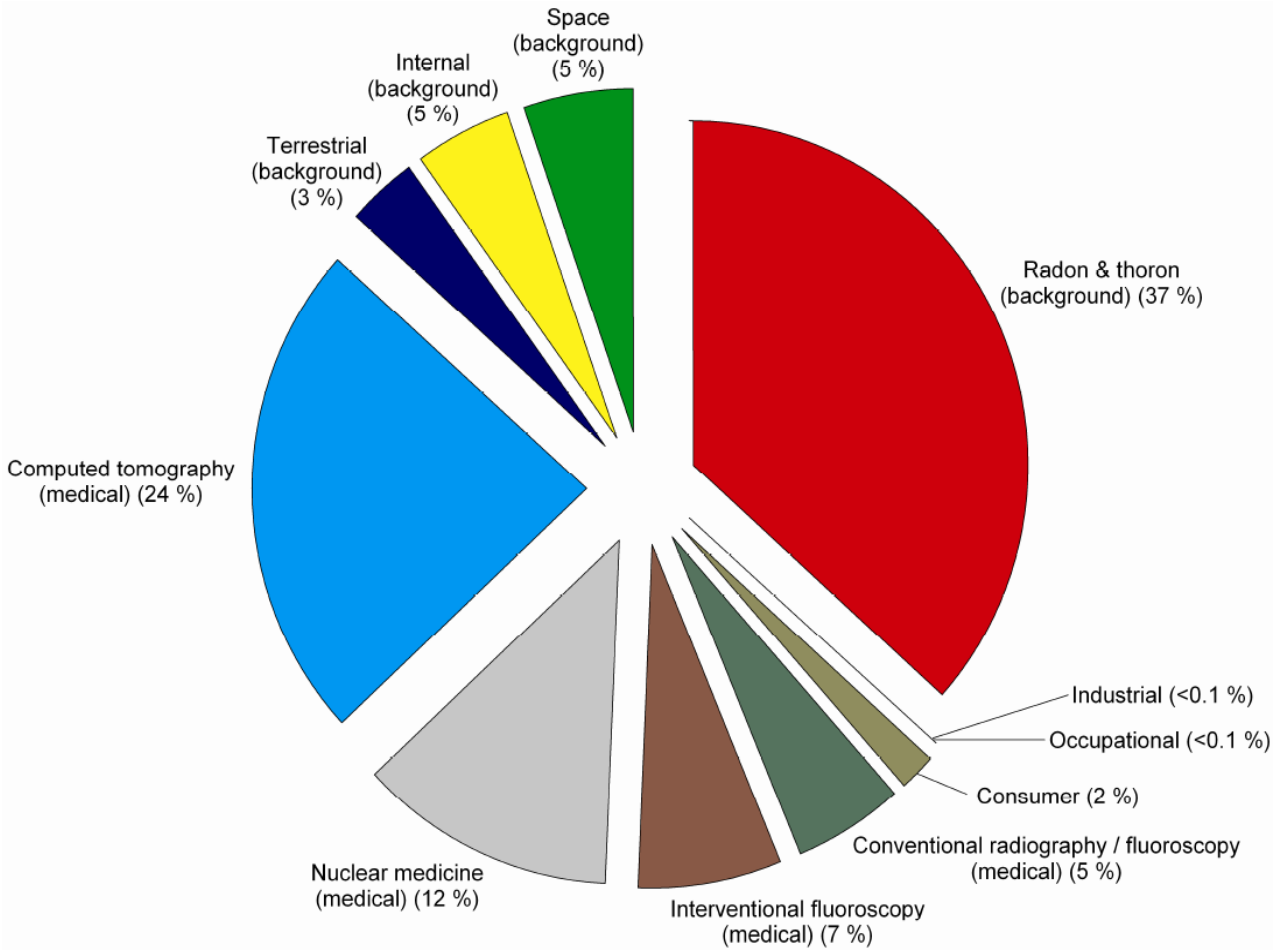


Figure 3-81. Major Sources of Radiation Exposure in the U.S. (from NCRP 2009) (percentages based on an annual average dose of 620 mrem)

3.4 SAVANNAH RIVER SITE PERFORMANCE ASSESSMENTS

3.4.1 E-Area Low-level Waste Facility Performance Assessment

3.4.1.1 General Performance Assessment and Facility Information for the E-Area Low-level Waste Facility

The SRS ELLWF began radiological waste disposal operations in 1994 (WSRC 2008a). A PA was first prepared for the ELLWF in 1994 (WSRC 1994b), in conformance with the DOE order in force at that time for LLW disposal (DOE Order 5820.2A (DOE 1988)). Subsequent to that first PA, DOE issued a new radioactive waste management order (DOE Order 435.1-1 (DOE 1999a)). PA revisions were made in accordance with maintenance requirements of the 1999 Order, in 2000 (McDowell-Boyer et al. 2000) and most recently in 2008 (WSRC 2008a). Discussions presented herein concerning the ELLWF are primarily drawn from the most recent E-Area PA (i.e., WSRC 2008a).

The ELLWF is located in the General Separations Area (GSA), which is in the central region of the SRS (Figure 3-82). It is an elbow-shaped, cleared area, which curves to the northwest, situated immediately north of the MWMF (Figure 3-82). The ELLWF is comprised of 200 acres (100 acres currently allocated for waste disposal, and an adjacent 100 acres potentially available for waste disposal) and a surrounding buffer zone that extends out to the 100-m point of compliance. Disposal units within the footprint of the LLW facilities include the ST, ET, CIG Trenches, the LAWV, the ILV, and the NRCDA (Figure 3-83).

Closure of the ELLWF will be conducted in three phases: operational closure, interim closure, and final closure. Operational closure will be conducted during the 25-year operational period (30-year period for ST and ET) as disposal units are filled. Interim closure will take place following the end of operations and will consist of an area-wide runoff cover along with additional grading over the trench units. Final closure of all disposal units in the ELLWF will take place at the end of the 100-year institutional control period and will consist of waste stabilization for the ST and ET and installation of an integrated closure system designed to minimize moisture contact with the waste and to serve as a deterrent to inadvertent intruders (WSRC 2008a).

3.4.1.2 Types of Disposal Units in the E-Area Low-level Waste Facility

The STs are below-grade earthen disposal units which are approximately 20 ft deep, 20 ft wide, and 656 ft long. Ten to 14 ft of undisturbed soil separates each trench. A set of five ST are grouped together, and eight groupings have currently been sited. Low-level radioactive waste has been placed within all eight units (Figure 3-83). Low-level waste to be disposed in the ST contains, or will contain, soil, debris, rubble, wood, concrete, equipment, and job control waste.

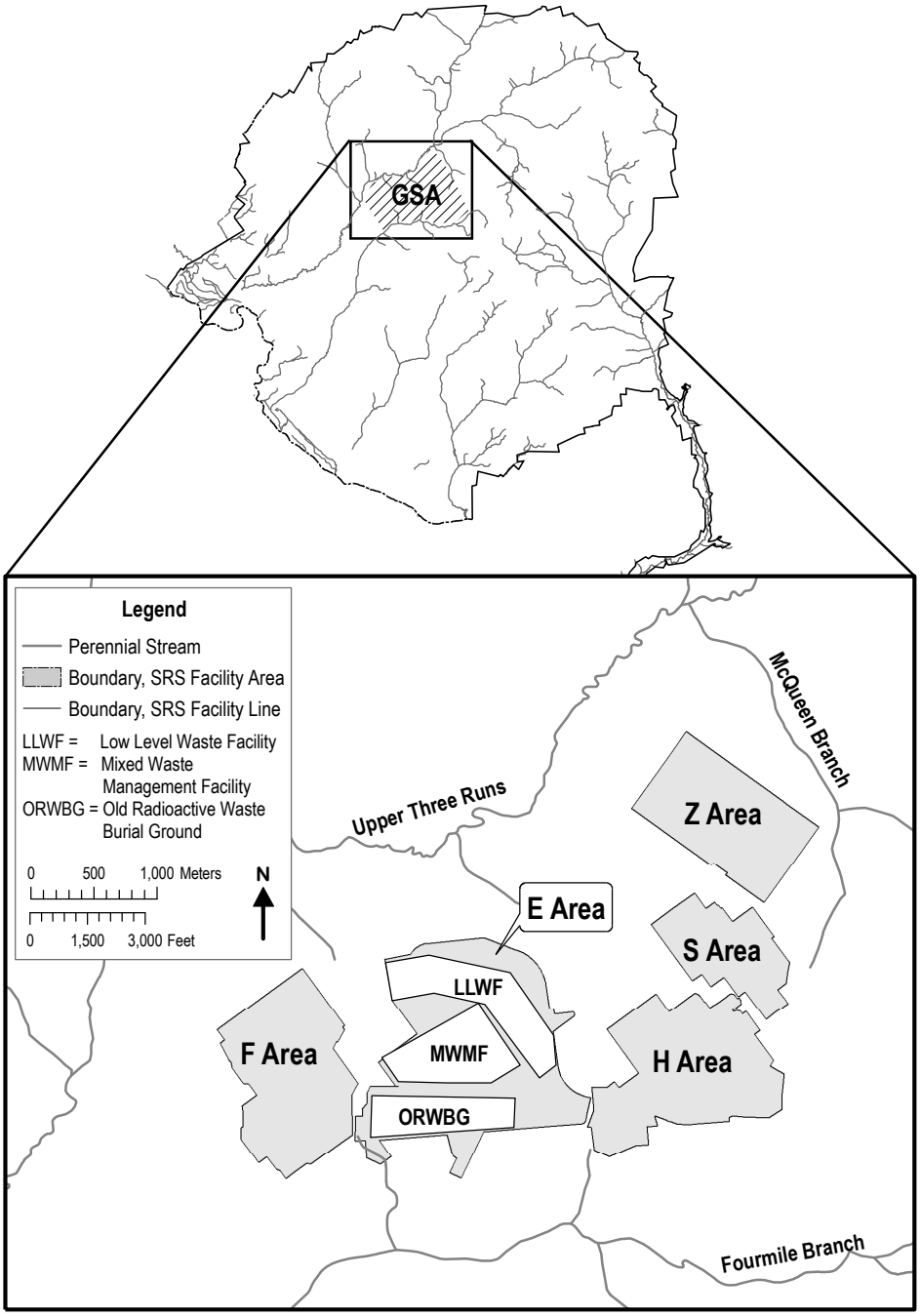


Figure 3-82. Location of the GSA and the ELLWF within the GSA (from WSRC 2008a)

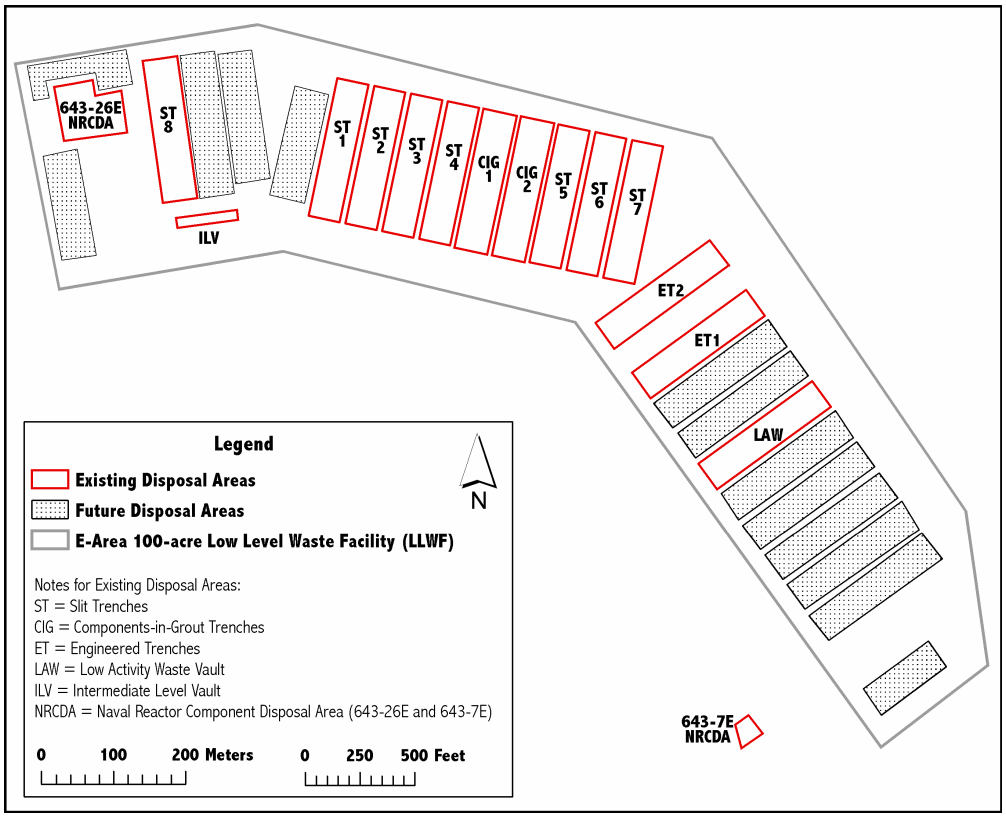


Figure 3-83. Location of Individual Disposal Units within the ELLWF (from WSRC 2008a)

The waste may be disposed as bulk waste, or contained within B-25 boxes, B-12 boxes, 55-gallon drums, SeaLand containers, and other metal containers. Containerized waste and large equipment are typically placed in one end of the trench with a crane, and voids between adjacent containers and the trench wall are filled with either bulk waste or clean soil to minimize water collection prior to closure. Once a section of an ST is filled, clean soil is bulldozed over that section, which is then graded to provide drainage off and away from the disposal operation, decreasing infiltration during the operational period.

The ETs are also below-grade earthen disposal units, which are approximately 650 to 656 feet long by 150 to 160 feet wide. ET #1 varies in depth from 16 to 25 feet, and ET #2 varies in depth from 14 to 23 feet. Low-level waste contained within B-25 boxes, B-12 boxes, 55-gallon drums, SeaLand containers, components, and other metal containers are placed within the trenches. B-25 boxes, the predominant wasteform received, are stacked tightly within the trenches to minimize void space. A berm is constructed around the top where the local terrain slopes towards each trench, to minimize runoff into the trenches during operations.

The bottom of each trench consists of compacted soil, a geotextile filter fabric, and approximately 6 inches of granite crusher run (from bottom to top), sloped to a sump (ET #1) or to a 24-inch steel pipe leading to the sump (ET #2). The sump sides and bottom are covered with concrete. The berm, permeable sloped base, steel pipe, and sump will work to minimize infiltration into the trenches, and movement of contaminants out of the base of the ET during the operational period. ET #1 will be closed in FY 2010; as part of the closure, the sump will also be closed. As a consequence, a new sump will be developed in ET #2.

The CIG disposal units are below-grade earthen trenches and contain grout-encapsulated waste components. Two CIG trench units are anticipated, each divided into five trenches separated by a nominal 10 feet of undisturbed soil. The unit footprint is 656 ft long by 157 ft wide. A typical cross-sectional cut (vertical and perpendicular to the trench length) through a section of a CIG trench is 20 ft by 20 ft. Waste components to be disposed in the CIG trench units consist of large radioactively-contaminated equipment and other smaller wasteforms such as B-25 boxes. Structural stability of the CIG trenches is ensured for a minimum of 300 years after disposal by either filling components with grout or controlled low-strength material (CLSM); or by, for most trench segments, overlaying the contents with 20-inch steel-reinforced concrete mats, such that subsidence is not anticipated during that period of time (Phifer et al. 2006). This engineered aspect of the CIG units delays the increased infiltration that would occur as a result of subsidence. Prior to subsidence of the CIG trenches, the encapsulating grout itself provides a barrier to human intrusion and limits infiltration of water.

The LAW Vault disposal unit is an above-grade reinforced concrete structure, which contains waste components in B-25 boxes, 55-gallon drums and other containers. It is approximately 643 feet long, 145 feet wide, and 27 feet high at the roof crest. The LAW Vault is divided into three modules, each containing four cells. The waste within the containers typically includes job control waste, and contaminated scrap metal, soil and rubble. During the operational period, water entrance into the LAW Vault is minimized through the sub-drainage system, doors on external personnel and forklift openings, the waterproof membrane roofing, and the gutter/downspout system. At the end of the operational period, reinforced concrete will be used to seal exterior vault openings, and those between modules, further minimizing water infiltration until failure of the roof slab, which is estimated to occur at about 2,800 years. Inadvertent intrusion is prevented by the presence of the roof slab prior to structural failure.

The ILV disposal unit is a below-grade reinforced concrete structure, which contains grout-encapsulated waste components. It consists of two modules, which together encompass a 278.83-foot by 48.5-foot area. The existing vault is the only one anticipated to be needed. The Intermediate-Level Tritium (ILT) module, which is 26-foot deep, consists of two cells, one of which (Cell #1) was designed to receive tritium crucibles (tritium crucibles are no longer produced as a waste form). The Intermediate-Level Non-Tritium (ILNT) module, which is 28-foot 5-inches deep, will consist of seven identical cells. Cell # 2 of the ILT and the seven ILNT cells will receive job control waste, scrap hardware, and contaminated soil and rubble, which are contained within metal or concrete containers.

The ILV is used to dispose of waste containers for which radiological dose and radionuclide concentration limits would be exceeded if placed in more cost-effective LLW units (e.g., trenches or the LAW Vault). During the operational period, water entrance into the ILV is minimized through the sub-drainage system, the 30-inch thick waterproofed concrete walls, and the cell rain covers. At the end of the operational period, a permanent reinforced concrete roof slab will be placed on the vault, along with an overlying bonded-in-place fiberboard insulation and waterproof membrane roofing to further limit infiltration until structural failure of the roof, which is estimated to occur at about 6,700 years. Inadvertent intrusion is prevented by the presence of the roof slab prior to structural failure.

The NRCDAs are two separate areas within E-Area which are used for disposal of reactor components from the U.S. Navy. The areas are above-grade gravel pads on which are placed waste shipping/disposal casks containing waste NR components. The 643-7E area, containing 41 casks, is approximately 0.3 acres and is closed to future receipts. It has an interim soil cover in place. The newer 643-26E area is approximately 1.4 acres, and has the capacity to receive 100 casks. Wastes within the casks consist of activated corrosion-resistant metal alloy, and auxiliary equipment primarily contaminated with activated corrosion products (crud) at low levels. The thickness of the steel outer containers varies depending on the wastes contained. For the more highly radioactive components (the activated corrosion-resistant metal alloy), the cask thickness is approximately one foot, with welds at least four inches in thickness. The auxiliary equipment requiring less shielding is contained within thinner-walled casks.

3.4.1.3 E-Area Low-level Waste Facility Performance Assessment Requirements

DOE Order 435.1-1 (DOE 1999a) requires that PAs for LLW disposal facilities assess all-pathway, air, and hypothetical inadvertent intruder doses; radon flux; and impacts to water resources, according to the *Radioactive Waste Management Manual* (DOE M 435.1-1, DOE 1999b). The assessment of radon flux and water resource impacts is for purposes of establishing limits on radionuclides that may be disposed near-surface (DOE 1999b).

All of the PA-required assessments are utilized at the SRS for the purposes of establishing limits on radionuclides that may be disposed in the ELLWF, based upon conformance with the performance measures at the points of compliance outlined in Table 3-24, over a 1,000-yr period after closure of the facility (DOE 1999b). At the SRS, the appropriate measure for protection of water resources has been determined to be the SDWA MCLs (40 CFR Part 141 and WSRC 2008a). The MCLs applicable to radionuclides in drinking water, and thus in groundwater at the SRS, are listed in Table 3-25.

Table 3-24. PA Performance Measures for Low-level Waste Facilities at the SRS, in Accordance with DOE 435.1-1 (DOE 1999b)

Performance Objective or Assessment Requirement	Performance Measure	Point of Compliance
All pathways[a]	≤25 mrem in a year, not including doses from radon and progeny	Point of highest projected dose beyond a 100-m buffer zone surrounding the disposed waste.
Air pathway[a]	≤10 mrem in a year, not including doses from radon and progeny	Point of highest projected dose beyond SRS boundary (during 100-yr institutional control period), and beyond a 100-m buffer zone surrounding the disposed waste. (after 100-yr institutional control period)
Radon[a]	average flux of ≤ 20 pCi/m^2/s	Disposal facility surface
Hypothetical inadvertent intruder[b]	100 mrem in a year from chronic exposure	Disposal facility
	500 mrem from a single event	Disposal facility
Impact on Water Resources[b] (Groundwater)	The SRS interpretation is that concentrations of radioactive contaminants should not exceed SDWA MCLs (40 CFR Part 141).	Point of highest projected dose or concentration in groundwater beyond a 100-m buffer zone surrounding the disposed waste.

[a] listed as a Performance Objective in DOE M 435.1-1, for which it must be shown that a reasonable expectation exists that the objective will be met for waste disposed after September 26, 1988.

[b] listed as an Assessment Requirement of PAs in DOE M 435.1-1 for purposes of establishing limits on radionuclides that may be disposed near-surface in a low-level waste disposal facility.

Table 3-25. Maximum Contaminant Levels (MCLs) for Radionuclides in Drinking Water (from 40 CFR Part 141)

Component	MCL
Beta-Gamma Dose	4 mrem/year
Gross Alpha[a]	15 pCi/L
Radium[b]	5 pCi/L
Uranium	30 µg/L

[a] excluding U and Rn, but including Ra226
[b] Ra226 and Ra228

3.4.1.4 E-Area Low-level Waste Facility Performance Assessment Results

The pathway screening analysis in the ELLWF PA indicated that surface water pathways were insignificant in the assessments which potentially include groundwater pathways (i.e., the all-pathways dose and water resource protection assessments), because the point of compliance occurs in groundwater (100-m downgradient from the disposal facility) before the plume is projected to contact the nearest surface water (WSRC 2008a). Thus, the focus of the ELLWF PA assessment of the all-pathways dose and water resources protection was on potential groundwater contamination.

By comparing the calculated maximum groundwater concentrations, all-pathway and air-pathway doses, intruder doses, and radon fluxes at the respective points of compliance to respective performance measures (Table 3-24), preliminary inventory limits were developed for each disposal unit in the ELLWF PA. Groundwater plume interaction factors were developed to adjust the groundwater-based inventory limits as necessary to account for potential plume overlap for the multiple disposal units present in the facility. Results of sensitivity and uncertainty analyses were also considered in comparing the calculated disposal limits with projected radionuclide inventory for the ELLWF (WSRC 2008a).

The final radionuclide disposal limits were compared with projected radionuclide inventories for the ELLWF disposal units and a sum-of-fractions (SOF) calculation of limits for each disposal unit was derived. A separate SOF is calculated for the air pathway, radon pathway, each relevant intruder scenario, and separate time frames for the groundwater pathway, as appropriate (WSRC 2008a). For the PA-based limits (represented by the performance measures in Table 3-24), the total disposal unit inventory for each radionuclide is divided by its corresponding limit for each of these calculations. A SOF fraction less than one indicates compliance with the limit corresponding to a particular performance measure. The derived SOF limits are summarized in Table 3-26, by reporting only the largest SOF calculated for each unit. None of the SOFs is greater than one; therefore, there is a reasonable expectation that all of the performance measures set forth in DOE M 435.1-1 (see Table 3-24) will continue to be met throughout the life of the ELLWF.

Table 3-26 indicates that the critical pathway (the pathway associated with the maximum SOFs) is the groundwater pathway, since the beta-gamma component of the MCLs is the performance measure associated with the maximum SOF for all units but the NRCDA. The PA results indicate that the radionuclides of most significance when the calculated disposal limits are compared to the ELLWF projected inventory at facility closure are C14, H3, I129, and Tc99 and their special waste forms.

For the NRCDA, the maximum SOF evaluated at a distance 100 m from the disposal area is much less than one, indicating that although the air pathway is the route of exposure potentially most important for that type of disposal unit, this pathway is of minor consequence for the ELLWF when compared to the groundwater pathway for the other units. Furthermore, the 100-m distance is not relevant to the CA, since the point of assessment (POA) for the CA is at the first publicly accessible location where radionuclide-contaminated media originating from the GSA, and thus E-Area, occurs (Phifer and Wilhite 2008). For radionuclides in air originating from the NRCDA, this POA would correspond to the SRS site boundary, which is approximately 6.5 mi from the disposal area, and thus the exposures associated with the air pathway and the NRCDA are considerably less at this distance.

Table 3-26. Maximum Sum of Fractions Using Projected Final Inventories

Disposal Unit[a]	Maximum SOF	Performance Measure	Major Nuclides
ST			
East	9.3E-01	4 mrem/year beta-gamma 12-100 years	I129, H3, C14
Central	9.3E-01	4 mrem/year beta-gamma 12-100 years	Tc99, I129, H3
West	9.5E-01	4 mrem/year beta-gamma 0-12 years	I129, H3
ET	9.6E-01	4 mrem/year beta-gamma 12-100 years	Tc99, C14, I129, H3
CIG Trench	9.6E-01	4 mrem/year beta-gamma 125-1125 years	C14, H3, I129
LAWV	9.2E-01	4 mrem/year beta-gamma	I129, C14
ILV	8.2E-01	4 mrem/year beta-gamma 200-1100 years	I129
NRDCA			
643-7E	2.6E-01	10 mrem/year Air	C14
642-26E	1.3E-01	10 mrem/year Air	C14

[a] ST = Slit Trench, ET = Engineered Trench, CIG = Component-in-Grout, LAWV = Low-Activity Waste Vault, ILV = Intermediate-Level Vault, and NRCDA = Naval Reactor Component Disposal Area

3.4.1.5 E-Area Low-level Waste Facility Performance Assessment Conceptual Models

The PA conceptual model of release of radionuclides from the disposal units in the ELLWF addressed both generic wasteforms (containing radionuclides identified on the basis of a screening analysis) and special wasteforms (containing radionuclides associated with specific waste streams) for the groundwater analyses. The list of radionuclides for which groundwater transport calculations were performed for the ELLWF PA included the following: Am241, Am243, C14, Cf249, Cf251, Cl36, Cm244, Cm245, Cm246, Cm247, Cm248, H3, I129, K40, Mo93, Nb94, Ni59, Np237, Pd107, Pu238, Pu239, Pu240, Pu241, Pu242, Pu244, Ra226, Se79, Sn126, Sr90, Tc99, Th230, Th232, U233, U234, U235, U236, U238, and Zr93.

For radionuclides associated with generic wasteforms, the release mechanisms are assumed to be instantaneous within the waste zone, using distribution (or sorption or partition) coefficient (K_d) values to characterize the availability for subsequent transport in the waste zone. For some of the special wasteform radionuclides, the release is not assumed to be instantaneous, in order to account for the presence of ion-exchange resins in the wasteform and/or rate-limiting processes such as diffusion, dissolution, and/or corrosion. The minimum depth from the base of the disposal units to the water table is assumed to range from 25 to 54 ft, based on long-term average water table data and average disposal unit elevations (WSRC 2008a).

The conceptual model of groundwater flow and transport for the ELLWF PA was developed based on the evaluation of water elevation data from the well network in the E-Area vicinity, as well as other information derived from the extensive hydrogeologic characterization data collected in this area. The aquifers considered potentially contaminated by radionuclides leaching from the ELLWF include the UTR aquifer unit, in which the water table resides, and the underlying Gordon aquifer. Contamination will not enter the deep Crouch Branch aquifer in this area because of the upward gradient that exists between this lower aquifer and the overlying Gordon aquifer near UTR stream (Figure 3-37 and WSRC 2008a).

Figure 3-84 shows the potentiometric surface for the water table (UTR) aquifer in the GSA, based on long-term head data, the location of the groundwater divide in this area, and the ELLWF disposal units. Based on the water table head contours and the location of the groundwater divide and creeks relative to the disposal units, the conceptual model of the saturated zone for the ELLWF PA assumes that flow in the water table unit under the ELLWF is northward toward UTR (UTR). The potentiometric surface for the Gordon aquifer unit (Figure 3-45) indicates a north-northwest flow direction under the entire GSA, due to the influence of the incisement of UTR in this aquifer (WSRC 2008a). Thus, radioactive contaminants in all aquifer units of concern with respect to releases from the ELLWF will eventually be discharged to UTR, unless they decay to stable species in transit.

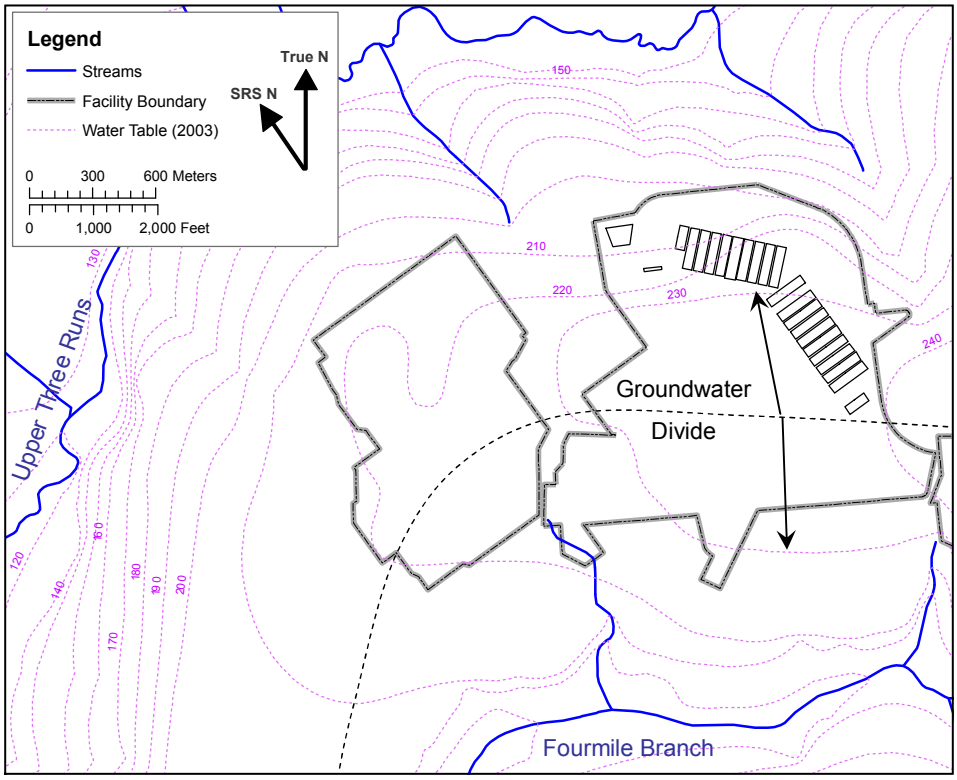


Figure 3-84. Water Table Elevation (in ft above msl) Contours in the ELLWF Vicinity

3.4.2 F-Tank Farm Performance Assessment

3.4.2.1 General Performance Assessment and Facility Information for the F-Area Tank Farm

A PA in support of the eventual closure of the FTF underground radioactive waste tanks and ancillary equipment was completed in 2008 (WSRC 2008b). This PA provides the technical basis used to demonstrate compliance with both the Ronald W. Reagan National Defense Authorization Act (NDAA) for Fiscal Year 2005 Section 3116 criteria (DOD 2004), and the facility closure requirements of DOE Order 435.1-1 (DOE 1999a). Revision 1 of the FTF PA is under development to incorporate comments from the NRC, EPA, and SCDHEC, and it will be issued in FY 2010.

The FTF is located in the central region of the SRS, within the GSA, occupying 22 acres in F-Area (Figure 3-85). The FTF is an active waste storage facility consisting of 22 carbon steel waste tanks, the layout of which is shown in Figure 3-86, which store liquid radioactive waste generated primarily from the F-Canyon PUREX process. The FTF began radioactive operations in 1954. Two of the 22 tanks (Tanks 17 and 20) were closed in 1997 by cleaning and filling with grout.

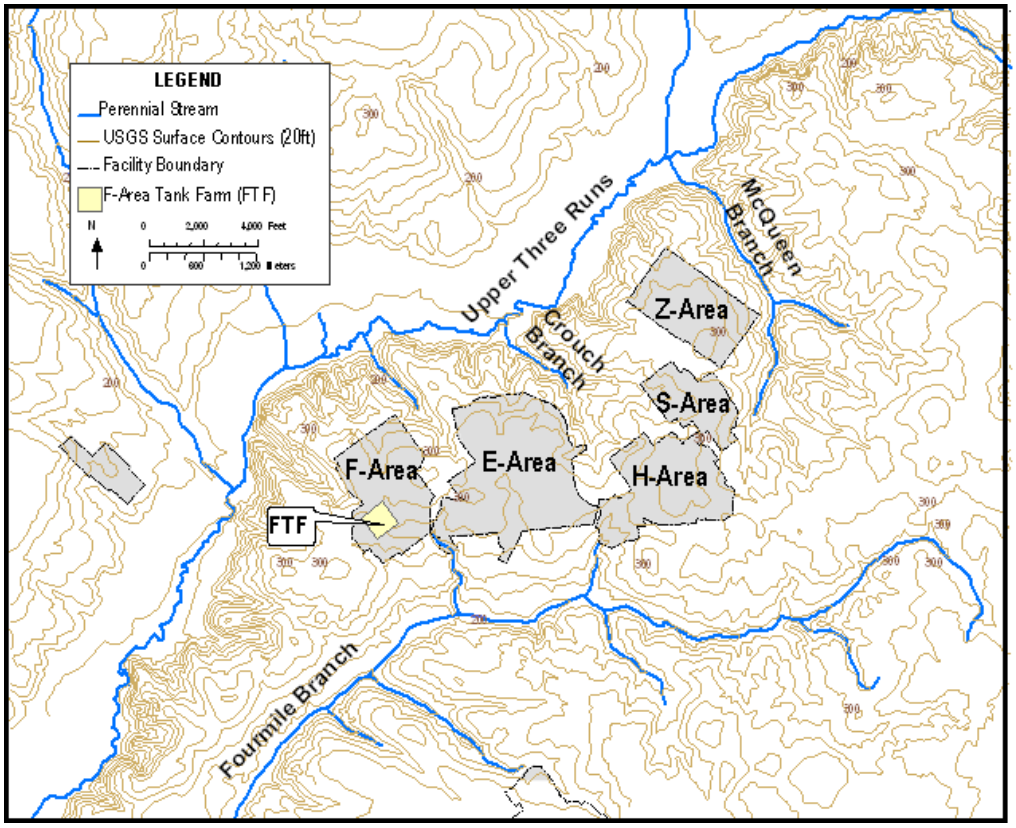


Figure 3-85. Location of the FTF within F-Area (from WSRC 2008b)

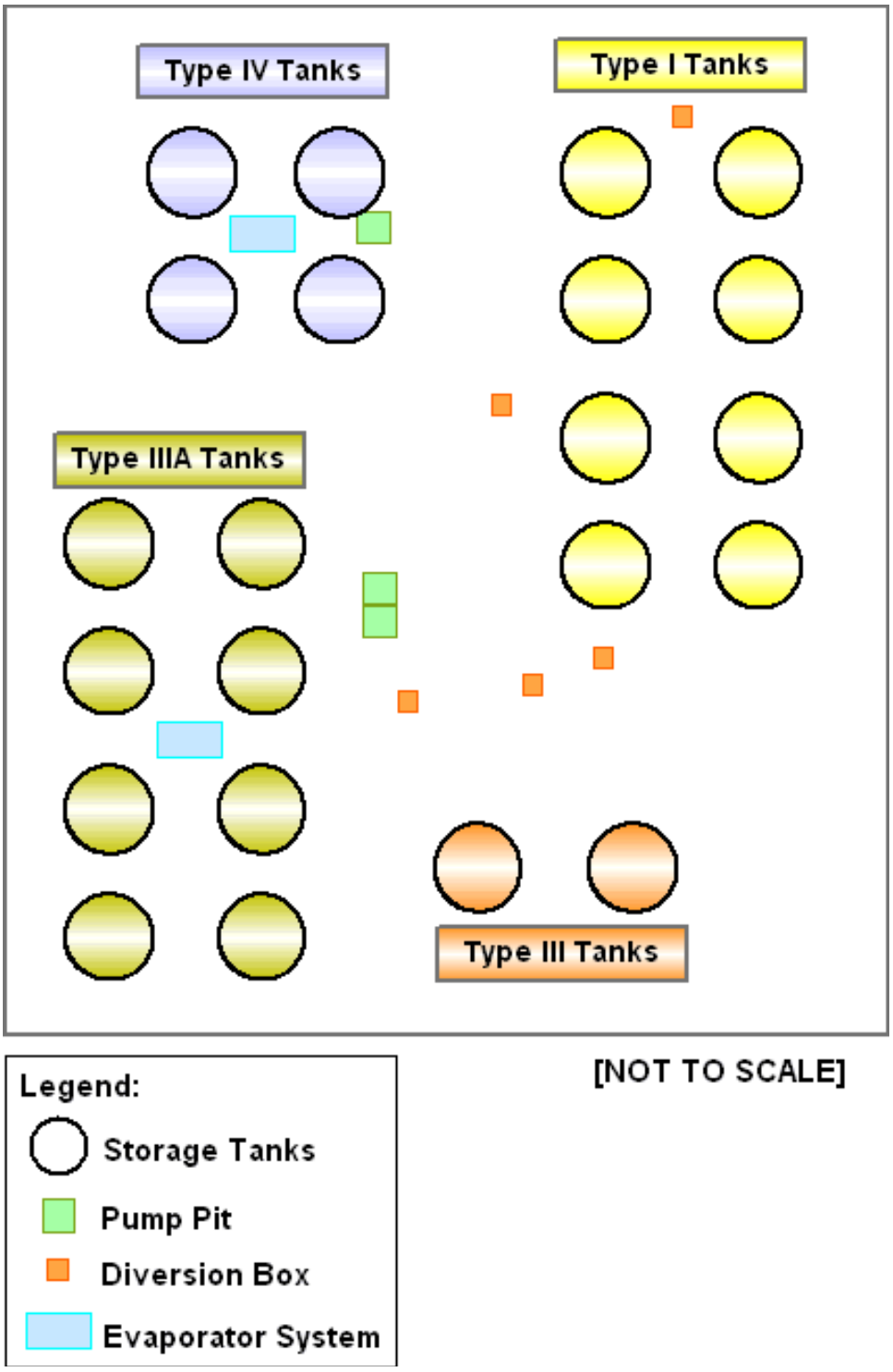


Figure 3-86. Layout of the F-Tank Farm in F-Area (from WSRC 2008b)

The FTF waste tanks were built during four separate construction periods, with a different tank design for each period, leading to the designation of the following four different tank groups (WSRC 2008b):

- Tanks 1 through 8 are Type I tanks and were constructed between 1951 and 1953
- Tanks 17 through 20 are Type IV tanks and were constructed between 1956 and 1958
- Tanks 33 and 34 are Type III tanks and were constructed between 1969 and 1972
- The fourth group of eight tanks, which consists of Tanks 25 through 28 and 44 through 47 are Type IIIA tanks and were constructed in two phases between 1975 and 1978, and 1977 and 1980, respectively

The proposed sequence of events for closure of FTF is as follows:

- Closure of the remaining Type IV tanks, the Type I tanks and finally the Type III/IIIA tanks. The ancillary equipment such as transfer lines, pump tanks and pits, diversion boxes (DBs) and valve boxes will be closed as appropriate with a goal of closing geographic sections of FTF in stages.
- Following closure of a geographic section such as the Type IV tanks and 242-F Evaporator area, the area will be left in an interim closure state in preparation for final closure. For example, the area may be filled in with backfill after closure of the individual waste tanks and ancillary equipment to establish an even grade elevation with the remainder of FTF.
- Following closure of all FTF waste tanks and ancillary equipment, FTF will undergo a Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) closure in accordance with the FFA (FFA 1993, WSRC 2008b).

The FTF is currently in the operational period, during which residual waste removal from the waste tanks is in progress, and the tanks will be grouted in accordance with FFA commitments (FFA 1993). It is currently anticipated that the operational period will last until between 2020 and 2030. Once the FTF waste tanks and ancillary equipment have been grouted, a closure cap will be installed and a 100-year period of institutional control will follow. Installation of the FTF closure cap will be coordinated with CERCLA closure activities in the area, and will be designed with an appropriate interface with adjacent CERCLA closure systems (WSRC 2008b).

3.4.2.2 Waste Configuration in the F-Area Tank Farm at Closure

There are two primary categories of waste facilities that will be stabilized for contaminant retention in FTF:

- Waste Tanks
- Ancillary Equipment

Waste tanks refer to the 22 subsurface tanks in FTF designed for storing aqueous liquid wastes. Ancillary equipment refers to the other equipment used in the FTF to transfer waste (e.g., transfer lines, pump tanks) and reduce waste volume though evaporation (e.g., the evaporator systems) (WSRC 2008b).

The waste tanks are all built of carbon steel and reinforced concrete. Although the three principal types of waste tanks in FTF (Types I, III/IIIA, and IV) were constructed at different times during which design features were improved on, all of the waste tanks are of the same basic construction design (WSRC 2008b).

The main component of a waste tank is the primary tank where the liquid waste is contained. The primary tank is cylindrical in shape and made of carbon steel. Each primary tank type differs in size and capacity. Type I and III/IIIA tanks are enclosed by a secondary containment shell, which is larger in diameter than the primary tank. The secondary containment shell, like the primary tank, is constructed of carbon steel. Since the secondary containment is larger in diameter than the primary tank, an area is formed between them called the annulus. The annulus differs in size and capacity for each tank type. A reinforced concrete vault surrounds the secondary containment. A working slab underlies the basemat of the Type I and Type III/IIIA tanks (WSRC 2008b). Type IV tanks do not have a secondary containment.

The eight Type I tanks in the FTF are 75 feet in diameter and 24.5 feet high, with a nominal operating capacity of 750,000 gallons. The working slab for a Type I tank is 4 inches thick, with an approximate 42-ft radius. A 30-in reinforced concrete base (i.e., the basemat) sits on top of the working slab. A 3-in layer of grout sits on top of the basemat, and the primary container sits above the grout. The inner radius of the cylindrical carbon steel primary container is 37.5 ft, and the inner height is 24.5 ft. The walls of the primary container are welded to the top and bottom of the waste tank. The annulus pan (secondary liner) for the Type I tanks forms an annulus space 2.5 ft wide between the secondary and primary liners. An 80- ft inner diameter vault surrounds the Type I tank primary container; the vault is formed by a 22-in thick reinforced concrete roof and walls that surround the primary container and connect to the basemat. Twelve concrete and steel columns support the roof of a Type I tank. These columns were made from 0.5-in thick carbon steel pipes, with a 2-ft outside diameter, welded to a steel bottom plate and filled with concrete (WSRC 2008b).

The two Type III and eight Type IIIA tanks in the FTF are 85 ft in diameter and 33 ft high, with a nominal operating capacity of 1,300,000 gallons. Type III tanks have a 4-in working slab that extends 30 ft beyond the edge of the waste tank and slopes away from the tank. The Type III tank basemat, made of reinforced concrete, has a 3.5-ft minimum thickness (5.3 ft at drop panel at tank center) with a radius of approximately 45 ft. Type IIIA tanks have a 4-in working slab that extends under and beyond all eight Type IIIA tanks. The Type IIIA tank basemat, made of reinforced concrete, has a 3.6-ft minimum thickness (6.3 ft at drop panel at tank center) with a radius of approximately 45 ft. The Type III/IIIA tank liners are made of concentric cylinders joined to washer-shaped top and bottom plates. The thickness of the carbon steel plates varies from 0.5 in to 0.875 in. The secondary liner is made from 0.375-in carbon steel and is the same height as the primary tank with a diameter of 90 ft (creating a 2.5-ft annulus for both Type III and Type IIIA tanks). The Type III/IIIA tanks are completely enclosed in a concrete vault. The vault roof is at least 48-in thick and the walls are 30-in thick.

The four Type IV tanks in the FTF are 85 feet in diameter and approximately 34 ft high at the side wall, with a nominal operating capacity of 1,300,000 gallons. The 4-in thick reinforced concrete slab basemat of the Type IV tank is covered with a wire mesh, and topped with 3 in of cement, which together comprise a nominal 7-in basemat. The Type IV tank primary container is a cylinder (open at top), made of 0.375-in carbon steel plates. There is no secondary containment structure for Type IV tanks. These tanks are completely enclosed in a concrete vault, built around the primary tank. The vault walls form a cylinder 85 ft in diameter and 33 ft high surmounted by the dome ring. The core wall is 7-in thick at the top and 11-in at the bottom. The total wall thickness, including bands and cover, is 10-in at the top and 15-in at the bottom.

The FTF contains ancillary equipment with a residual radiological inventory that is accounted for as a part of facility closure in the PA. This ancillary equipment includes buried pipe (transfer lines), pump tanks, and evaporators, all of which have been in contact with liquid waste over the operating life of the facility. The ancillary equipment was used in the FTF to transfer waste and reduce waste volume though evaporation. The amount of contamination on these components depends on such factors as the service life of the component, its materials of construction, and the contaminating medium in contact with the component (WSRC 2008b).

The following FTF equipment is specifically addressed in the PA (WSRC 2008b):

- The FTF transfer line system (approximately 45,000 linear feet of transfer lines), including transfer line jackets, Leak Detection Boxes (LDBs) and other transfer line secondary containment systems (e.g., the Type I tank transfer line encasements).
- The FTF pump tanks (FPT-1, FPT-2, and FPT-3) and PPs (FPP-1, FPP-2, and FPP-3) and FTF catch tank.
- The 242-F evaporator system, including the evaporator cell and support tanks (e.g., Mercury Collection Tank, Cesium Removal Column (CRC) Pump Tank, CTS pump tank, and overheads tanks).
- The 242-16F evaporator system, including the evaporator cell and support tanks (e.g., Mercury Collection Tank, CRC Pump Tank, and overheads tanks).

In May 2002, DOE issued the *Savannah River Site High-Level Waste Tank Closure, Final Environmental Impact Statement* (DOE 2002) in which alternatives for waste tank cleaning and stabilization were considered. The alternative selected for the FTF (and the HTF) involves emptying, cleaning, and filling with grout, as a means to provide long-term stability, to enhance retention of radionuclides by using specialized grout formulations, and to minimize worker exposures.

In addition to grouting of the tanks, ancillary equipment may also be grouted in place. Whether piping will be grouted will be determined based upon grouting practicality and elimination of subsidence potential. At this time there are no plans to grout underground transfer lines. Ancillary equipment including the evaporator buildings, the catch tank, PPs, and DBs to the extent practical will remain in place and be grouted, as practical, to eliminate subsidence potential. An exception to leaving ancillary equipment in place will be made for equipment that is significantly higher in elevation than the adjacent waste tanks and would therefore result in a significant increase in the closure cap elevation.

3.4.2.3 F-Area Tank Farm Performance Assessment Requirements

As noted in Section 3.4.2.1, the FTF PA has been prepared to provide the technical basis to demonstrate compliance with both the Ronald W. Reagan NDAA for Fiscal Year 2005 Section 3116 criteria (DOD 2004), and the facility closure requirements of DOE Order 435.1-1 (DOE 1999a). The NDAA references as applicable the 10 CFR 61 performance objectives. The 10 CFR 61 performance objectives include:

- 25 mrem/yr all-pathways total effective dose equivalent limit
- 500 mrem/yr whole body-dose intruder protection limit (as interpreted by the NRC [NRC 2007])

In addition the FTF PA provides information relative to the following DOE Manual DOE M 435.1-1, Chapter IV (DOE 1999b) facility closure requirement performance objectives/measures for completeness:

- 25 mrem/yr all-pathways total effective dose equivalent limit
- 500 mrem/yr acute intruder total effective dose equivalent limit
- 100 mrem/yr chronic intruder total effective dose equivalent limit
- 10 mrem/yr air pathway limit, excluding radon
- 20 $pCi/m^2/s$ average radon flux limit
- MCLs for radionuclides in drinking water (40 CFR 141 and WSRC 2008a)

The MCLs applicable to radionuclides in drinking water, and thus in groundwater at the SRS, are listed in Table 3-25. In the absence of MCLs, groundwater concentrations were compared to calculated Preliminary Remediation Goals (PRGs) in the FTF PA, which are risk-based tools used to evaluate and clean up contaminated sites (WSRC 2008b). The NDAA Section 3116 criteria, as interpreted by the NRC (NRC 2007), require that a 10,000-yr period after closure be evaluated, rather than the 1,000-yr period specified in the DOE Order (DOE 1999b).

3.4.2.4 F-Area Tank Farm Performance Assessment Results

The FTF PA results for 60 radionuclides (plus daughter isotopes) considered in the FTF analyses are summarized in Table 3-27 and the accompanying Note 1 and Note 2. The FTF results indicate that the groundwater pathway is the most important pathway for exposure to radionuclides released from the facility. The calculated air pathway doses are a fraction of the groundwater pathway doses (Table 3-27). The dose for the chronic intruder agriculture (post drilling) scenario was due almost entirely to ingestion of vegetables contaminated with drill cuttings (WSRC 2008b).

For the peak groundwater concentrations calculated, three radionuclides (Np237, U233, and U234) exceeded the PRG over the 10,000-yr period at a distance of 100 m from the FTF. No MCLs were exceeded at 100 m. All radionuclides were well below the MCL or PRG at the seeplines, evaluated at both interfaces of the groundwater aquifers with UTR and Fourmile Branch (WSRC 2008b).

Table 3-27. Summary of Radiological Results for F-Tank Farm

Location	Peak Within 10,000 Years		
	All-Pathways Dose[a] (mrem/yr)	Groundwater Pathway Dose (mrem/yr)	Air Pathway Dose (mrem/yr)
100 m from FTF	1.4 at ~ year 10,000	1.3 at ~ year 10,000	0.14
At Seepline	0.25 at ~ year 3,800	0.21 at ~ year 3,800	0.035

Note 1: The peak intruder dose is 73 mrem/yr at year 101 from a chronic scenario; drilling through a transfer line and using groundwater concentrations at the maximum 1-m FTF location.
Note 2: The peak radon flux at the ground surface is 9.3E-08 pCi/m^2/sec.
Source: WSRC 2008b.
[a] All-Pathways Dose assumes groundwater is the only drinking water source.

3.4.2.5 F-Area Tank Farm Performance Assessment Conceptual Models

The conceptual model of radionuclide release from the grouted tanks and ancillary equipment in the FTF considered a non-uniform leaching rate dependent on the chemical state of pore fluid contacting the stabilized contaminant at any given time. Different release models were used for different types of tanks and equipment. The radionuclides for which groundwater transport calculations were performed for the FTF PA are listed in Table 3-28.

Table 3-28. Radionuclides used in FTF Modeling (from WSRC 2008b)

Ac227	Cf251	Cs134	Nb94	Pu240	Sb126	Th230
Al26	Cf252	Cs135	Ni59	Pu241	Sb126m	Th232
Am241	Cm242	Cs137	Ni63	Pu242	Se79	U232
Am242m	Cm243	Eu152	Np237	Pu244	Sm151	U233
Am243	Cm244	Eu154	Pa231	Ra226	Sn126	U234
Ba137m	Cm245	Eu155	Pm147	Ra228	Sr90	U235
Bk249	Cm247	H3	Pr144	Rh106	Tc99	U236
C14	Cm248	I129	Pu238	Ru106	Te125m	U238
Ce144	Co60	Na22	Pu239	Sb125	Th229	Y90
Cf249						

In this table Cf251, Cf252, Ra228, and Th232 are considered only as daughter products of other isotopes

The conceptual model of vadose zone transport for the FTF assumed that the most likely distance between the bottom of the tanks and the water table ranged from 1.2 to 19.9 ft. The minimum distance was assumed to range from 0.1 to 7.9 ft, assuming the maximum water table elevation rise over the average elevation is 12 ft (WSRC 2008b).

The conceptual model of groundwater flow and transport for the FTF PA was developed by SRNL based on the evaluation of water elevation data from the well network in the F-Area vicinity, as well as other information derived from the extensive hydrogeologic characterization data collected in the area. The aquifers considered potentially contaminated by radionuclides leaching from the FTF include the UTR aquifer unit, in which the water table resides, and the underlying Gordon aquifer. Contamination will not enter the deep Crouch Branch aquifer in this area because of the upward gradient that exists between this lower aquifer and the overlying Gordon aquifer near UTR stream (Figure 3-37 and WSRC 2008b).

Figure 3-87 shows the potentiometric surface for the water table (UTR) aquifer in the GSA, based on mean 2003 data, the location of the groundwater divide in this area during 2003, and the operational areas, including F-Area. Based on the water table contours and the location of the groundwater divide, the conceptual model of the saturated zone for the FTF PA assumes flow in the water table unit both northward toward UTR, and southward, toward Fourmile Branch (WSRC 2008b). The potentiometric surface for the Gordon aquifer unit (Figure 3-45) indicates a north-northwest flow direction under the entire GSA, due to the influence of the incisement of UTR in this aquifer.

3.4.3 H-Tank Farm Performance Assessment

3.4.3.1 General Performance Assessment and Facility Information for the H-Area Tank Farm

A PA is currently being developed to support the eventual closure of the HTF underground radioactive waste tanks and ancillary equipment. This PA provides the technical basis used to demonstrate compliance with both the Ronald W. Reagan NDAA for Fiscal Year 2005 Section 3116 criteria (DOD 2004), and the facility closure requirements of DOE Order 435.1-1 (DOE 1999a).

The HTF is located in the central region of the SRS, within the GSA, occupying 45 acres in H-Area (Figure 3-88). The HTF is an active waste storage facility consisting of 29 carbon-steel waste tanks, the layout of which is shown in Figure 3-89, that store or once stored liquid radioactive waste generated primarily from the H-Canyon Facility (H-Area's main processing facility). Tank 16 in HTF is no longer in service, its primary tank having been chemically cleaned in 1970.

The HTF waste tanks were built during four separate construction periods, with a different tank design for each period, leading to the designation of the following five different tank groups:

- Tanks 9 through 12 are Type I tanks and were constructed in the early 1950's.
- Tanks 13 through 16 are Type II tanks and were constructed between 1955 and 1956.
- Tanks 21 through 24 are Type IV tanks and were constructed between 1958 and 1962.
- Tanks 29 through 32 are Type III tanks and were constructed between 1967 and 1970.
- The fifth group of thirteen tanks, which consists of Tanks 35 through 43 and 48 through 51 are Type IIIA tanks and were constructed between 1974 and 1981.

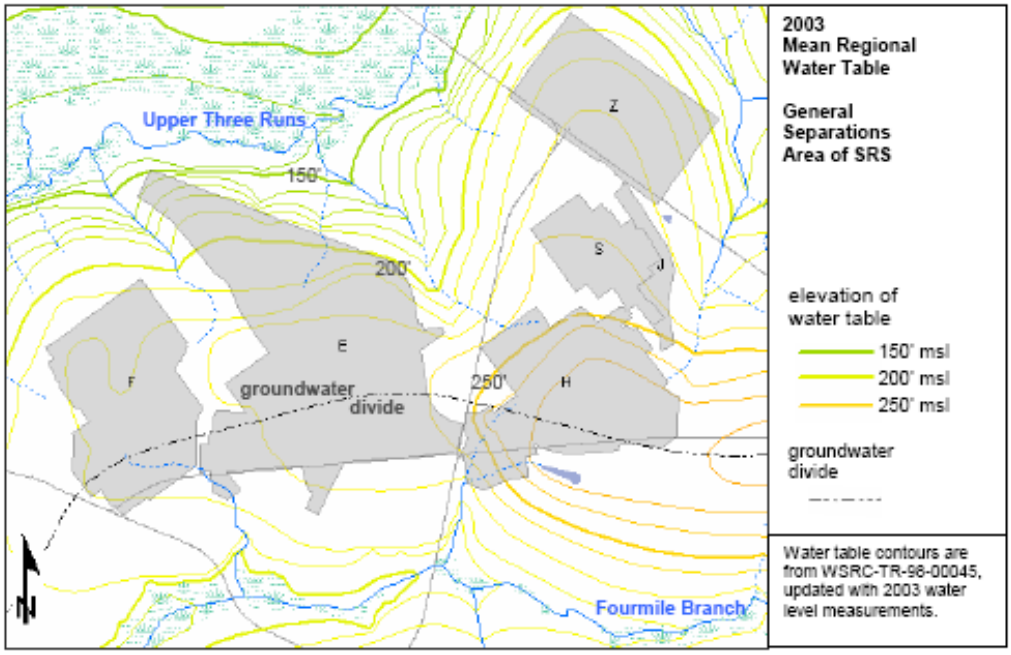


Figure 3-87. Mean Water Table Elevations in the GSA in 2003 and Location of Water Table Divide Relative to Operational Areas

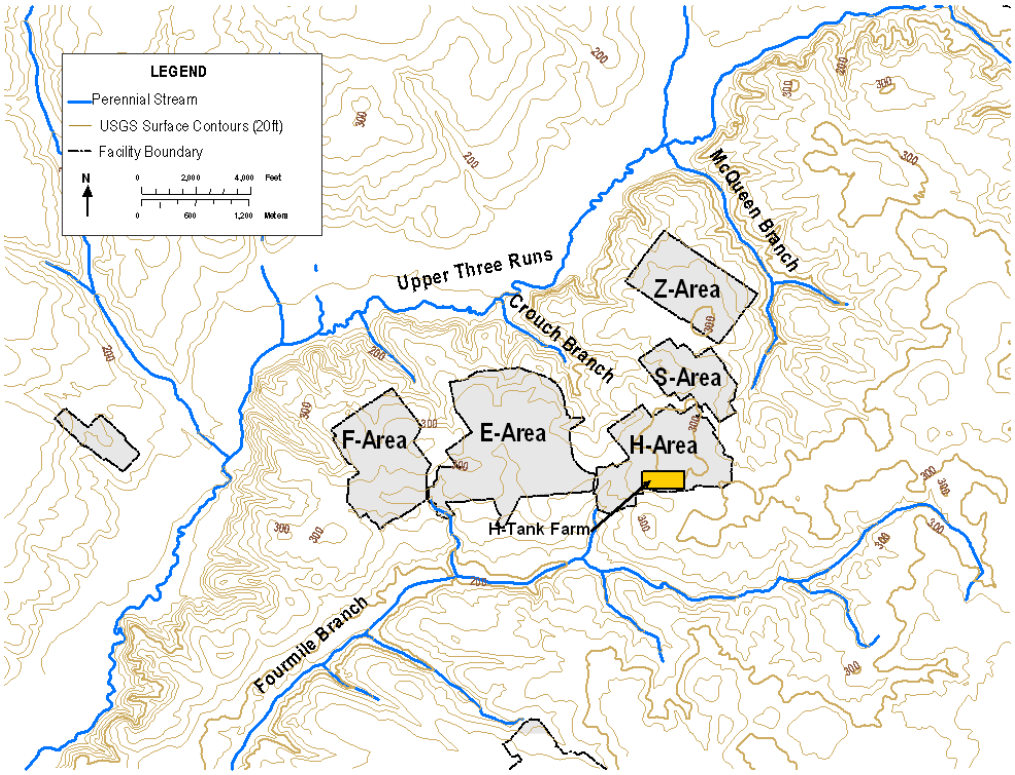


Figure 3-88. Location of the HTF within H-Area

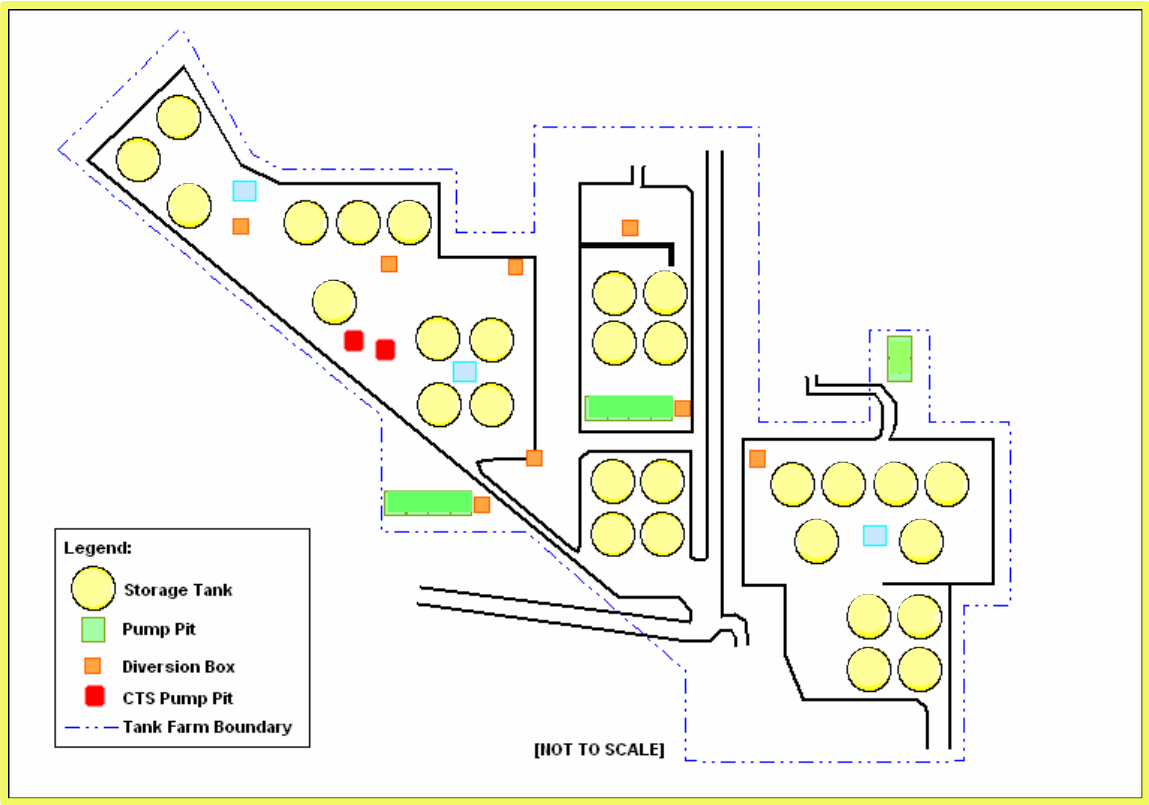


Figure 3-89. Layout of the H-Tank Farm in H-Area

The proposed sequence of events for closure of HTF is as follows:

- Closure of the Types I, II, and IV tanks and finally the Type III/IIIA tanks. The ancillary equipment, such as transfer lines, pump tanks and pits, DBs and valve boxes, will be closed as appropriate with a goal of closing geographic sections of HTF in stages.
- Following closure of a geographic section such as the Type IV tanks and 242-H Evaporator area, the area will be left in an interim closure state in preparation for final closure. For example, the area may be filled in with backfill after closure of the individual waste tanks and ancillary equipment to establish an even grade elevation with the remainder of HTF.

Following closure of all HTF waste tanks and ancillary equipment, HTF will undergo closure in accordance with the FFA and CERCLA.

3.4.3.2 Waste Configuration in the H-Area Tank Farm at Closure

Two primary categories of waste facilities will be stabilized for contaminant retention in the HTF:

- Waste Tanks
- Ancillary Equipment

Waste tanks refer to the 29 subsurface tanks in the HTF designed for storing aqueous liquid wastes. Ancillary equipment refers to the other equipment used in the HTF to transfer waste (e.g., transfer lines, pump tanks) and reduce waste volume though evaporation (e.g., the evaporator systems).

The waste tanks are all built of carbon steel and reinforced concrete. Although the four principal types of waste tanks in the HTF (Types I, II, III/IIIA, and IV) were constructed at different times during which design features were improved on, all of the waste tanks are of the same basic construction design. Details concerning Types I, III/IIIA, and IV tanks were provided in 3.4.2.2 for the FTF. The Type II tanks were slightly larger than the Type I tanks, were enclosed by a secondary containment shell, and were designed for sludge of a higher temperature. Only a single center column was used for support to allow the tank bottom to expand and contract more freely (Reed et al. 2002).

Details of the grouting of the ancillary equipment and the tanks will be available when the PA is completed. Once the HTF waste tanks and ancillary equipment have been grouted, a closure cap will be installed and a 100-year period of institutional control will begin.

3.4.3.3 H-Area Tank Farm Performance Assessment Requirements

As noted in Section 3.4.3.1, the HTF PA is being prepared to provide the technical basis to demonstrate compliance with both the Ronald W. Reagan NDAA for Fiscal Year 2005 Section 3116 criteria (DOD 2004), and the facility closure requirements of DOE Order 435.1-1 (DOE 1999a). The NDAA references as applicable the 10 CFR 61 performance objectives. The 10 CFR 61 performance objectives include:

- 25 mrem/yr all-pathways total effective dose equivalent limit
- 500 mrem/yr whole body-dose intruder protection limit (as interpreted by the NRC (NRC 2007))

In addition the HTF PA will provide information relative to the following DOE Manual DOE M 435.1-1, Chapter IV (DOE 1999b) facility closure requirement performance objectives/measures for completeness:

- 25 mrem/yr all-pathways total effective dose equivalent limit
- 500 mrem/yr acute intruder total effective dose equivalent limit
- 100 mrem/yr chronic intruder total effective dose equivalent limit

- 10 mrem/yr air pathway limit, excluding radon
- 20 pCi/m^2/s average radon flux limit
- MCLs for radionuclides in drinking water (40 CFR 141 and WSRC 2008a)

The MCLs applicable to radionuclides in drinking water, and thus in groundwater at the SRS, are listed in Table 3-25. In the absence of MCLs, groundwater concentrations will be compared to calculated PRGs, which are risk-based tools used to evaluate and clean up contaminated sites (WSRC 2008b).

The NDAA Section 3116 criteria, as interpreted by the NRC (NRC 2007), require that a 10,000-yr period after closure be evaluated, rather than the 1,000-yr period specified in the DOE Order (DOE 1999b).

3.4.3.4 H-Area Tank Farm Performance Assessment Results

The PA for the HTF is currently in progress. Results are not yet available. However it is anticipated that the HTF results will be similar to that of FTF (see Section 3.4.2.4), in that the groundwater pathway is anticipated to also be the most important pathway for exposure to radionuclides released from the facility.

3.4.3.5 H-Area Tank Farm Performance Assessment Conceptual Models

Although not formally documented at this writing, the conceptual model of groundwater flow and transport for the HTF PA will be developed similarly to that for the FTF PA, based on the evaluation of water elevation data from the well network in the H-Area vicinity, as well as other information derived from the extensive hydrogeologic characterization data collected in the area. The aquifers considered potentially contaminated by radionuclides leaching from the FTF include the UTR aquifer unit, in which the water table resides, and the underlying Gordon aquifer. Contamination will not enter the deep Crouch Branch aquifer in this area because of the upward gradient that exists between this lower aquifer and the overlying Gordon aquifer near UTR stream (Figure 3-37 and Section 3.2.10.2).

The mean 2003 potentiometric surface for the water table (UTR) aquifer in the GSA is shown in Figure 3-87. The groundwater divide relative to H-Area for that year is also shown on this figure. This figure indicates that, like F-Area, the HTF is likely near the groundwater divide, and flow in the water table aquifer may be in the direction of both UTR and Fourmile Branch. The potentiometric surface for the Gordon Aquifer (Figure 3-44) indicates a north-northwest flow direction under the entire GSA, due to the influence of the incisement of UTR in this aquifer.

3.4.4 Saltstone Disposal Facility Performance Assessment

3.4.4.1 General Performance Assessment and Facility Information for the Z-Area Saltstone Disposal Facility

The Saltstone Facility (located in the Z-Area of SRS) consists of two facility segments: the SPF, which receives and treats the salt solution to produce saltstone grout, and the SDF, which consists of vaults used for the final disposal of the saltstone grout. The SPF is permitted as a wastewater treatment facility per the SCDHEC Regulations R.61-67, and the SDF is permitted as a Class 3 Industrial Solid Waste Landfill per SCDHEC Regulations R.61-107.16. Construction of SPF and the first two vaults of SDF were completed between February 1986 and July 1988. The Saltstone Facility started radioactive operations June 1990. Future vaults will be constructed on a "just-in-time" basis in coordination with salt processing production rates.

A PA was first prepared for the Z-Area SDF in 1992 (MMES 1992) in conformance with the DOE order in force at that time for LLW disposal (DOE Order 5820.2A [DOE 1988]). Subsequent to that first PA, DOE issued a new radioactive waste management order (DOE Order 435.1-1, DOE 1999a), and the Ronald W. Reagan NDAA for Fiscal Year 2005 Section 3116 criteria (DOD 2004) was promulgated; both of which are applicable to the Z-Area SDF. A revised SDF PA is in predecisional draft form (LWO 2009), having been developed in conformance with the requirements from both the Order and Act. Discussions presented herein concerning the Z-Area SDF are primarily drawn from this most recent draft PA.

The SDF (at Z-Area) is located in the central region of the SRS, in the GSA (Figure 3-90). Z-Area was chosen for the SDF site based on considerations of depth to the water table, distance to surface water and the public, available surface area, surface topography, and its proximity to the wastewater generation site (MMES 1992). Z-Area is located approximately 6.2 mi from the nearest SRS site boundary and lies on a local topographic high at approximately 290 feet above mean sea level (WSRC 2007a). This elevated area of approximately 160 acres is well drained, as evidenced by the lack of marshes or other bodies of standing water. Z-Area is bounded by two streams: UTR and McQueen Branch, a tributary to UTR. McQueen Branch is located approximately 1.0 mi to the east and 0.75 mi to the northeast of Z-Area, and UTR is located approximately 1.0 mi to the northwest (WSRC 2007a). Figure 3-91 illustrates the anticipated layout of the SDF.

There are currently two existing rectangular saltstone vaults (Vaults 1 and 4) and one pair of cylindrical disposal cells (Disposal Cell 2A and 2B) under design within the SDF. The two existing vaults (shown as "#1" and #4" in Figure 3-91) are constructed of reinforced concrete containing slag (LWO 2009). Slag has also been incorporated into the saltstone composition, and will be used in the cylindrical disposal cells.

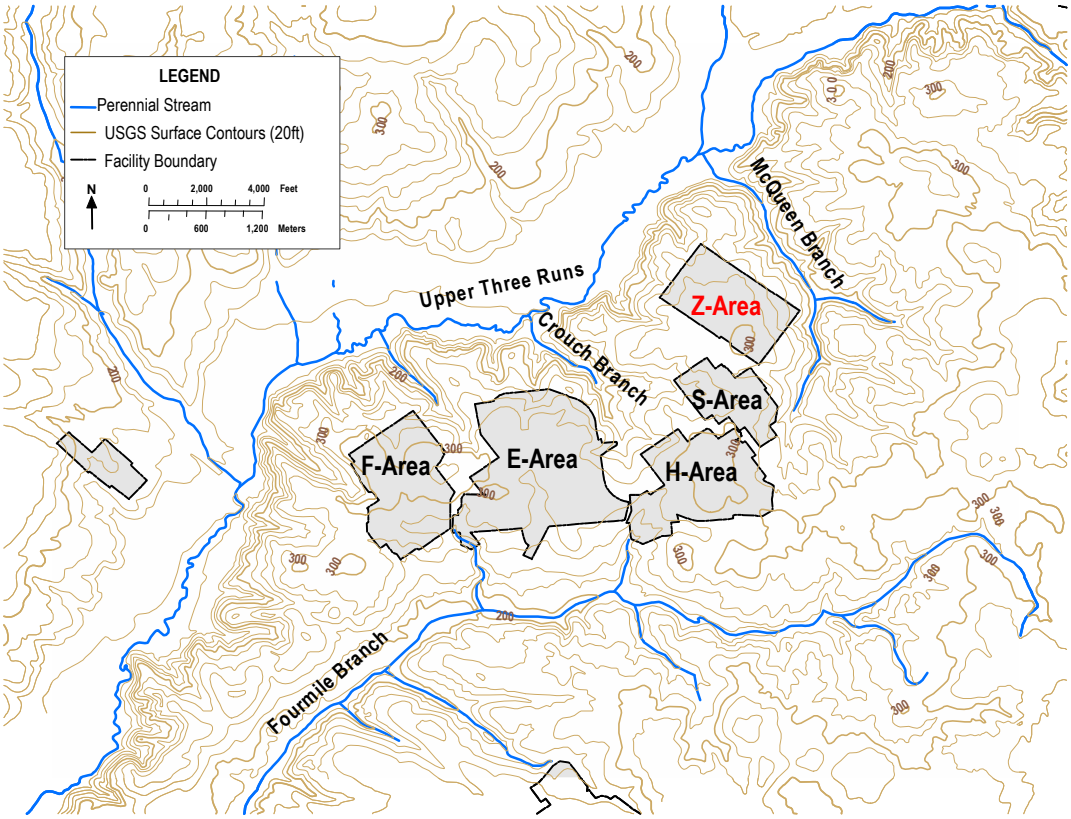
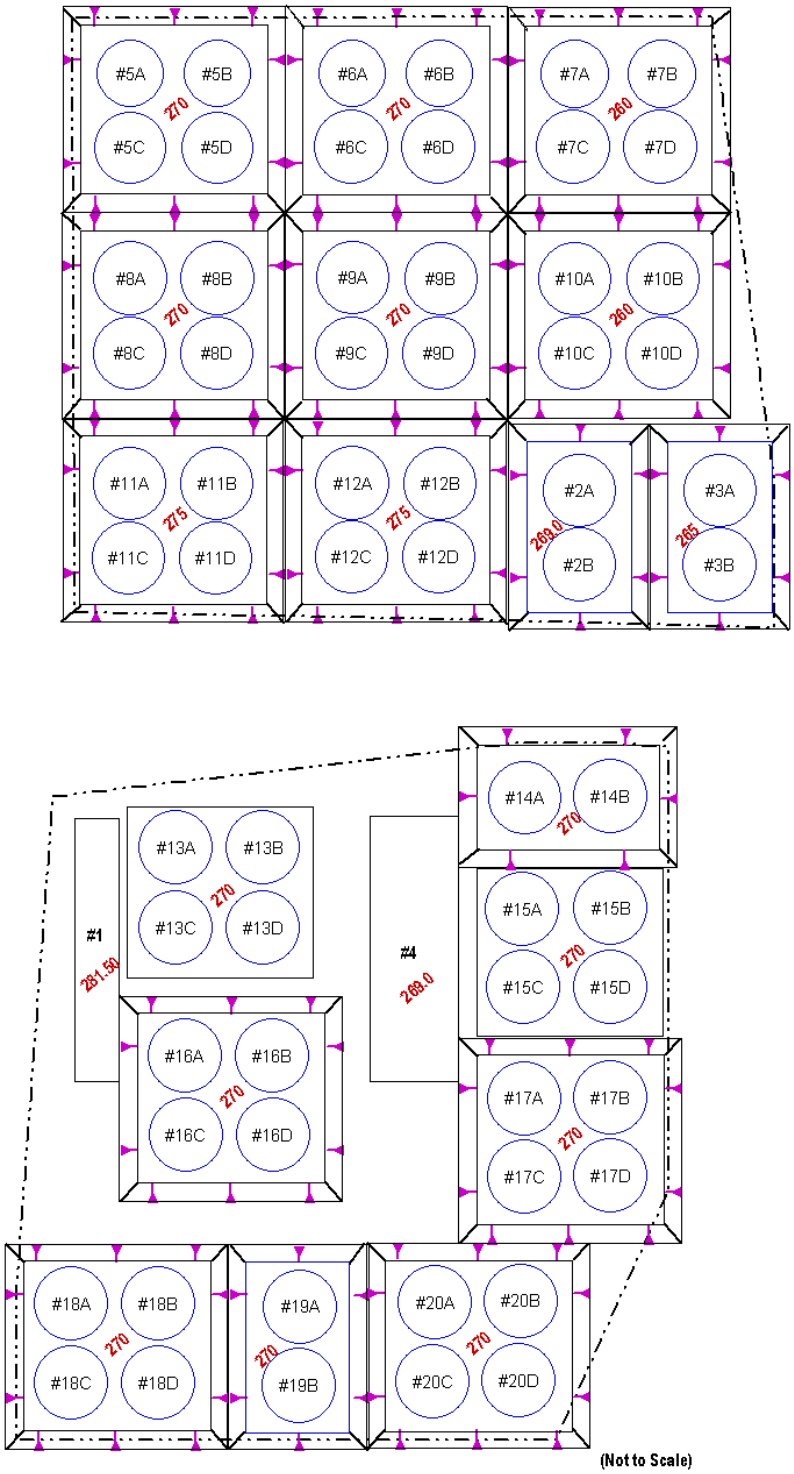


Figure 3-90. Location of the GSA and Z-Area

The SDF is currently in the construction-operational period, during which the disposal cells will be constructed and waste will be placed. Three of the six cells of Vault 1 have been filled with grout, clean capped, and the permanent roof has been sealed.

An engineered closure cap will be installed over the SDF following the closure of the final vaults and cylindrical disposal cells. The SDF closure cap is primarily intended to provide physical stabilization of the site, to minimize infiltration, and to provide an intruder deterrent. It is anticipated that the SDF closure cap will consist of multiple layers to accomplish these purposes (Jones and Phifer 2008). The upper-most vegetative layer will be underlain by soil and other materials to inhibit erosion, infiltration, and intrusion.

Final closure will consist of site preparation and construction of an integrated closure system composed of one or more closure caps installed over all the disposal cells, and construction of a drainage system at the end of the operational period.



Note: The numbering identifications for Vaults 1 and 4 and Disposal Cells 2A and 2B in this figure are fixed. The numbering identification for the balance of the (future) disposal cells should be considered placeholders.

Figure 3-91. General Layout of the SDF

3.4.4.2 Saltstone Disposal Facility Waste Form and Disposal Vaults and Cells

For disposal, salt solution from the FTF and HTF will be sent to the Saltstone Facility to be mixed with dry materials consisting of cement, slag, and flyash to form a grout for permanent disposal. The grout is pumped into the SDF disposal vaults and cells, where it is allowed to harden into a concrete-like non-hazardous solid waste form called saltstone. The grout formulation prepares the grout to be free-flowing and self-leveling. The consistency of the grout allows it to flow to the walls of the SDF cells. Minimal material segregation occurs during the pouring and curing process (WSRC 2005b).

There are projected to be 64 disposal cells to be identified by number beyond the existing two rectangular disposal vaults. Currently Vaults #1 and #4 have been built and contain disposed material.

Vault #1 is an above grade, rectangular, reinforced concrete vault. It is approximately 600-ft long, 100-ft wide and 27-ft high. It is divided into two units each 100-ft by 300-ft, with a 3-in separation gap between the units. Each unit is further divided into three cells of 100-ft by 100-ft.

Vault #4, also above grade, has the dimensions of approximately 600-ft long, 200-ft wide, and 30-ft high, and is divided into 12 cells, with each cell measuring approximately 100-ft x 100-ft. It is very similar to Vault #1, but twice the width. It is divided into two units each 200-ft by 300-ft, with a 3-in separation gap between units.

Vault 1 and 4 systems filled with saltstone consist of the following:

- Controlled compacted backfill soil base
- 2-ft thick, reinforced concrete, floor slab
- 1.5-ft thick, reinforced concrete walls
- 24-ft of saltstone or other cementitious waste forms poured into the cell from above (25-ft for Vault 4)
- Minimum 6-in thick clean grout cap (15-in thick minimum for Vault 4)
- Poured-in-place concrete roof with an approximately 2% slope

Eight cells in Vault 4 have been modified with a drainage system designed to relieve the hydrostatic pressure on the vault walls. This sheet drain system will collect any liquid that may appear between the solidified grout and the vault wall. Collected liquid will be returned through pipes to the SPF and used in the production of grout.

Disposal Cell 2 is currently under construction. It will consist of two identical cells, Cell 2A and 2B. Both cells are 150-ft in diameter by 22-ft high (interior height), below grade (to the roof), cylindrical, reinforced concrete tanks. Each tank (cell) consists of the following:

- Controlled compacted backfill soil base
- 4-in thick lower mud-mat of concrete
- Geosynthetic clay liner (GCL) consisting of a minimum 0.75 lbs/ft^2 sodium bentonite
- 100 mil HDPE geomembrane
- 4-in thick upper mud-mat of Class III sulfate resistant concrete
- Minimum 8-in thick cast-in-place reinforced concrete floor slab of Class III sulfate resistant concrete
- Minimum 8-in thick pre-cast reinforced concrete walls of Class III sulfate resistant concrete
- The exterior side of the walls covered with a 100 mil HDPE geomembrane
- Maximum 20-ft of saltstone or other cementitious waste form poured into the tank through a roof penetration
- Minimum 2-ft clean non-radioactive grout cap to fill between the saltstone and roof poured into the tank through roof penetrations
- Minimum 8-in thick reinforced concrete roof of Class III sulfate resistant concrete at a minimum 2% slope - in place prior to the saltstone pour
- Roof penetrations will exist to pour saltstone, to pour the clean grout cap, for ventilation, for monitoring (temperature and cameras), and for personnel access

The saltstone waste stream is high in sulfates, which can attack hardened concrete. Due to the high-sulfate nature of the waste stream that will be disposed in the disposal cell, its floor, walls, and roof are being constructed with a Class III sulfate resistant concrete mix which is designed to retard the ingress and movement of water and sulfate attack. Pozzolans and slag are added to the cement to reduce the water to cementitious materials ratio. They also help reduce the saturated hydraulic conductivity of the concrete. In addition, the walls are constructed with wire pre-stressing and the exterior is backfilled, which allows the walls to be under compression, which further minimizes water penetration into the walls.

Prior to closure, the roof penetrations in the disposal cells will have to be sealed with Class III sulfate resistant concrete, and a GCL overlain by an HDPE geomembrane will have to be placed around the disposal cell roof. Future cells are planned similar to Disposal Cell 2.

3.4.4.3 Saltstone Disposal Facility Performance Assessment Requirements

As noted in Section 3.4.4.1, the revised SDF PA has been drafted, and provides the technical basis to demonstrate compliance with both DOE Order 435.1-1 (DOE 1999a) and the Ronald W. Reagan NDAA for Fiscal Year 2005 Section 3116 criteria (DOD 2004).

DOE Order 435.1-1 (DOE 1999a) requires that PAs for LLW disposal facilities assess all-pathway, air, and hypothetical inadvertent intruder doses; radon flux; and impacts to water resources, according to the *Radioactive Waste Management Manual* (DOE M 435.1-1, DOE 1999b).

For the Z-Area SDF, the PA-required assessments will be utilized at the SRS for determining conformance of the results based on the projected inventory in the waste stream with the performance measures at the points of compliance outlined in Table 3-24, over a 1,000-yr period after closure of the facility (DOE 1999b). As with the other SRS PA efforts, the appropriate measure for protection of water resources has been determined to be the SDWA MCLs (40 CFR Part 141 and WSRC 2008a). The MCLs applicable to radionuclides in drinking water, and thus in groundwater at the SRS, are listed in Table 3-25.

The NDAA references as applicable the 10 CFR 61 performance objectives. The 10 CFR 61 performance objectives include:

- 25 mrem/yr all-pathways total effective dose equivalent limit
- 500 mrem/yr whole body-dose intruder protection limit (as interpreted by the NRC [NRC 2007])

The NDAA Section 3116 criteria, as interpreted by the NRC (NRC 2007), require that a 10,000-yr period after closure be evaluated, rather than the 1,000-yr period specified in the DOE Order (DOE 1999b).

3.4.4.4 Saltstone Disposal Facility Performance Assessment Results

Radiological dose to human receptors was analyzed in the revised SDF PA in the all-pathways analysis, the inadvertent intruder analysis, and the air pathway analysis, for comparison to the relevant performance measures. Seventy isotopes were modeled for the groundwater transport and intruder analyses, and eight were modeled in the air pathways analysis. Radon-222 fluxes were also evaluated (LWO 2009).

The pathways that were identified that result in potential dose contributions were leaching from the saltstone waste form resulting in contamination of the local groundwater and gaseous diffusion into the atmosphere above the disposal units. For the inadvertent intruder analysis, only the chronic intruder agriculture scenario was considered to credibly result in potential exposure.

The SDF PA results indicate that all of the regulatory limits for the SDF are met within the 10,000 year period of compliance. The peak 100m all-pathways dose in the 10,000 year performance period is 1.2 mrem/yr in a region of the groundwater flow path that is most directly impacted by Vault 4. Vault 4 is the primary contributor (>90%) to the 100m peak all-pathways dose in this region at year 10,000. The 100m all-pathways dose during the 10,000 year performance period is primarily associated with Ra226. While there is very little Ra226 in the disposal units, the Ra226 is a daughter product of Th230, of which there is an appreciable quantity in Vault 4. The all-pathways dose was dominated by the groundwater pathway, with the primary exposure pathway contributors being water ingestion, fish ingestion, and vegetable ingestion. The airborne dose contribution to the all-pathways dose was negligible.

The peak dose for the chronic intruder agriculture scenario in the 10,000 year performance period was 1.1 mrem/yr at year 10,000, primarily due to water ingestion (0.7 of the 1.1 mrem/yr). The principal radionuclide contributor to the peak intruder dose was Ra226.

The annual dose from airborne releases resulted in a total dose of less than 4.0E-09 mrem/yr, principally from H3 in the first 10 years after closure, at 100m from the SDF. The peak instantaneous flux of 2.0E-13 pCi/m^2/sec was estimated to occur above Vault 1 at approximately 10,000 years after closure.

3.4.4.5 Saltstone Disposal Facility Performance Assessment Conceptual Models

The list of seventy radionuclides for which groundwater transport and intruder dose calculations were performed for the SDF PA is listed in Table 3-29.

Table 3-29. Radionuclides used in SDF Modeling (LWO 2009)

Ac227	Cf251	Cs134	Nb93m	Pu238	Sb125	Th230
Al26	Cf252	Cs135	Nb94	Pu239	Sb126	Th232
Am241	Cl36	Cs137	Ni59	Pu240	Sb126m	U232
Am242m	Cm242	Eu152	Ni63	Pu241	Se79	U233
Am243	Cm243	Eu154	Np237	Pu242	Sm151	U234
Ba137m	Cm244	Eu155	Pa231	Pu244	Sn126	U235
Bk249	Cm245	H3	Pd107	Ra226	Sr90	U236
C14	Cm247	I129	Pm147	Ra228	Tc99	U238
Ce144	Cm248	K40	Pr144	Rh106	Te125m	Y90
Cf249	Co60	Na22	Pt193	Ru106	Th229	Zr93

Radionuclides of potential importance in the air pathway were identified by reviewing the list of elements in the initial waste tank inventory that were judged to have the potential to form a vapor phase in the waste form. Of those, only C14, Cl36, H3, I129, Sb125, Se79, Sn126, and Tc99 were determined to be present in the actual SDF inventory, and thus of potential significance.

Under Z Area only, the minimum depth to the water table from the ground surface in any given year is estimated to be 43 ft on the basis of water table fluctuations from several years' data. The average water table beneath the SDF is approximately 210 feet to 230 ft above msl with the probable maximum water table calculated to be about 234 ft to 248 ft above msl. For perspective, the base elevations above msl of existing Vaults 1, 4, and future Disposal Cells 2A and 2B are 281.5 ft, 269 ft, and 269 ft, respectively

The release of radiological and chemical contaminants from the SDF disposal units is simulated using an Integrated Site Conceptual Model (ISCM) approach for evaluating facility performance (LWO 2009). The mechanism generally controlling the release of contaminants from the saltstone is the adsorption characteristic of the saltstone expressed by the K_d which is element dependent and differs as the saltstone oxidation potential (E_h) and pH conditions change over time (e.g., from a reducing cementitious medium to an oxidized cementitious medium). The release of some species, most notably, Tc99 is highly influenced by E_h conditions.

This ISCM approach considers the integrity of the proposed SDF closure cap, the cementitious barriers of each disposal unit (roof, walls, and floor) and the saltstone. These key elements control overall contaminant release and transport to groundwater exposure points. Flow into and out of the disposal unit is impacted by the material properties of the closure cap and the cementitious materials comprising the disposal unit and the saltstone material. Time varying degradation of the closure cap and of the cementitious materials is included in the model.

The mobile contaminants in the closed SDF are considered to be released from the disposal unit and gradually migrate downward through unsaturated soil via advection-dominated transport. Transport of some contaminant species is significantly retarded by sorption to native and backfilled soils.

The conceptual model of groundwater flow and transport for the SDF PA was developed similarly to that for the other operational areas in the GSA, based on the evaluation of water elevation data from the well network in the Z-Area vicinity, as well as other information derived from the extensive hydrogeologic characterization data collected in the area. During the more than 15 years since the original Z-Area PA was published, new boreholes have been completed at the SDF site and significant additional data have been acquired from this location, thus improving the understanding of the stratigraphy beneath the facility.

The aquifers considered potentially contaminated by radionuclides leaching from the SDF include the UTR aquifer unit, in which the water table resides, and the underlying Gordon aquifer. Contamination will not enter the deep Crouch Branch aquifer in this area because of the upward gradient that exists between this lower aquifer and the overlying Gordon aquifer near UTR stream (Figure 3-37 and Section 3.2.10.2).

The SDF straddles a groundwater divide between UTR and McQueen Branch (LWO 2009). Some contaminants will be transported via groundwater through near surface aquifers and outcrop at either McQueen Branch or UTR. The potentiometric surface for the Gordon aquifer unit (Figure 3-45) indicates a north-northwest flow direction under the entire GSA, due to the influence of the incisement of UTR in this aquifer.

3.4.5 Transuranic Pad 1 Performance Assessment

3.4.5.1 General Performance Assessment and Facility Information for the E-Area Transuranic Pad 1

Currently the TRU waste stored on TRU Pad 1 is slated for shipment to the WIPP in Carlsbad NM for disposal in accordance with the SRS End State Vision (DOE 2005). In order to aid in the determination as to whether this approach is appropriate, a PA, which assumes that the TRU Pad 1 waste is disposed in place at SRS, is currently in the early stages of development. The PA may be prepared to evaluate the on-site disposal of TRU Pad 1 waste against both the 40 CFR Part 191 standards and DOE Order 435.1-1 performance measures for disposal of LLW. Conformance with 40 CFR Part 191 is required for the disposal, after January 1994, of all DOE TRU waste at a location other than the WIPP.

The following information generally describes the TRU Pad 1 facility, and is largely taken from the 2007 *Documented Safety Analysis for the Solid Waste Management Facility*, in which this facility is located (WSRC 2007b). TRU Pad 1 is located within the E-Area burial ground (643-7E), as shown in Figure 3-92, just south of the southeast corner of the MWMF.

TRU Pad 1 is an at-grade concrete pad. Although TRU waste on TRU Pad 1 is buried under an interim soil cover, it is considered an operational facility because the waste is considered to be stored rather than disposed. The waste overpacked in concrete containers on Pad 1 was received from Mound and Los Alamos Scientific Laboratory (LASL, now called Los Alamos National Laboratory). TRU waste items were placed in the containers from these facilities. These containers were shipped to SRS between 1970 and 1972. The containers contain heat source scrap (Pu238) and contaminated equipment items. Maintenance activities are conducted to maintain the soil cover (WSRC 2007b).

TRU Pad 1 is constructed of 12-inch thick reinforced concrete and is approximately 125 feet long by 50 feet wide. Pad 1 is used for storage of TRU waste contained within concrete culverts and concrete boxes, the layout of which is depicted in Figure 3-93.

TRU Pad 1 contains concentrations of Pu238 that exceed 100 nCi/g (3,700 Bq/g), which would normally be shipped to WIPP for disposal. However, to meet the transportation requirements for off-site shipment, the waste in question must be packaged. The nature of the Pu238 material creates concerns with regard to worker safety during handling, prompting consideration for on-site disposal, which would eliminate the need to open existing waste packages[1]. Pad 1 currently holds 266 m^3 of waste containing about 180,000 Ci, of mostly Pu238 (Cook 2005). About 80% of the Pu by weight was Pu238, and 16% was Pu239 (Towler 1979).

[1] Currently, a portion of the soil covering the culverts on Pad 1 has been removed. About one-third of the culverts have been removed and some of the culverts have been opened and the drums have been prepared for shipment to WIPP. Plans are to ship as much of this material to WIPP as can be done safely. Any residual material will have to be disposed onsite.

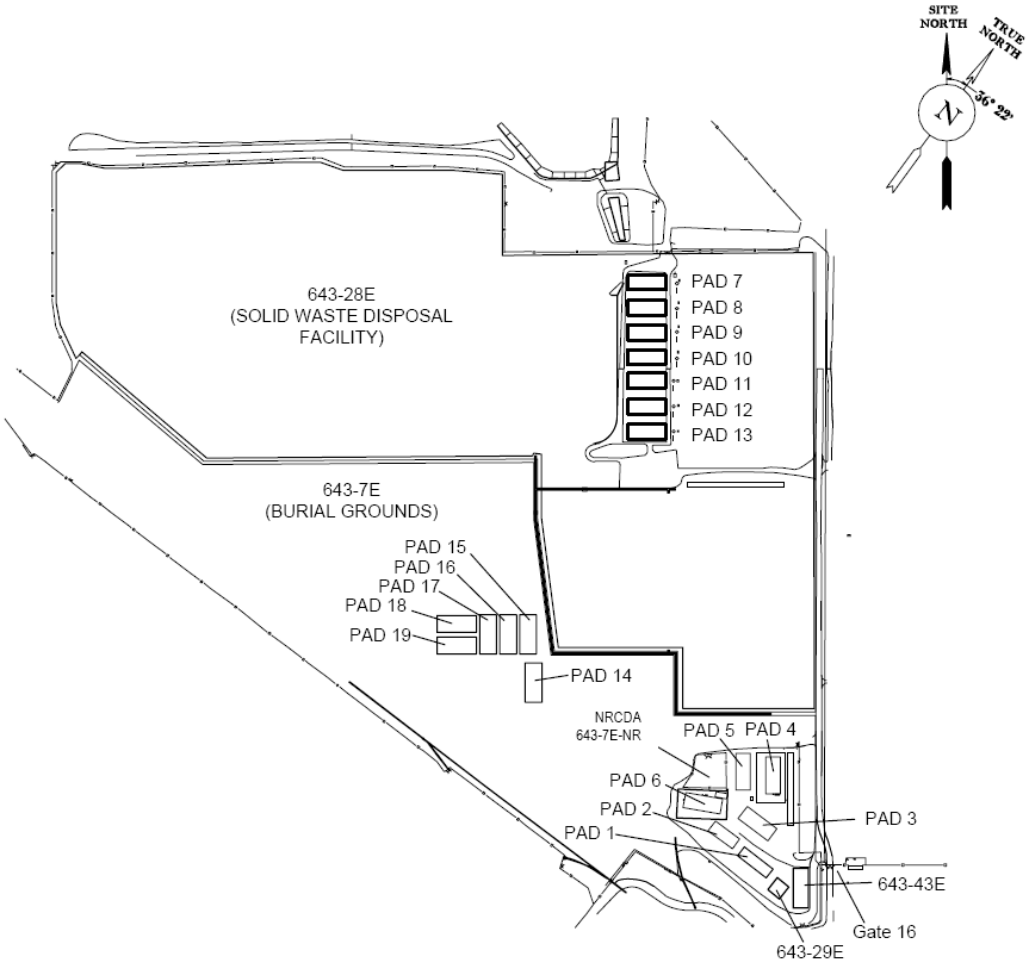


Figure 3-92. Location of TRU Pads 1 through 19 in the E-Area Solid Waste Disposal Facility (SWDF) (from WSRC 2007b)

3.4.5.2 Transuranic PAD 1 Waste and Containers

The Mound waste is contained in 39 plywood boxes and 739 steel drums, which are stored in 7 large concrete boxes and 85 concrete culverts on the SRS. Six of the concrete boxes and 83 of the culverts are on TRU Pad 1 (Figure 3-93). The LASL waste is stored in 241 steel drums, which are stored on TRU Pad 1 in 11 culverts. About 50% of the waste in the various container types is considered “burnable” waste (Towler 1979).

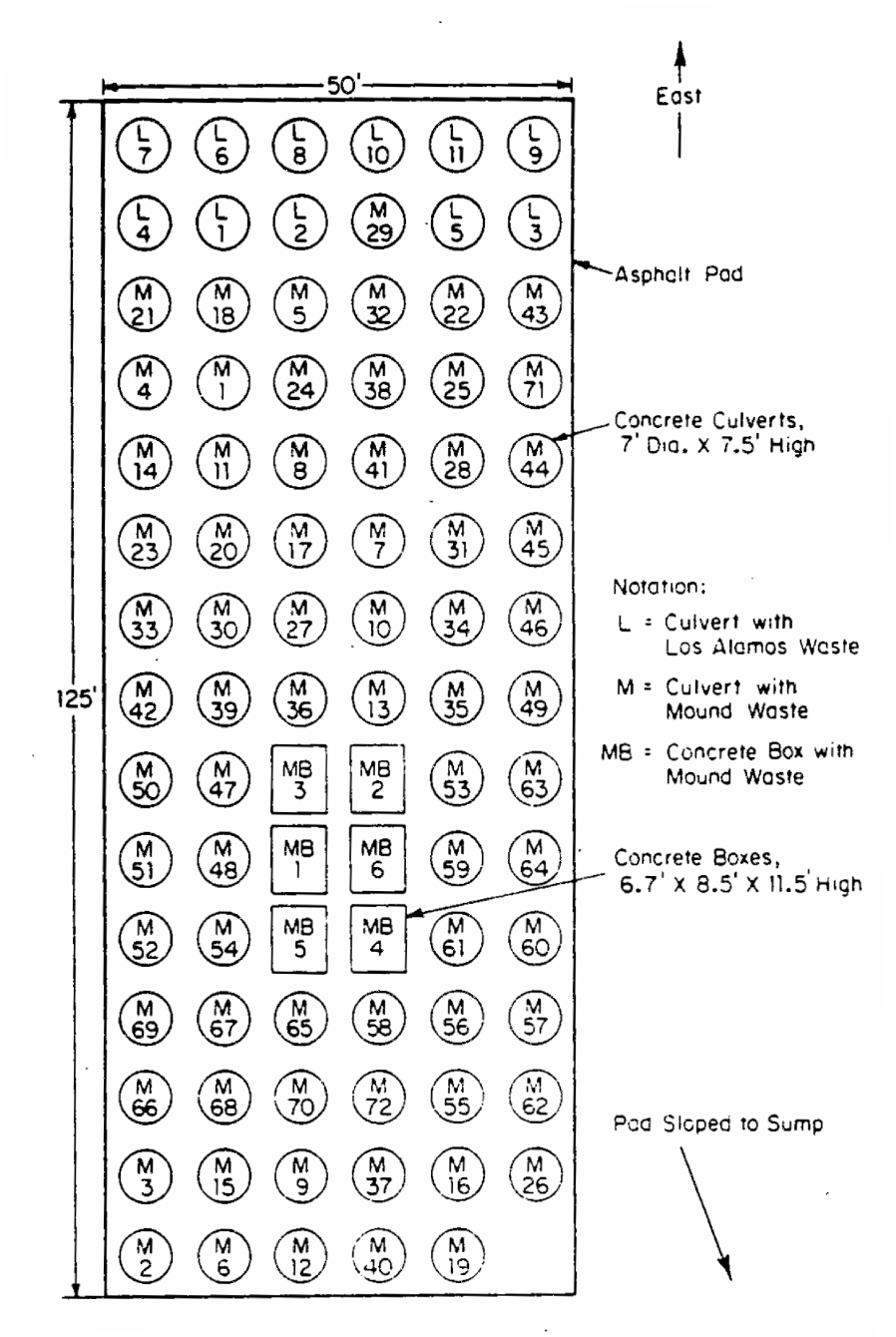


Figure 3-93. Layout of TRU Pad 1, Showing Position of Mound and Los Alamos Pu238 Waste (Towler 1979)

The concrete boxes on TRU Pad 1 holding the Mound waste are reinforced, with 6-in thick walls and lids. The 83 culverts on Pad 1 are constructed similarly to the concrete boxes, but are cylindrical, with 6-in thick reinforced concrete walls. Each culvert can hold fourteen 55-gal drums, or twenty four 30-gal drums. A very detailed accounting of the contents of the Mound and LASL waste packaging, and curie content by package, is provided by Towler (1979).

3.4.5.3 Transuranic Pad 1 Performance Assessment Requirements

Under DOE Order 435.1-1 (DOE 1999a), TRU waste is "radioactive waste containing more than 100 nanocuries (3700 Bq) of alpha-emitting TRU isotopes per gram of waste, with half-lives greater than 20 years …" excluding waste otherwise classified as high-level radioactive waste, waste that the Secretary of Energy determines does not need the degree of isolation required by 40 CFR Part 191 disposal regulations, or waste that the NRC has approved for disposal in accordance with 10 CFR Part 61 (DOE 1999b). Under the DOE Order, it is specified that waste that satisfies this definition of TRU waste shall be disposed in accordance with the standards of 40 CFR Part 191, *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes*. It is also assumed that the DOE Order 435.1-1 performance measures for disposal of LLW would be applicable to such TRU waste disposal at a location other than WIPP.

If TRU Pad 1 is classified as a TRU waste disposal facility, it is anticipated that both the 40 CFR Part 191 standards and the DOE Order 435.1-1 performance measures for disposal of LLW would be applicable. The 40 CFR Part 191 standards include:

- Containment Requirements
  - Less than a one in ten chance of greater than 100 curies of Pu238 reaching the accessible environment over 10,000 years per 1 million curies disposed.
  - Less than a one in a thousand chance of greater than 1,000 curies of Pu238 reaching the accessible environment over 10,000 years per 1 million curies disposed.
- Individual Protection Requirements
  - An all-pathways dose to any member of the public in the accessible environment less than or equal to 15 mrem/yr over 10,000 years.
- Groundwater Protection Requirements
  - Reasonable expectation that 10,000 years of undisturbed performance after disposal shall not cause the levels of radioactivity in any underground source of drinking water, in the accessible environment to exceed the limits specified in 40 CFR Part 141, as they existed on January 19, 1994.

The applicable DOE Order 435.1-1 performance measures are outlined in Table 3-24.

3.4.5.4 Transuranic Pad 1 Performance Assessment Results

The PA for TRU Pad 1 is in the early stages of production. Results are not yet available. The results, when available, will be utilized as a management tool to help evaluate whether or not the TRU Pad 1 waste should be sent to WIPP for disposal or remain at SRS for disposal.

3.4.5.5 Transuranic Pad 1 Performance Assessment Conceptual Models

The conceptual model of groundwater flow and transport for the E-Area TRU Pad 1 will be developed based on the evaluation of water elevation data from the well network in the E-Area vicinity, as well as other information derived from the extensive hydrogeologic characterization data collected in that area. The aquifers considered potentially contaminated by radionuclides leaching from the TRU Pad 1 will include the UTR aquifer unit, in which the water table resides, and the underlying Gordon aquifer. Contamination will not enter the deep Crouch Branch aquifer in this area because of the upward gradient that exists between this lower aquifer and the overlying Gordon aquifer near UTR stream (Figure 3-37 and WSRC 2008a).

Figure 3-84 and Figure 3-87 show the potentiometric surface for the water table (UTR) aquifer, the location of the groundwater divide, and the operational areas, including E-Area, in the GSA. Based on the information provided in these figures, the conceptual model of the saturated zone for the E-Area TRU Pad 1 will likely assume that flow in the water table unit under the facility is northward toward UTR, due to its location just north of the groundwater divide. The potentiometric surface for the Gordon aquifer unit (Figure 3-45) indicates a north-northwest flow direction under the entire GSA, due to the influence of the incisement of UTR in this aquifer (WSRC 2008a). Thus, radioactive contaminants in all aquifer units of concern with respect to releases from the E-Area TRU Pad 1 will eventually be discharged to UTR, unless they decay to stable species in transit.

3.4.6 F-Area Material Storage Performance Assessment

3.4.6.1 General Facility Information for the F-Area Materials Storage

The FAMS facility is slated for in-situ disposal (ISD) in accordance with the SRS End State Vision (DOE 2005). FAMS contains residual $Np237O_2$, $Pu238O_2$, and $Pu239O_2$. It has not yet been determined to what extent FAMS will be decontaminated. After decontamination it is possible that FAMS will still contain material, which would be classified as TRU waste (i.e., greater than 100 nCi/g). If it is determined that after decontamination that FAMS will still contain TRU waste, a PA will be prepared to evaluate the on-site disposal of TRU waste within FAMS against both the 40 CFR Part 191 standards and DOE Order 435.1-1 performance measures for disposal of LLW. Conformance with 40 CFR Part 191 is required for the disposal, after January 1994, of all DOE TRU waste at a location other than the WIPP. The following information generally describes the FAMS facility, and is largely taken from the 2003 Final Safety Analysis Report (FSAR) for this facility (WSRC 2003d).

The FAMS facility is located in the northern corner of F-Area (Figure 3-94), in the GSA (see Figure 3-82 for location of F-Area in the GSA), and thus is in the central part of the SRS. Building 235-F, which houses the FAMS facility, is at an elevation of over 300 ft above msl. The nearest significant stream is UTR, located about 0.75 mi northwest of the building, flowing at elevations less than 150 ft above msl (WSRC 2003d).

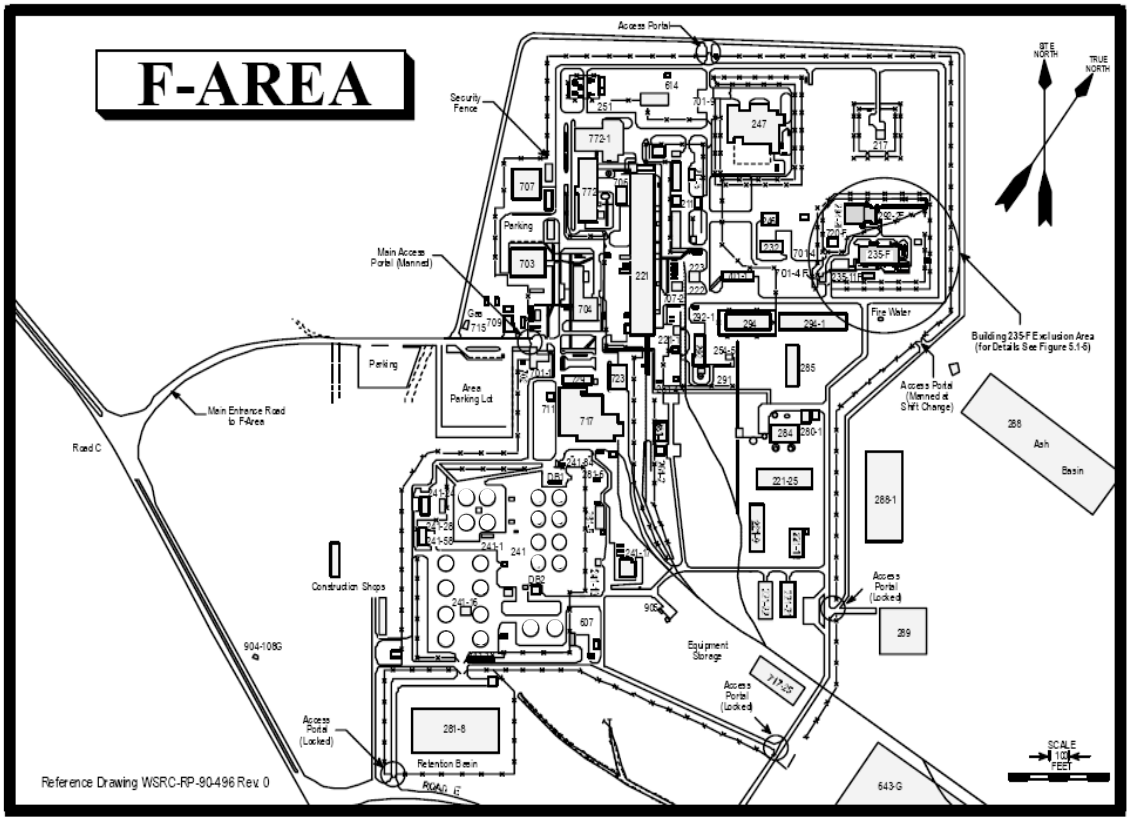


Figure 3-94. Location of F-Area Materials Storage within F-Area (from WSRC 2003d)

Building 235-F was constructed in the 1950s by E.I. du Pont de Nemours and Company (Du Pont) as part of the original SRS project and has been used for a variety of missions since that time. Two of the major historical missions located in the facility included the Special Products Fabrication Facility (SPFF) and the heat source production for the space program (WSRC 2006e). For the two decades preceding the decision to deinventory the facility in 2005, the predominant mission of the FAMS facility has been to provide for storage of excess plutonium materials.

During the mid to late 1960s, the FAMS facility housed the SPFF. The SPFF originally consisted of twenty-six gloveboxes, which included material blending equipment, furnaces, cold press, hot press, welding equipment, and non-destructive assay facilities. Prior to 1975, the SPFF supported programs processing U238, Np237, and Pu239 oxide. To permit new facility construction in the mid 1970's, the SPFF was decontaminated and decommissioned, except for the first nine (9) gloveboxes which were renamed the Actinide Billet Line (AB Line); a process facility critical to the heat source production mission (WSRC 2006e).

Research and fabrication activities associated with Pu238 encapsulation for NASA space missions were carried out beginning in the mid-1970s through 1983. The research mission of the FAMS facility ended in the 1980s (SRS 2007g). The facility's primary mission between the late 1980s and 2006 was the receipt, storage, and disbursement of plutonium bearing materials to support both the SRS and the DOE complex. Four vaults provided for storage and management of these materials. The facility also supported specific missions for the repackaging of these materials (WSRC 2003d, WSRC 2006e). The final shipment of special nuclear material left Building 235-F on Nov. 30, 2006 (SRS 2007h).

Building 235-F is a two-story, windowless, blast-resistant process building, constructed of reinforced concrete, that is approximately 222 ft long, 109 ft wide and 28 ft high (i.e., approximately 50,000 sq ft). It contains four process facilities (all of which have been deactivated) in addition to the four storage vaults. The facilities include the AB Line; the Plutonium Fuel Form (PuFF) Facility, including an (old) metallography glovebox line; the Plutonium Experimental Facility (PEF); and the New Metallography Laboratory (New Met Lab) was constructed to support operations in the PuFF Facility and PEF but was never started up. The New Met Lab has been modified to support miscellaneous operations and is renamed the Material Transfer Room (WSRC 2003d).

Largely in support of the heat source production mission, the AB Line was used to fabricate billets from NpO_2 and aluminum for extrusion into reactor targets, although it was also used to fabricate PuO_2 billets for use in other reactor missions. The NpO_2 targets were irradiated in the reactors to produce Pu238. The PEF was used to develop the manufacturing processes carried out on a large scale in the PuFF Facility, where heat source spheres and pellets were made from $Pu238O_2$ feed powder. The new Met Lab was constructed to provide analytical support to the heat source development and manufacturing operations (WSRC 2003d).

The four vaults in the FAMS facility were used to provide secure storage for nuclear materials in solid forms. The vaults were designed to allow storage of Department of Transportation/DOE-approved shipping containers. The new Met Lab was renamed the Material Transfer Room, and served to support vault operations when repackaging was required. No processing of nuclear materials was performed in the Material Transfer Room and no direct handling of nuclear material was performed during repackaging operations; the material being handled in the room was always confined within at least one closed container during all operations (WSRC 2003d).

There are no plans to activate any of these facilities, and the 2003 FSAR is based on the assumption that AB Line, PuFF Facility, and PEF continue to satisfy the restrictions associated with Cold Standby as defined in the Technical Safety Requirements (TSRs) discussed in that report. Most of the hazards in Building 235-F are those widely encountered in industrial settings; however, residual contamination remains in the PuFF Facility ($Pu238O_2$), PEF ($Pu238O_2$), and AB Line ($Np237O_2$ and $Pu239O_2$) (WSRC 2003d).

3.4.6.2 F-Area Materials Storage Performance Assessment Requirements

Under DOE Order 435.1-1 (DOE 1999a), TRU waste is "radioactive waste containing more that 100 nanocuries (3700 Bq) of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years …" excluding waste otherwise classified as high-level radioactive waste, waste that the Secretary of Energy determines does not need the degree of isolation required by 40 CFR Part 191 disposal regulations, or waste that the NRC has approved for disposal in accordance with 10 CFR Part 61 (DOE 1999b). Under the DOE Order, it is specified that waste that satisfies this definition of TRU waste shall be disposed in accordance with the standards of 40 CFR Part 191, *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes*. It is also assumed that the DOE Order 435.1-1 performance measures for disposal of LLW would be applicable to such TRU waste disposal at a location other than WIPP.

If TRU waste is to remain within FAMS after its ISD, it is anticipated that both the 40 CFR Part 191 standards and the DOE Order 435.1-1 performance measures for disposal of LLW would be applicable. The 40 CFR Part 191 standards include:

- Containment Requirements
  - Less than a one in ten chance of greater than 100 curies of Pu238 reaching the accessible environment over 10,000 years per 1 million curies disposed.
  - Less than a one in a thousand chance of greater than 1,000 curies of Pu238 reaching the accessible environment over 10,000 years per 1 million curies disposed.
- Individual Protection Requirements
  - An all-pathways dose to any member of the public in the accessible environment less than or equal to 15 mrem/yr over 10,000 years.
- Groundwater Protection Requirements
  - Reasonable expectation that 10,000 years of undisturbed performance after disposal shall not cause the levels of radioactivity in any underground source of drinking water, in the accessible environment to exceed the limits specified in 40 CFR Part 141, as they existed on January 19, 1994.

The applicable DOE Order 435.1-1 performance measures are outlined in Table 3-24.

3.4.6.3 F-Area Materials Storage Conceptual Models

Although not formally documented at this writing, the conceptual model of groundwater flow and transport for the FAMS will be developed similarly to that for the FTF PA, based on the evaluation of water elevation data from the well network in the F-Area vicinity, as well as other information derived from the extensive hydrogeologic characterization data collected in the area. The aquifers considered potentially contaminated by radionuclides leaching from the FTF include the UTR aquifer unit, in which the water table resides, and the underlying Gordon aquifer. Contamination will not enter the deep Crouch Branch aquifer in this area because of the upward gradient that exists between this lower aquifer and the overlying Gordon aquifer near UTR stream (Figure 3-37 and Section 3.2.10.2).

The mean 2003 potentiometric surface for the water table (UTR) aquifer in the GSA is shown in Figure 3-87. The groundwater divide relative to F-Area for that year is also shown on this figure. This figure indicates that the FAMS is north of the groundwater divide, and thus flow in the water table aquifer will likely be in the direction of UTR. The potentiometric surface for the Gordon Aquifer unit (Figure 3-44) indicates a north-northwest flow direction under the entire GSA, due to the influence of the incisement of UTR in this aquifer.

3.4.7 Hydrogeological Context of General Separations Area Performance Assessment Facilities in the Composite Analysis

In the Composite Analysis (CA), the various SRS PA results must be considered within the context of "all sources of radioactive material that may be left at the DOE site and may interact with the low-level waste disposal facility, contributing to the dose projected to a hypothetical member of the public from the existing or future disposal facilities" (DOE 1999b).

Within the GSA, groundwater discharging to UTR and Fourmile Branch flows under a number of facilities, including those in E-Area, F-Area, H-Area, S- Area and Z-Area (see Figure 3-95). Within E-Area, the MWMF and the ORWBG are currently the source of radioactive groundwater contaminants, as seen in the tritium plume maps arising from these areas in Figure 3-47 and Figure 3-48. These plume maps indicate the movement of shallow groundwater plumes located in the vicinity of the MWMF toward UTR, while those arising from the ORWBG flow toward Fourmile Branch to the south.

In addition to other E-Area facilities in the GSA, F-, H-, S-, and Z-Area are potential sources of radionuclide contamination to one or both of these two local creeks via contaminated groundwater, due largely to the presence of contained radioactive waste in these areas. The FTF and HTF partially overlie both groundwater basins (Figure 3-95) such that future leaching from these facilities after closure of the tanks could lead to discharge to both UTR and Fourmile Branch. In both F- and H-Areas, seepage basins and other uncontained sources also exist that are in various stages of remediation, which are potential sources of discharge to UTR (DOE 2005).

Outside of the GSA, there are also a number of potential sources of radioactive contamination to UTR, including sources in A-, M-, and R-Area. Some of the contaminated groundwater in A- and M-Areas is intercepted by UTR, or its tributary, Tims Branch. In R-Area, the R Reactor seepage basins have been a source of contamination of the groundwater that discharges to Mill Creek, which is also a tributary to UTR (DOE 2005). There are also a number of potential sources of radioactive contamination to Fourmile Branch outside of the GSA. These sources include those in C-Area, including the C Reactor seepage basins and those in N-Area (DOE 2005).

Once contaminants from the GSA reach UTR or Fourmile Branch, they are transported downstream and are eventually discharged to the Savannah River. The first POA for the CA (Phifer and Wilhite 2008) relevant to the PA facilities in the GSA occurs at the mouth of UTR and Fourmile Branch at the Savannah River (Figure 3-96). At the UTR mouth POA, radionuclides are derived from the GSA, A/M-Areas and portions of R-Area. At the Fourmile Branch mouth POA, radionuclides are derived from the GSA, C-Area, and N-Area. As GSA radionuclides are released to the Savannah River from the mouth of UTR and Fourmile Branch, they will also mix with those released from M-Area and D/T-Area to the Savannah River Floodplain Swamp (and Beaver Dam Creek for D-Area). Contaminants from Pen Branch and Steel Creek enter the Savannah River at the POA in the Savannah River at Steel Creek (Figure 3-96) and mix with the sources including the GSA. Radionuclide sources in K-Area, portions of N-Area and waste units associated with L-Area contribute to the releases from Pen Branch. At the POA located at LTR (Figure 3-96), GSA discharges will mix with radionuclides from the aforementioned sources, as well as those from portions of R-Area and P-Area (DOE 2005).

For areas of the SRS where the Crouch Branch Aquifer can be potentially contaminated by radionuclides (A/M-Area, K-Area, P-Area and R-Area), discharge from this lower aquifer to the Savannah River will result in mixing of radionuclides with those from the GSA. For A/M-Area, radionuclides discharged to the Savannah River via the Crouch Branch Aquifer will mix downstream with those from the GSA that enter the Savannah River at the mouth of UTR. For K-, P-, and R-Areas, discharges via Crouch Branch to the Savannah River will mix with radionuclides release from the GSA downstream of the mouth of UTR. Radionuclides discharge from Crouch Branch to the Savannah River, but not to SRS site streams.

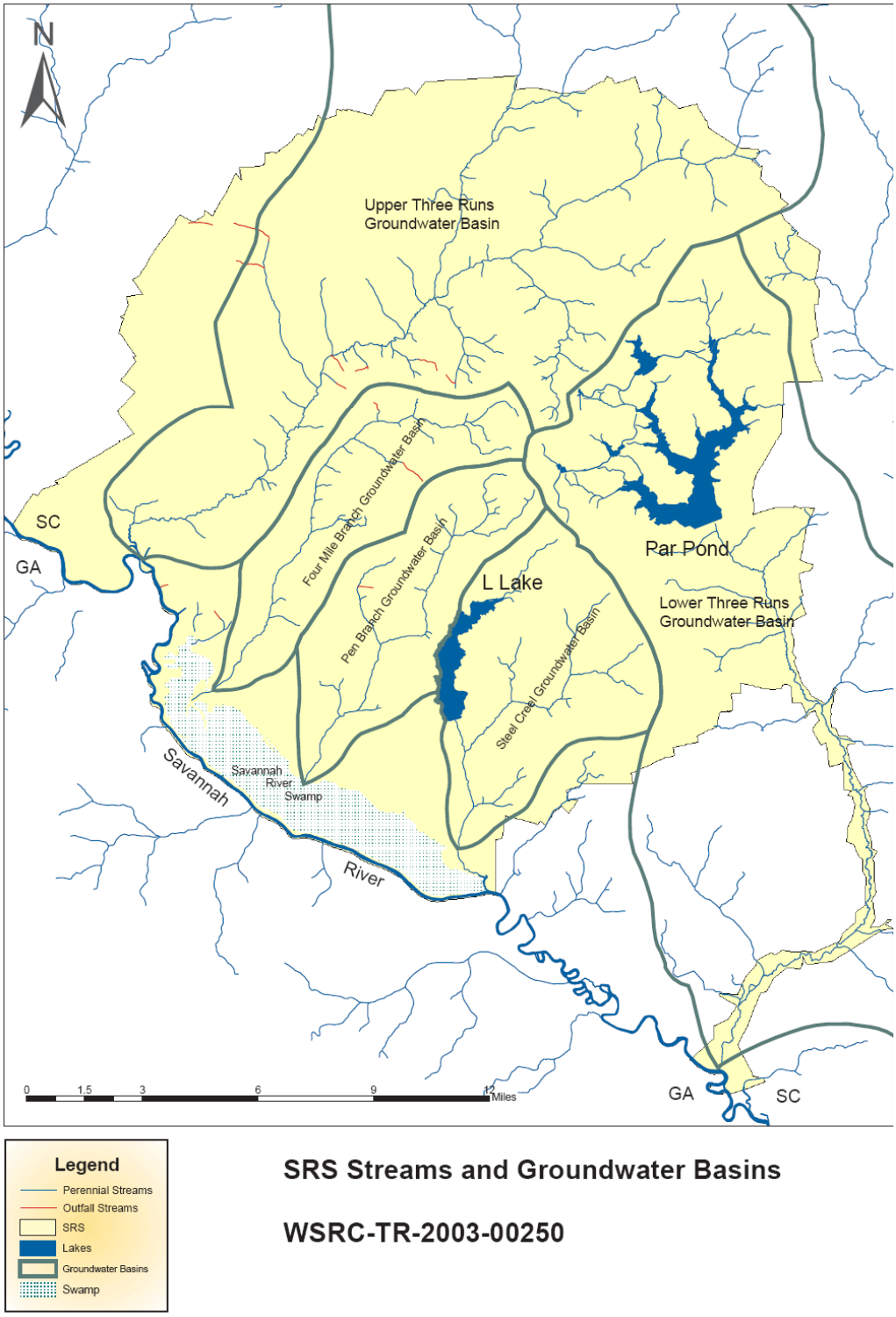


Figure 3-95. SRS Streams and Groundwater Basins

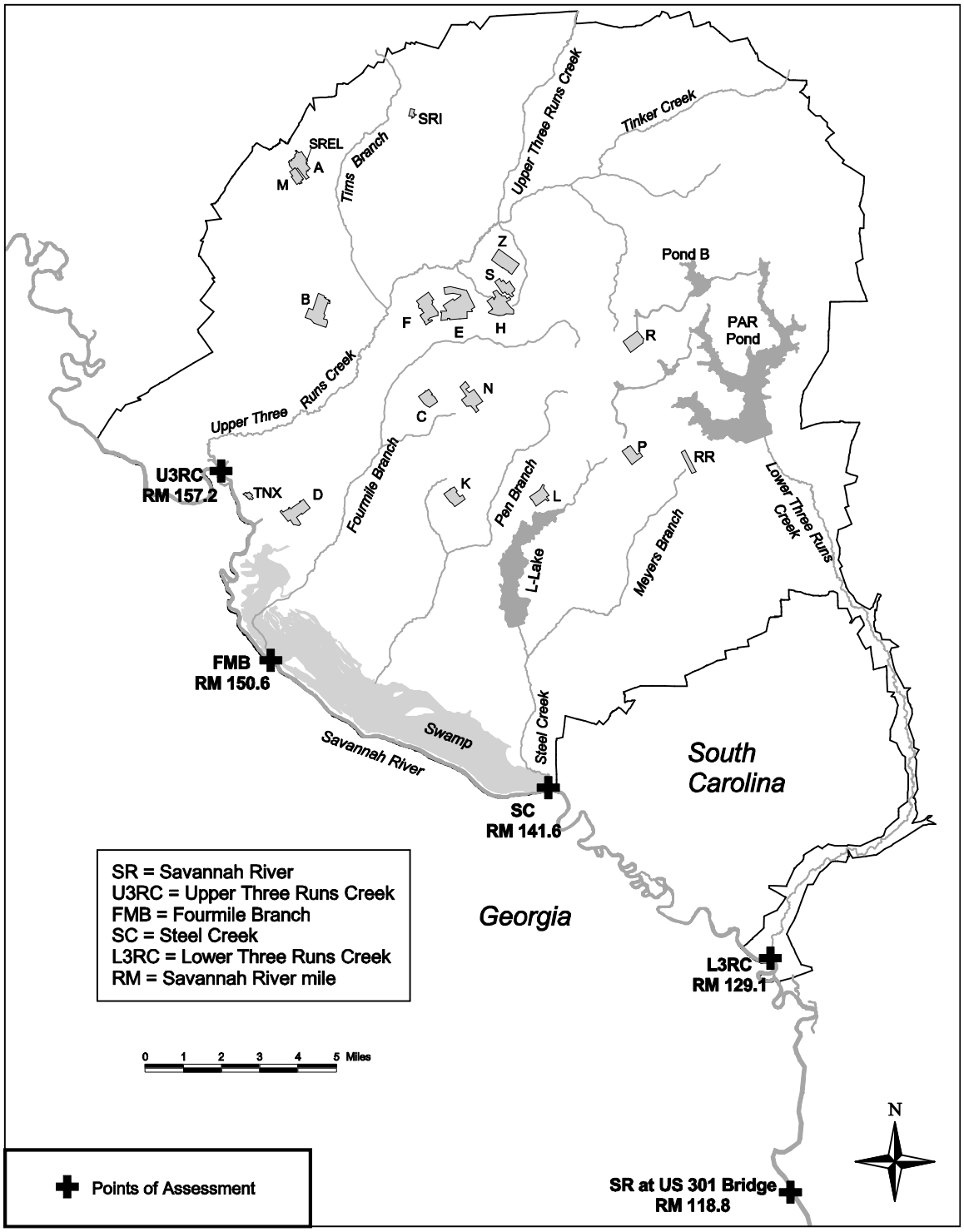


Figure 3-96. Location of SRS CA Points of Assessment (from Phifer and Wilhite 2008)

3.5 RELATED DOCUMENTATION

This section provides a summary of the site-specific documentation, including regulatory documentation, which may impact the analysis or conclusions of the CA. The following related documentation discussed in detail elsewhere within this CA has not been reproduced in this section but is referenced for completeness:

- SRS PAs are addressed in detail in Section 3.4.
- Specific CERCLA, RCRA, D&D, and radioactive liquid waste storage tank closure documentation from which inventory information was derived is discussed in detail in Section 8.0. However the SRS FFA (FFA 1993) discussed in Section 3.5.5 below forms the context under which all CERCLA, RCRA, and D&D actions are performed.
- Specific CERCLA, RCRA, D&D, and high level tank closure documentation that form the basis for the source release models utilized within the CA are discussed in detail in Section 9.0.

This section addresses the following site-specific documentation, which may impact the analysis or conclusions of the CA:

- SRS End State Vision (DOE 2005)
- SRS Comprehensive Plan/Ten Year Site Plan (SRNS 2009b)
- SRS Strategic Plan (DOE 2009)
- SRS Land Use Control Assurance Plan (WSRC 1999d)
- SRS Federal Facilities Agreement (FFA 1993)
- DOE Closure Planning Guidance (DOE 2004)
- SRS Groundwater Protection Program (WSRC 2004d)
- SRS Related NEPA Documentation
- SRS Safety Basis Documentation
- SRS Annual Environmental Monitoring Reports

3.5.1 United States Department of Energy Savannah River Site End State Vision, July 26, 2005 (DOE 2005)

The goal of the SRS End State Vision (DOE 2005) is to provide a vision for the disposal of all SRS DOE Office of Environmental Management (EM) nuclear material and waste hazards permanently, the decommissioning of all SRS EM facilities, and the remediation of all SRS inactive waste sites. This document does not specifically address facilities with enduring missions such as the SRS Tritium Facilities, which are under the jurisdiction of the DOE National Nuclear Security Administration (NNSA).

This document describes SRS hazards, planned end states for these facilities and waste sites, and SRS land use plans. The End State is assumed to occur in 2025. To address facility end states, an EM Integrated Deactivation and Decommissioning Plan is provided, which indicates the final disposition of each facility, either demolition or ISD. To address waste site end states, a Risk-Based End State Plan is provided, which indicates the planned remedial actions for each waste site. Like the SRS Comprehensive Plan/Ten Year Site Plan, which is addressed below, the SRS End State Vision is founded on the following basic assumptions about land ownership and use.

- The entire site will be owned and controlled by the federal government in perpetuity
- The property will be used only for industrial purposes
- Site boundaries will remain unchanged
- Residential use will not be allowed onsite

DOE solicited public input into the SRS End State Vision. The SRS End State Vision document contains an appendix that addresses public comments received, including recommendations/endorsement from the SRS Citizens Advisory Board (CAB).

3.5.2 Savannah River Site Comprehensive Plan/Ten Year Site Plan (SRNS 2009b)

The SRS Comprehensive Plan/Ten Year Site Plan (SRNS 2009b) provides the framework for integrating the SRS mission and vision with ecological, economic, cultural, and social factors in a regional context and to support decision-making for near-term and long-term use of the site. This plan reflects a cooperative working relationship between DOE and the state of South Carolina. The SRS Comprehensive Plan/Ten Year Site Plan describes the current site conditions, defines a vision for the evolution of the site over the next 50 years, outlines actions to achieve the vision, and guides the allocation of resources toward attainment of that vision. This plan provides guidance and direction for the future physical development of the site and provides a framework within which detailed analyses will be conducted to determine the courses of action required to reach optimum site configuration. The plan is based on specific assumptions. If these assumptions were to change, the plan would be updated to reflect the changed conditions. Chapter 5 of the SRS Comprehensive Plan/Ten Year Site Plan contains the Future Land Use Plan. Guidelines on which SRS land use is based include:

- Giving priority to protection of workers and the public
- Maintaining site security
- Maintaining other appropriate institutional controls
- Considering worker, public, and environmental risks, benefits, and costs
- Restricting the use of programs for units regulated under CERCLA or RCRA
- Maintaining existing SRS boundaries
- Continuing federal ownership of the land
- Prohibiting residential use of any SRS land

DOE considered stakeholder input on future use of the site property, as was solicited in development of the SRS End State Vision. Chapter 5 of the SRS Comprehensive Plan/Ten Year Site Plan describes future use of the site that was developed with input from public meetings, workshops, and consultation with state and federal agencies.

3.5.3 Savannah River Site Strategic Plan, May 2009 (DOE 2009)

The most recent SRS Strategic Plan (DOE 2009) was issued in May 2009. The SRS Strategic Plan has several purposes. The first is to establish the connection between the DOE Strategic Plan and SRS vision. This connection is achieved through provision of specific direction to SRS to support the mission and vision of DOE. The second function is to communicate the vision for the future of SRS. The final major purpose is to supply the link that communicates these requirements to all lower level Site plans and identifies the responsibilities each Site organization has in accomplishing DOE's vision.

In practical terms, the plan clearly presents the courses of action necessary to expedite cleanup of SRS and to continue the site's 50-year tradition of protecting the nation. To meet changing national needs, a new element of Site revitalization has also been added to this plan. Site revitalization is communicated by identifiable strategies for an orderly transition to future new missions, which demonstrates the site's readiness to assume new roles and provide vital services to America.

3.5.4 Land Use Control Assurance Plan for the Savannah River Site, January 8, 2008 (WSRC 1999d)

On April 21, 1998, the United States EPA Region IV Federal Facilities Branch implemented a policy to seek development of Land Use Control Assurance Plans (LUCAPs) by federal facilities that utilize land use controls (LUCs) as components of RCRA and CERCLA remedies. The Savannah River Operations Office recognizes that this EPA policy guidance does not carry the force of law as is established through rulemaking. Thus, as a matter of comity and cooperation and, most importantly, for the purpose of advancing the goal of protecting human health and the environment at the SRS, the LUCAP (WSRC 1999d) has been developed as a means to facilitate accomplishment of this shared objective.

Because the SRS is currently and will likely remain under Federal ownership, future site remedy decisions will take LUCs into account as appropriate. LUCs are defined as any restriction or control that limits the use of and/or exposure to any portion of that property. The term encompasses "institutional controls," such as those involved in real estate interests, government permitting, zoning and deed notices. LUCs may also include restrictions on access by means of engineered barriers such as fences or security guards. When LUCs are implemented as part of a site remedy decision, a plan will be prepared to ensure that LUCs will be maintained for as long as necessary to keep the selected remedy fully protective of human health and the environment.

The SRS LUCAP (WSRC 1999d) provides a listing of all waste units by watershed for which LUCs are part of the selected remedy. Additionally a unit-specific Land Use Control Implementation Plan (LUCIP) has been developed for each of these waste units and is provided as an appendix to the LUCAP.

To ensure that other land uses do not interfere with the LUCs specified for each waste unit per their LUCIP, each waste unit is identified in the SRS Site Use Program. The SRS Site Use Program coordinates the use of all lands and waters on the SRS. No use of land is undertaken without prior approval documented by a Site Use Permit. Each waste unit, including a 200-foot buffer zone, is identified on the Site Use maps to ensure that no other land uses are undertaken at the waste units.

3.5.5 Federal Facilities Agreement for the Savannah River Site, October 2007 (FFA 1993)

The FFA describes the requirements for environmental protection agreed to by the three concerned parties: DOE, EPA and SCDHEC. It addresses waste site remediation, facility decommissioning, and high-level radioactive waste tank system closure.

The general purposes of the FFA are to:

1. Ensure that the environmental impacts associated with past and present activities at the site are thoroughly investigated and that appropriate corrective/remedial action is taken as necessary to protect the public health and welfare and the environment
2. Ensure that all releases of hazardous substances, pollutants or contaminants as defined by CERCLA and all releases of hazardous wastes or hazardous constituents as defined by RCRA are addressed so as to achieve a comprehensive remediation of the site
3. Prevent, mitigate, or abate releases or threatened releases of hazardous substances from high-level radioactive waste tank systems prior to final corrective/remedial action at the site
4. Establish a procedural framework and schedule for developing, implementing, and monitoring appropriate response actions at the site in accordance with CERCLA, the National Contingency Plan (NCP), RCRA, and in accordance with applicable South Carolina law
5. Facilitate cooperation, exchange of information and participation of the Parties, and provide for effective public participation
6. Minimize the duplication of investigative and analytical work, and documentation, and ensure the quality of data management
7. Ensure that corrective/remedial action(s) at the site will be in compliance with Applicable or Relevant and Appropriate Requirements (ARARs)
8. Expedite response actions with a minimum of delay

9. Establish a basis for a determination that the DOE has completed the Remedial Investigation/Feasibility Study(s) (RI/FS), remedial design(s) and remedial action(s) at the site pursuant to CERCLA and applicable South Carolina laws
10. Integrate response actions under CERCLA and this Agreement with RCRA Facility Investigations/Corrective Measures Study(s) (RFIs/CMSs) and corrective measures now being conducted under RCRA and applicable South Carolina laws

The specific purposes of the FFA with SRS are to:

1. Identify operable units which are candidates for investigation and possible corrective/remedial action at the site
2. Establish requirements for the performance of an RFI/RI to determine fully the nature and extent of the threat to the public health or welfare or the environment caused by the release or threatened release of hazardous substances, pollutants or contaminants at the site
3. Establish requirements for the performance of an CMS/FS for the site to identify, evaluate, and select alternatives for the appropriate corrective/remedial action(s) to prevent, mitigate, or abate the release or threatened release of hazardous substances, pollutants or contaminants at the site in accordance with CERCLA and in compliance with ARARs identified pursuant to this Agreement
4. Identify the nature, objective and schedule of response actions to be taken at the site. Response actions at the site shall attain that degree of remediation of hazardous substances, pollutants or contaminants mandated by CERCLA and in compliance with ARARs identified pursuant to this Agreement
5. Implement the selected operable unit(s) and final corrective/remedial action(s) in accordance with CERCLA and in compliance with ARARs identified pursuant to this Agreement
6. Establish requirements for the SRS high-level radioactive waste tank system(s) identified in Appendix B to this Agreement to ensure structural integrity, containment and detection of releases, and source control pending final corrective/remedial action at the site
7. Meet the requirements of Section 120(e)(2) of CERCLA, 42 U.S.C. § 9620(e)(2), for an interagency agreement between the Parties
8. Provide for continued operation and maintenance following implementation of the selected corrective/remedial action(s)
9. Assure compliance with Federal and State hazardous waste laws and regulations for matters covered by this Agreement
10. Expedite remediation of the site to the extent necessary to protect human health and welfare and the environment

11. Provide for State involvement in the initiation, development, selection, and enforcement of corrective/remedial actions to be undertaken at the site, including the review of all applicable data as it becomes available and the development of studies, reports, and action plans; and to identify and integrate State ARARs in accordance with CERCLA

3.5.6 Closure Planning Guidance, June 2004 (DOE 2004)

The Closure Planning Guidance document (DOE 2004) defines the Department of Energy's program for cleanup of legacy waste in terms of safety expectations, performance metrics, management responsibility, and corporate business practices. The guidance and its performance expectations will remain in effect for 5 years, until 2009. Any changes will be based on the major uncertainties and external factors discussed in Section IV. Should changes occur, they will be explained and documented in the Office of EM's annual report.

More specifically, the document identifies the high-level work scope, schedule, and cost estimate for EM's plan to clean up legacy waste resulting from the cold war by fiscal year 2035. The planning elements function as the cleanup objectives, goals, and performance targets for each EM organization involved in the cleanup effort. Within this guidance, the Assistant Secretary for EM assigns responsibility to individuals for planning, developing, and implementing elements of the cleanup plan using the management principles presented in DOE Manual 413.3-1, *Project Management for the Acquisition of Capital Assets*. Additionally, major EM internal business initiatives in response to recommendations resulting from EM's Top-to-Bottom Review are described.

Key elements of EM's closure planning guidance include the following:

- Safety Emphasis - EM is totally responsible for safe conduct of operations at all its facilities, in all its cleanup activities, and in all supporting work initiatives. Although EM's path forward includes initiatives related to acquisition strategy, contract management, risk reduction, project management, performance targets, and performance oversight, the focus on safety is paramount. EM's emphasis on safety, including its initiative to continuously improve safety performance, is described.
- Performance Metrics - The cleanup work scope is described using 16 objectively determined performance metrics that provide a quantitative measure of progress on the cleanup program. These metrics, often referred to as corporate performance measures, are defined and quantified for individual cleanup sites.
- Cleanup Completion Dates - Major schedule commitments for the life-cycle of the cleanup program include the dates by which EM plans to complete cleanup activities at each of its field locations. These cleanup completion dates are identified.
- Annualized Cost Estimates - The EM plan completes cleanup of legacy waste for a total life-cycle cost estimate of $142 billion (constant FY 2003 dollars). Annualized cost estimates for the required cleanup performance are identified at the individual site level.

- Project Management Expectations - Although specific cleanup challenges, regulatory constraints, and decision-making processes vary from site to site, EM is steadfast in its expectations for identifying, planning, and accomplishing cleanup activities in accordance with the principles of DOE Manual 413.3-1. These principles of project management will be applied to all EM cleanup activities. EM Headquarters will provide oversight to ensure that the principles of DOE Manual 413.3-1 produce genuine value added to the cleanup effort, rather than unnecessary, bureaucratic compliance.
- EM's Corporate Business Practices - Responses to recommendations from the Top-to-Bottom Review are identified.
- Accelerated Risk Reduction - A technically based effort to accelerate risk reduction within the EM cleanup program is under way. The purpose of this effort is to review, analyze, and technically challenge major programmatic constraints within the existing EM cleanup program while improving safety performance and protecting workers, the public, and the environment. Major issues being explored for the potential to accelerate risk reduction include, but are not limited to, a nationwide acquisition strategy and breakthrough technology advances.
- Annual Report - Not only does the document present the quantitative elements of EM's life-cycle cleanup program, but its format is designed to support an annual assessment of the program's progress. Appendix 1 will be updated and published annually to report EM's progress against its life-cycle cleanup plan for each cleanup site, including the resources consumed and specific progress made. EM's progress toward implementing its internal initiatives will also be described.
- Individual Responsibility - Throughout the document, individual responsibility for cleanup activities, internal initiatives, and attainment of objectives for accelerated cleanup are assigned. As employees change their assignments as a result of career development, continuity of responsibility for these initiatives will be retained and documented in EM's annual report.

3.5.7 Savannah River Site Groundwater Protection Program, March 2004 (WSRC 2004d)

The SRS Groundwater Protection Program document (WSRC 2004d) contains a concise and systematic description of how the major components of the groundwater protection program are implemented at SRS. The major components of the program include:

- Source Control: Necessary source control associated with waste sites, facilities to be decommissioned, and the high-level radioactive waste tank systems are determined based upon the process outlined within the FFA (FFA 1993) and implemented in accordance with the resulting RCRA/CERCLA RODs.

- Monitoring: four primary types of groundwater monitoring are conducted at SRS as appropriate:
  - Short-term monitoring to demonstrate that no further action is required
  - Mixing zone monitoring to confirm mixing zone effectiveness and that groundwater corrective action is not required
  - Corrective action monitoring to assess the effectiveness of corrective actions and ensure that groundwater contaminant levels are maintained below groundwater protection standards after completion of corrective measures, and
  - Site perimeter monitoring to ensure that contaminated groundwater does not move offsite and to monitor seeplines
- Corrective Action: Necessary groundwater corrective actions are primarily driven by RCRA or CERCLA and are determined based upon the process outlined within the FFA (FFA 1993) and implemented in accordance with the resulting RCRA/CERCLA RODs.
- Well Abandonment: Wells are prioritized for abandonment based on the threat they pose to groundwater resources. The factors examined in characterizing the threat include proximity to contamination, depth, construction method, casing material, and age. Another factor in prioritizing abandonments is a desire to integrate the abandonments as much as practical into the site's overall facilities disposition plan.

3.5.8 NEPA Documentation Relevant to the CA

NEPA documentation is used in the early planning stages of major federal projects to analyze at a very high level several options for those actions. The CA analyzes the final stage of the operation of the Savannah River Site. For the most part, the CA relies on documentation that is specific to the actual facility or action selected for implementation, such as safety documentation or closure plans. While many actions at SRS have been examined in NEPA documentation, the only cases where information from NEPA documentation would be used as the primary source of data in the CA would be for facilities or actions that are in the planning stage, where no more specific information was available.

In this CA, only the planned MOX Fuel Fabrication Facility fell into this category. The document used was the *Environmental Impact Statement on the Construction and Operation of a Proposed Mixed Oxide Fuel Fabrication Facility*, NUREG-1767 (NRC 2005).

Two other Environmental Impact Statement (EIS) documents are of interest to this CA. Both cover a broad scope of major actions which will be carried out over a number of years. These are briefly described below.

In 1987 DOE issued a programmatic and project-specific EIS (DOE/EIS-0120) to support the selection of a programmatic waste management strategy for SRS and to consider the environmental impacts of several specific projects, including closure and cleanup of active and inactive waste management sites; establishment of new waste storage and disposal facilities; and alternative means of discharging disassembly basin purge water from SRS reactors (DOE 1987). A ROD was issued in March 1988.

In 2002 DOE issued an EIS for the closure of the high level waste tanks at SRS (DOE/EIS/0303). The preferred alternative was identified as Stabilize Tanks—Fill with Grout. This selection is used to guide development and implementation of closure of the high-level radioactive waste tanks and associated equipment at the SRS. Following bulk waste removal, DOE will clean the tanks if necessary to meet the performance objectives contained in the General Closure Plan and the tank-specific Closure Module, and then fill the tanks with grout (DOE 2002). A ROD was published in August 2002.

Table 3-30 lists all of the Environmental Impact Statements that have been prepared that concern the Savannah River Site.

Table 3-30. EIS Documents Prepared for the SRS

Title and Publication Date	Document Number
Long-Term Management of Defense High-Level Radioactive Wastes, November 1979	DOE/EIS-0023
Waste Isolation Pilot Plant, September 1997	DOE/EIS-0026
Waste Management Operations-Supplement to ERDA-1537, Double-Shell tanks for Defense High-Level Radioactive Waste Storage, April 1980	DOE/EIS-0062
Defense Waste Processing Facility & Salt Processing Alternatives, February 1982	DOE/EIS-0082
Defense Waste Processing Facility, Supplemental November 1994	DOE/EIS-0082-S
Savannah River Site Salt Processing Alternatives Final Supplemental Environmental Impact Statement, June 2001	DOE/EIS-0082-Salt Processing Alt.
L-Reactor Operation, May 1984	DOE/EIS-0108
Waste Management Activities for Groundwater Protection at SRS, December 1987	DOE/EIS-0120- FINAL
Alternate Cooling Water Systems, February 1988	DOE/EIS-0121- FINAL

Table 3-30. EIS Documents Prepared for the SRS (continued)

Title and Publication Date	Document Number
Siting, Construction and Operation of New Production Reactor Capacity, April 1991	DOE/EIS-0144 Draft Summary
Continued Operation of K-, L-, and P-Reactors, December 1990	DOE/EIS-0147
Final Programmatic Environmental Impact Statement (PEIS) for Tritium Supply and Recycling, November 1995	DOE/EIS-0161
Final Waste Management Programmatic Environmental Impact Statement For Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste, May 1997	DOE/EIS-0200
Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs, April 1995	DOE/EIS-0203-Summary
Savannah River Waste Management, July 1995	DOE/EIS-0217
Proposed Nuclear Weapons Nonproliferation Policy Concerning FRR SNF, February 1996	EIS-0218-F
F-Canyon Plutonium Solutions, December 1995	DOE/EIS-0219
Interim Management of Nuclear Materials, October 1995	DOE/EIS-0220
Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement (Summary Only), December 1996	DOE/EIS-0229
Final Programmatic Environmental Impact Statement for Stockpile Stewardship and Management, November 1996	DOE/EIS-0236
Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement, June 1996	DOE-EIS-0240
Shutdown of River Water System at the Savannah River Site, May 1997	DOE/EIS-0268
Accelerator for Production of Tritium at the Savannah River Site, March 1999	DOE/EIS-0270

Table 3-30. EIS Documents Prepared for the SRS (continued)

Title and Publication Date	Document Number
Construction & Operation of a Tritium Extraction Facility at the Savannah River Site, March 1999	DOE/EIS-0271
Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Site, August 1998	DOE/EIS-0277
Spent Nuclear Fuel Management Final Environmental Impact Statement, March 2000	DOE/EIS-0279
Surplus Plutonium Disposition Final Environmental Impact Statement, November 1999	DOE/EIS-0283 DOE/EIS-0283-SA2 Waste Solidification Bldg.
Production of Tritium in a Commercial Light Water Reactor, March 1999	EIS-0288 Summary
The Savannah River Site High-Level Waste Tank Closure Final Environmental Impact Statement, May 2002	DOE/EIS-0303 Final
Barnwell Nuclear Fuel Plant, July 1974	Docket No. 50-332
Construction and Operation of a Mixed Oxide Fuel Fabrication Facility, January 2005	NUREG-1767

3.5.9 Safety Basis Documentation

SRS Safety Basis Documents describe and analyze activities at SRS such as receipt, storage, processing, and handling of radiological or chemical materials. There is a separate document for each processing building that handles such material. These reports document the safety authorization basis for managing these activities and demonstrate that these activities have no unacceptable radiological or chemical risks to the offsite public, site workers, or the environment.

Each report provides an accident analysis presented as a deterministic analysis that utilizes a graded approach in which the consequences are calculated based on maximum facility inventories. Offsite doses from all the accident scenarios are each within the SRS evaluation guidelines. This approach is bounding, that is conservative in nature where a more realistic analysis would be expected to yield lower consequence estimates. In addition, this type of analysis eliminates questions concerning uncertainties, and is considered a more cost-effective approach.

Fire, explosion, loss of confinement, direct radiological exposure, external hazards (such as vehicle or handling accidents), and natural phenomena (seismic and wind) are potential initiating events for public and worker exposures and environment damaging releases. The hazards associated with these broad categories were evaluated to provide a basis for selecting a set of bounding accidents to be analyzed, and the identification of Safety Systems, Structures, and Components.

For a number of facilities information contained in safety documentation was used to estimate the residual inventory to remain in those facilities at site closure. The Safety Basis documents used for this purpose in this CA are:

- Central Laboratory Facility – Buildings 772-F, 772-1F and 772-4F Safety Analysis Report (WSRC-SA-96-26 Rev. 4 November 2006 (WSRC 2006f))
- Safety Analysis – 200 F-Area Savannah River Site Building 235-F Final Safety Analysis Report (U) (WSRC-RP-89-575 Rev. 3 January 2003) [UCNI] (WSRC 2003d)
- Safety Analysis Report Savannah River Site FB-Line, F-Canyon, FA-Line and Outside Facilities F-Canyon Complex Safety Analysis Report (WSRC-SA-2001-00004 Rev. 5 July 2007) (WSRC 2007e)
- HB-Line Safety Analysis Report (U) (WSRC-SA-2001-0009 August 2006) [UCNI] (WSRC 2006g)
- Safety Analysis Report Savannah River Site H-Canyon & Outside Facilities, H-Area H-Canyon Safety Analysis Report (WSRC-SA-2001-00008 June 2007) (WSRC 2007f)
- Auditable Safety Analysis for Receiving Basin for Offsite Fuel and the Resin Regeneration Facility (U) (WSRC-TR-2006-00421 April 2007) (WSRC 2007g)
- SRNL Technical Area Documented Safety Analysis (WSRC-SA-2 Rev. 4 September 2007) (WSRC 2007h)
- Tritium Extraction Facility Safety Analysis Report Addendum (WSRC-SA-1-2-Vol-4 Rev. 2 July 2007) [UCNI] (WSRC 2007i)
- Tritium Facilities Safety Analysis Report (WSRC-SA-1-2-Vol 1 Rev. 9 September 2007) [UCNI] (WSRC 2007j)

3.5.10 Annual Environmental Monitoring Reports

Environmental monitoring is conducted extensively within a 2,000-square-mile network extending 25 miles from SRS, with some monitoring performed as far as 100 miles from the site. The area includes neighboring cities, towns, and counties in Georgia and South Carolina. Thousands of samples of air, rainwater, surface water, drinking water, groundwater, food products, wildlife, soil, sediment, and vegetation are collected by SRS and state authorities and analyzed for the presence of both radioactive and nonradioactive contaminants.

Compliance with environmental regulations and with DOE orders related to environmental protection provides assurance that onsite processes do not impact the public or the environment adversely. Such compliance is documented each year in a series of Annual Environmental Monitoring Reports. The Environmental Report for 2007 (Mamatey 2008) is the most recently published document in this series.

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4.0 ANALYSIS CRITERIA

A Composite Analysis (CA) is required by Department of Energy (DOE) Order 435.1, Radioactive Waste Management (DOE 1999a), in order to provide a reasonable expectation that DOE low-level waste (LLW) disposal, high-level waste tank closure, and transuranic (TRU) waste disposal, in combination with Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), Resource Conservation and Recovery Act (RCRA), and deactivation and decommissioning (D&D) actions, will not result in the need for future remedial actions in order to ensure radiological protection of the public. The Savannah River Site (SRS) CA is an analysis of the projected dose to a hypothetical future member of the public resulting from the E-Area Low Level Waste Facility (ELLWF), the Saltstone Disposal Facility (SDF), the closed F- and H-Area Tank Farms (FTF and HTF), E-Area TRU Pad 1 disposal, and F-Area Material Storage (FAMS) D&D along with all other SRS radioactive source locations, which may interact with radionuclide transport from these facilities (DOE 1999b, DOE 2001, DOE 2008). The CA evaluates the dose to the public at Points of Assessment (POAs), which are selected based upon the site's land use plans, over a specified Assessment Period (AP) after disposal facility and tank closure and/or all DOE site operations have ceased.

The initial SRS CA was prepared in 1997 (WSRC 1997), and the SRS CA provided herein is the first revision to the 1997 SRS CA. As a preliminary step to revision of the SRS CA, the criteria defined by the DOE as applicable to production of a CA and DOE comments received on the 1997 CA were reviewed and matrices were developed to ensure that all relevant requirements were reviewed, evaluated for applicability, and considered in the revised CA, as appropriate. This review, documented fully in Phifer et al. (2008), addressed Task 4.2 of the SRS Composite Analysis Program Plan (Phifer and Cook 2007). Task 4.2 specifies that the review of criteria and review comments will consult, at a minimum, the following documents: the 1997 CA entitled, "Composite Analysis E-Area Vaults and Saltstone Disposal Facilities" (WSRC 1997); the Low-Level Waste Disposal Facility Federal Review Group (LFRG) Review Team report (DOE 1998); the 1999 CA Addendum (Cook et al. 1999); the "Implementation Guide for use with DOE M 435.1-1" (DOE 1999b); the "Format and Content Guide for U.S. Department of Energy Low-Level Waste Disposal Facility Performance Assessments and Composite Analyses" (DOE 1999c); the 2002 CA Addendum (Cook et al. 2002); the report entitled, "Resolution of Final Composite Analysis Review Team Comments" (Cook 2003); the "Low-Level Waste Disposal Facility Federal Review Group Manual" (DOE 2008); and the 2007 CA Maintenance Program (WSRC 2007a).

The criteria matrices developed in Phifer et al. (2008) are represented in five tables, the first of which is included as Appendix B of this CA. This table shows where the SRS CA Core Team believes that the CA criteria listed in DOE M 435.1 and the LFRG Manual are met, thus facilitating the LFRG review of the CA. The remaining four tables were used by the CA Core Team during implementation of the various CA tasks, as guidance in ensuring that relevant requirements and previous review comments were appropriately considered.

Section 4.1 describes the performance measures relevant to CAs. Tasks 4.2.9 and 4.2.10 of the SRS Composite Analysis Program Plan (Phifer and Cook 2007) outline the requirements to establish appropriate POAs and an AP for the SRS CA revision. The bases, derived from consideration of the performance measures, for selecting the POAs and AP to be utilized for production of the SRS CA revision are discussed in Sections 4.2 and 4.3, respectively. The text in these two sections is largely excerpted from Phifer and Wilhite (2008).

4.1 PERFORMANCE MEASURES

DOE M 435.1-1 requires that, "For disposal facilities that received waste after September 26, 1988, a site-specific radiological composite analysis shall be prepared and maintained that accounts for all sources of radioactive material that may be left at the DOE site and may interact with the low-level waste disposal facility, contributing to the dose projected to a hypothetical member of the public from the existing or future disposal facilities. Performance measures shall be consistent with DOE requirements for protection of the public and environment and evaluated for a 1,000 year period following disposal facility closure." (DOE 1999c).

According to the "Format and Content Guide for U.S. Department of Energy Low-Level Waste Disposal Facility Performance Assessments and Composite Analyses" (DOE 1999c), "…the composite analysis evaluates the all-pathways dose to a hypothetical future member of the public from all sources considered in the analysis". Table 4-1 provides a summary of the performance measures and corresponding points of compliance for composite analyses. As shown in Table 4-1, two performance measures are considered – a dose limit and a dose constraint. The primary dose limit for radiological protection of the public from all sources and all pathways is 100 mrem in a year (DOE 1990). However, a dose constraint of 30 mrem in a year (DOE 1999c) is established for the composite analysis to ensure that the sources analyzed do not use an extraordinary portion of the primary dose limit. If the results of the composite analysis exceed the primary dose limit, mitigating actions must be taken before the dose limit is exceeded; if the results do not exceed the primary dose limit but do exceed the dose constraint, mitigating actions should be considered, but may not actually be taken. In applying these performance measures, it is appropriate to assume that DOE will control the land on which LLW is disposed and any surrounding land until the land can be safely released pursuant to DOE Order 5400.5 (or 10 CFR 384 when promulgated) [DOE 1990], or transferred to another authorized party. Therefore, the analyses performed for the composite analysis (i.e., the calculations performed to generate a result for comparison with a performance measure) should be prepared consistent with site plans for future land use and control. This will affect the selection of the assumed point of future public access and exposure used in the analyses."

The SRS environmental monitoring and associated Annual Environmental Reports will continue to be used to demonstrate current compliance with the DOE Order 5400.5 (DOE 1990) requirements for radiological protection of the public (i.e., 100 mrem/yr primary dose limit). Reference to this program and its past monitoring results, particularly for tritium, will be included in the CA for comparative purposes.

Table 4-1. Composite Analysis Components, Measures, and Points of Assessment

Component	Measure	Point of Assessment
All pathways	DOE primary dose limit of 100 mrem in a year (DOE Order 5400.5 (DOE 1990))	The point of highest projected dose at, or beyond, the projected boundary of land controlled by DOE. A more conservative (nearer the LLW disposal facility) boundary should be selected if land-use plans are uncertain.
All pathways	Composite Analysis dose constraint of 30 mrem in a year (DOE 1999c)	The point of highest projected dose at, or beyond, the projected boundary of land controlled by DOE. A more conservative (nearer the LLW disposal facility) boundary should be selected if land-use plans are uncertain.

4.2 POINTS OF ASSESSMENT

4.2.1 Point of Assessment Selection Basis

The DOE Radioactive Waste Management Manual (DOE 1999d) and DOE Implementation Guide (DOE 1999b) require that a composite analysis assess the potential dose to a hypothetical future member of the public. The LFRG manual (DOE 2008) states the following regarding the POA for the hypothetical future member of the public:

The POA "is the publicly accessible point of maximum impact reasonably expected for future members of the public for the time period of assessment. The point of assessment is justified and is supported by land use plans or reasonably conservative assumptions that are justified. Changes in the point of assessment as a function of time are justified."

The DOE format and content guide for Composite Analyses (DOE 1999c) provides further clarification regarding the POA. This guide states that the POA is to be outside the boundary of land controlled by DOE. Additionally it states that selection of the POA and any shrinkage in the land controlled by DOE over time should be consistent with site-specific land use and facility plans. Further it states that sensitivity and uncertainty analyses conducted for the CA should include land use controls and "the impacts of reasonable alternative uses of land outside those areas assumed to be permanently controlled by DOE for radiation protection of the public."

Appendix A of Phifer and Wilhite (2008) provides a listing of pertinent documents, including those referenced above, that provide the basis for POA selection, along with extracts from the documents concerning the POA.

4.2.2 Point of Assessment Selection Criteria

From the POA basis provided in Section 4.2.1 (and Appendix A of Phifer and Wilhite 2008), the following criteria for selecting POAs for this SRS CA have been established:

- The POA(s) will be the location(s) of maximum radiation dose to a hypothetical future member of the public outside the SRS land expected to be controlled by DOE, resulting from radionuclide migration from the ELLWF, SDF, FTF and HTF, E-Area TRU Pad 1, and FAMS (i.e., the facilities for which a CA is required). These will be locations where the public can reasonably be expected to have access to radionuclides, which have migrated from these SRS facilities.
- The dose at the POA(s) will also include the contribution from all other SRS radionuclide source locations, which are determined to have a remaining radionuclide inventory after all DOE site operations have ceased, which have not been screened out, and which could interact with the radionuclide migration from ELLWF, SDF, FTF and HTF, E-Area TRU Pad 1, and FAMS.
- The POA(s) will be selected based upon the SRS land use plans (DOE 2005, SRNS 2009). Changes in the locations of the POA(s) over the AP (i.e., with time) may be made (i.e., if the size of SRS shrinks), if justified by the SRS land use plans.
- CA sensitivity and uncertainty analysis will consider alternative POA(s), if justified by the SRS land use plans or other considerations. Such analysis may also consider alternative land uses outside the SRS land controlled by DOE.
- Various transport pathways (i.e., water [vadose zone, groundwater, and stream], air, and biosphere, etc.) will need to be considered for the establishment of both the POAs and alternative POAs, as necessary.

Using the above criteria POA(s) would only be established based upon radionuclide migration from ELLWF, SDF, FTF and HTF, E-Area TRU Pad 1, and FAMS (i.e., essentially radionuclide migration from facilities within the General Separations Area [GSA]). Radionuclide migration from other SRS radionuclide source locations would not be considered solely for the establishment of POA(s), but would be considered as they interact with migration from the ELLWF, SDF, FTF and HTF, E-Area TRU Pad 1, and FAMS.

4.2.3 Savannah River Site Land Use Plans

The following two documents constitute the current SRS Land Use Plan:

- Savannah River Site Comprehensive Plan/Ten Year Site Plan FY 2010-2019 (SRNS 2009)
- Savannah River Site End State Vision (DOE 2005)

4.2.3.1 Savannah River Site Comprehensive Plan/Ten Year Site Plan (SRNS 2009)

The Savannah River Site Comprehensive Plan/Ten Year Site Plan (SRNS 2009) has been developed with guidance and feedback from the State of South Carolina and community groups such as the SRS Citizens Advisory Board, the Central Savannah River Area (CSRA) Planners Group, the Savannah River Regional Diversification Initiative, Citizens for Nuclear Technology Awareness, local chambers of commerce, and local economic development organizations. The plan defines a vision for the evolution of SRS over the next 50 years and for the long-term stewardship of the site.

The plan (SRNS 2009) assumes that SRS will remain a vital national asset and anticipates the expansion of missions at the site. According to the plan the site boundaries will remain unchanged and land use will not significantly change over the next 50 years. However, over this time period the industrial footprint of the site will shrink consistent with the site's division into three land use planning zones: Site Industrial, Site Industrial Support, and General Support (Figure 4-1). As new nuclear and/or non-nuclear heavy industrial missions (i.e., hazardous and radiological facilities) are funded, such facilities will be located within the central Site Industrial Zone. Additionally, rather than upgrade or replace aging facilities outside the central industrial zone, these facilities will be taken out of service as their useful life ends. The Site Industrial Support and General Support Zones will accommodate uses of decreasing intensity, particularly as they approach the site's boundaries. The Site Industrial Support Zone will support relatively low impact activities such as administrative, research and development, and warehousing. The General Support Zone will support very low impact activities such as ecological research and natural resource management. This will tend to reconfigure SRS land use and consolidate more intensive activities at the center of the site to minimize the effect on surrounding communities, produce less impact to ecosystems, maintain controlled site access, and ensure the integrity of the established safety and security buffer.

Under this plan (SRNS 2009) DOE ownership of SRS will extend into the foreseeable future with maintenance of site security and other institutional controls in all zones, including environmental monitoring. Unrestricted public access and residential land use will be prohibited, thus minimizing potential liability from public ownership of land. The extent of environmental cleanup and D&D will be commensurate with this land use plan (i.e., cleanup to industrial (not residential) use). Restricted land use associated with CERCLA, RCRA, and D&D units will be instituted as needed (as documented within the SRS Land Use Control Assurance Plan (LUCAP), land use restrictions will remain for as long as the residual materials present a hazard [WSRC 1999]). Additionally the site's designation as a National Environmental Research Park (NERP) will continue.

Finally long-term stewardship of SRS is required by this plan (SRNS 2009). The plan recognizes that long-term stewardship activities will be required for the long-lived contaminants and wastes that will remain in place at SRS, precluding unrestricted use of the site. Therefore ownership of the entire SRS by the federal government will continue with unchanged boundaries. DOE will continue institutional controls, including ongoing monitoring and maintenance, to ensure that land use restrictions are maintained for as long as necessary to ensure the long-term effectiveness of the selected remedial, closure, and decommissioning actions.

4.2.3.2 Savannah River Site End State Vision

The Savannah River Site End State Vision (DOE 2005) describes current SRS conditions and the planned end states for disposing all nuclear material and waste hazards permanently, decommissioning all facilities, and remediating all inactive waste sites. Institutional controls that include access restrictions and an inspection, maintenance, and monitoring program will be instituted for all facilities and sites where waste is left in place. In regards to SRS land use the SRS End State Vision states that the entire site will be owned, controlled, and maintained by the federal government, most likely DOE, in perpetuity, as established by Congress. Site boundaries will remain unchanged and the site will be used for industrial purposes for future DOE and non-DOE missions. Land use zones have been established that concentrate future industrial land use operations toward the center of the site to form a central Site Industrial zone for continuing missions. The central Site Industrial zone is surrounded by concentric Site Industrial Support and General Support land use zones (Figure 4-1). Residential use will not be allowed onsite.

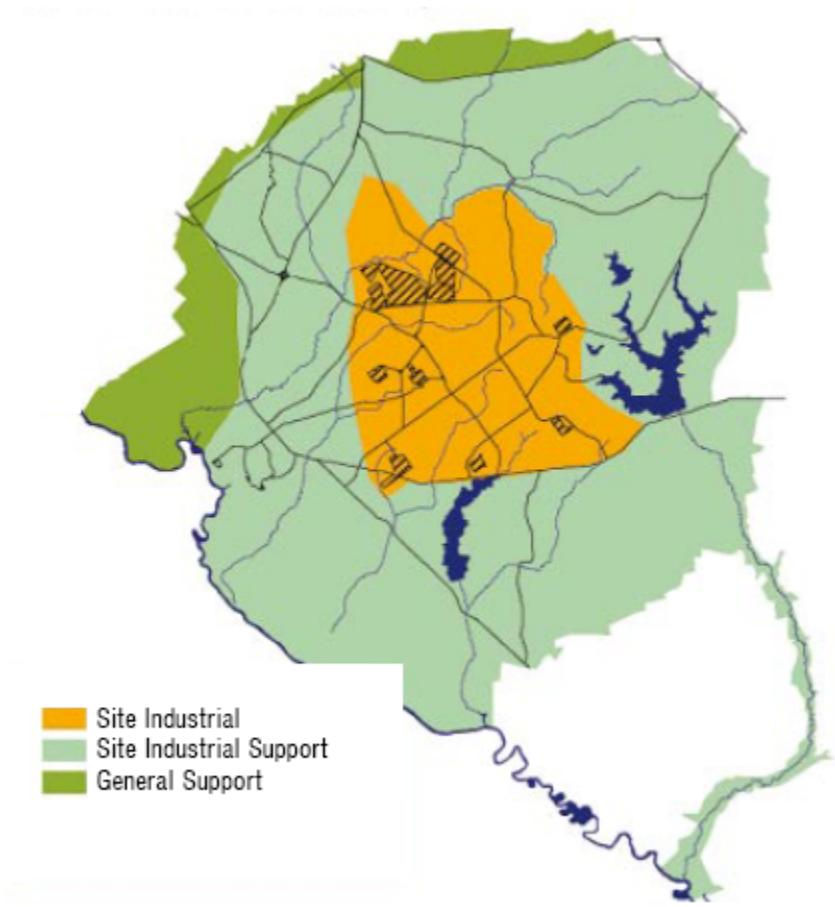

Figure 4-1. SRS Future Land Use Zoning

4.2.4 General Separations Area Hydrology

The ELLWF, SDF, FTF and HTF, E-Area TRU Pad 1, and FAMS (i.e., the facilities for which a CA is required) are all located within the GSA. Surface water and groundwater hydrology of the GSA are discussed in Sections 3.2.9 and 3.2.10, so only the points relevant to development of the POAs are summarized here.

The GSA is bordered by Upper Three Runs Creek to the north, McQueen Branch to the northeast, and Fourmile Branch (alternatively referred to as Four Mile Creek in some figures and referenced material) to the south. McQueen Branch is a tributary of Upper Three Runs Creek. Upper Three Runs Creek and Fourmile Branch empty into the Savannah River, which forms the southwest boundary of SRS. The water table aquifer in the GSA is divided into an upper aquifer zone and a lower aquifer zone separated by the tan clay confining zone. A groundwater divide exists within both water table zones with groundwater flow emptying into both Upper Three Runs Creek/McQueen Branch and Fourmile Branch.

The underlying Gordon aquifer is separated from the water table aquifer by the Gordon confining unit, commonly called the green clay. A downward gradient exists across the Gordon confining unit; however the bulk of groundwater flow within the water table aquifer empties into Upper Three Runs Creek / McQueen Branch and Fourmile Branch rather than migrating across the Gordon confining unit into the Gordon aquifer. The Gordon aquifer empties into Upper Three Runs Creek over the entire GSA, due to the incisement of the Gordon aquifer by Upper Three Runs Creek. No groundwater flow occurs from the Gordon aquifer to underlying aquifers due to the upward gradient across the underlying Meyers Branch confining system (Denham 1999 and Smits et al. 1997). Therefore, the water transport pathway (i.e., vadose zone, saturated zone, and streams) from radionuclide source locations within the GSA to a publicly accessible point is through the vadose zone, water table and Gordon aquifers to either Upper Three Runs Creek or Fourmile Branch and subsequently to the Savannah River. The public can first come into contact with radionuclide contaminated media originating from the GSA at the mouths of Upper Three Runs Creek or Fourmile Branch at the Savannah River.

4.2.5 Savannah River Site Composite Analysis Points of Assessment

Due to the combination of the SRS land use plans (Section 4.2.3) and the GSA hydrology (Section 4.2.4), the first publicly accessible location where radionuclide contaminated media originating from the GSA (through the water transport pathway) can be contacted is at the mouth of Upper Three Runs Creek and Fourmile Branch at the Savannah River. Other SRS streams, which empty directly into the Savannah River, include Steel Creek and Lower Three Runs Creek. While Pen Branch reaches the Savannah River swamp (Figure 4-2), it does not discharge directly into the Savannah River, rather all of its flow to the swamp exits the swamp to the river through Steel Creek. Radionuclides originating from the GSA will mix with radionuclides entering the Savannah River from Steel Creek and Lower Three Runs Creek within the Savannah River just downstream from these creeks.

Wilhite and Phifer 2008, as discussed in Section 7.1.3, defined two CA exposure scenarios which are bounding exposure scenarios, a recreational scenario and a residential scenario. The recreational scenario involves activities such as boating, swimming, fishing, shoreline exposure, and inadvertent ingestion of surface water. It is assumed that recreational boaters will have access to the mouths of on-site streams for the recreational scenario. The residential scenario involves use of contaminated surface water for drinking and other household uses, and for irrigation. As outlined in Section 4.2.3 residents within the SRS boundary are prohibited by SRS land use plans, therefore residents will not have access to on-site streams or groundwater. Therefore the residential scenario involves the use of water from the Savannah River and not from the on-site streams. The recreational scenario will be evaluated within the mouth of the streams prior to mixing with the Savannah River, whereas the residential scenario will be evaluated after mixing with the Savannah River.

Based upon this, the mouth of Upper Three Runs Creek, Fourmile Branch, Steel Creek, and Lower Three Runs Creek at the Savannah River will be taken as POAs for the recreational scenario. The Savannah River at Upper Three Runs Creek, Fourmile Branch, Steel Creek, and Lower Three Runs Creek will be taken as POAs for the residential scenario. Finally the Savannah River at the U.S. Highway 301 Bridge will also be taken as a POA, since it is the location where all surface water radionuclide migration from SRS intermingles and it is currently a monitoring point for the annual environmental reports (WSRC 2007b). Table 4-2 and Figure 4-2 show the locations of the POAs. Additionally alternative POAs within Upper Three Runs Creek and Fourmile Branch at the boundary of the Site Industrial zone will be considered as part of the CA sensitivity and uncertainty analysis (Phifer and Wilhite 2008; Phifer, Smith, et al. 2008; Wilhite and Phifer 2008; Smith et al. 2009).

Table 4-2. Locations of SRS CA Points of Assessment

Description	Savannah River Mile [1]
Upper Three Runs Creek at the Savannah River [2]	157.2
Fourmile Branch at the Savannah River [2]	150.6
Steel Creek at the Savannah River [2, 3]	141.6
Lower Three Runs Creek at the Savannah River [2]	129.1
Savannah River at the U.S. Highway 301 Bridge [4]	118.8

[1] Distance in miles from the Savannah River mouth.

[2] The recreational scenario is evaluated within the mouth of the streams prior to mixing with the Savannah River, whereas the residential scenario is evaluated after mixing with the Savannah River and includes the radionuclide contribution from all upgradient SRS sources. It is assumed that recreational boaters will have access to the mouths of on-site streams. However, residents within the SRS boundary are prohibited from having access to on-site streams or groundwater by SRS land use plans (Phifer and Wilhite 2008; Phifer, Smith, et al. 2008).

[3] Pen Branch enters Steel Creek prior to discharging to the Savannah River.

[4] The recreational and residential scenarios are evaluated in the Savannah River at the U.S. Highway 301 Bridge by combining the radionuclide contribution from all upgradient SRS sources.

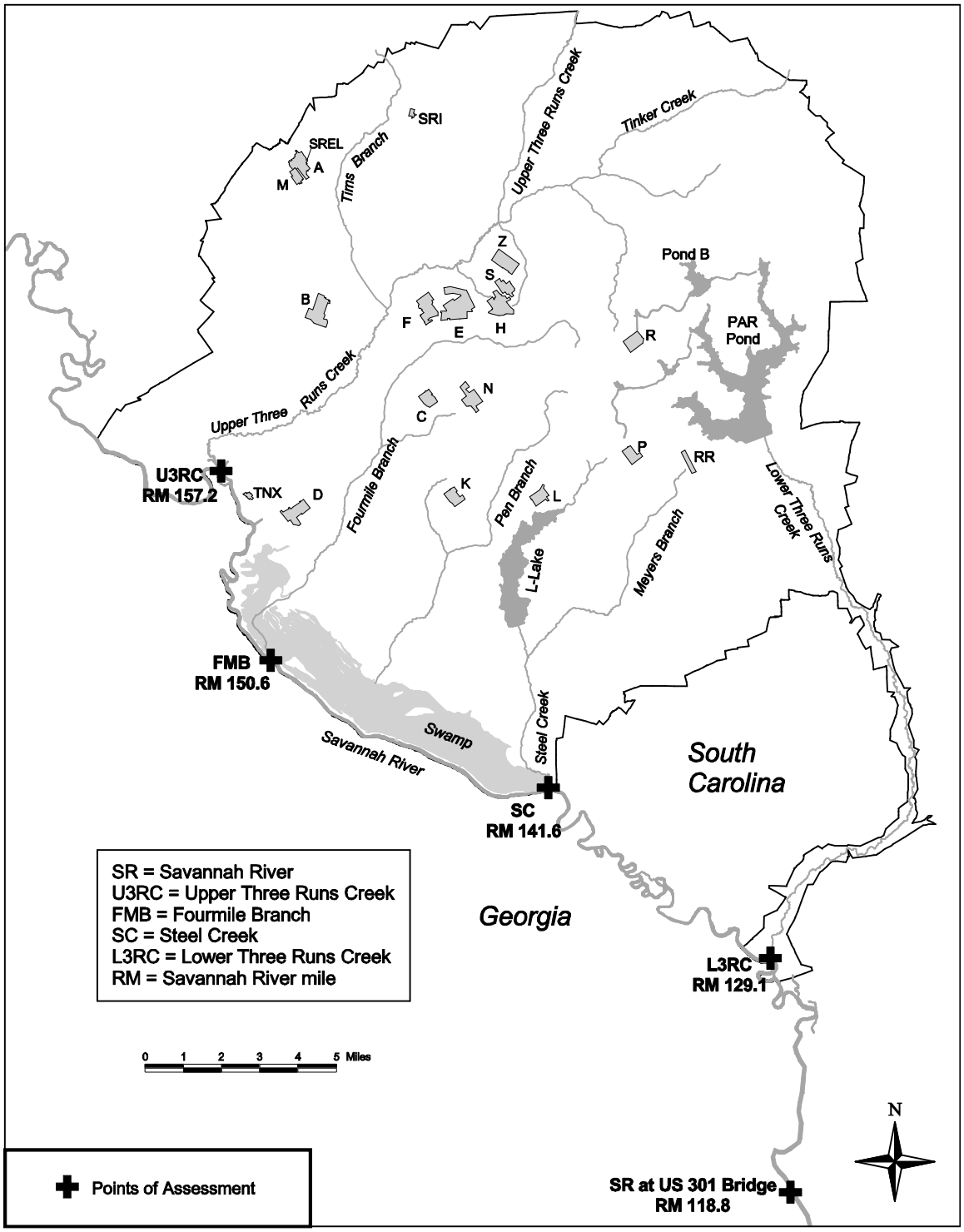

Figure 4-2. Locations of SRS CA Points of Assessment

4.3 ASSESSMENT PERIOD

The DOE Radioactive Waste Management Manual (DOE 1999d) and DOE Implementation Guide (DOE 1999b) require that the maximum dose and the time of maximum dose to a hypothetical member of the public be estimated over a 1,000-year AP following disposal facility closure. The LFRG manual (DOE 2008) further clarifies the AP as a 1,000-year period after all disposal facilities have been closed, all facility D&D completed, and operations at the DOE site have terminated. The DOE format and content guide for Composite Analyses (DOE 1999c) further states that as part of the sensitivity and uncertainty analysis the maximum dose and the time of maximum dose beyond the 1,000-year AP must also be estimated.

Additionally the Performance Assessments (PAs) for the SDF, FTF, and HTF are being evaluated over 10,000 years in conformance to Nuclear Regulatory Commission (NRC) requests (Phifer and Cook 2007). This CA was also being performed for these facilities. Therefore as part of the sensitivity and uncertainty analysis the maximum dose and the time of maximum dose over 10,000 years were also determined.

Appendix B of Phifer and Wilhite (2008) provides a listing of pertinent documents, including those referenced above, that provide the basis for AP selection, along with extracts from the documents concerning the AP.

As outlined above the AP is to start after all disposal facilities have been closed, all facility D&D completed, and operations at the DOE site have terminated. The following are statements pertinent to the start of the AP:

- The SRS Comprehensive Plan/Ten Year Site Plan (SRNS 2009) states that the site anticipates that environmental remediation activities will be completed by 2038, after which long-term stewardship activities will be fully implemented.
- The current SRS waste site remediation plan (WSRC 2003a) projects that by September 2038 all inactive waste sites would have been remediated under a baseline plan. Under a new accelerated strategy the plan projects the remediation of all waste sites by September 2023.
- The current SRS deactivation and decommissioning plan (WSRC 2003b) projects that by September 2020 all FTF and HTF tanks would have been grouted and closed. It further projects that by September 2025 all inactive waste sites would have been remediated and all facilities would have been decommissioned.
- The SRS End State Vision (DOE 2005) states, “The SRS Cleanup Project mission and goal is to complete the cleanup by 2025 and transition SRS to a site focused on national security. SRS will accommodate the ongoing national Nuclear Security Administration (NNSA) missions before and beyond 2025. SRS is not a DOE closure site.”

- The current SRS liquid waste disposition plan (WSRC 2007c) projects that by September 2030 all FTF and HTF tanks will have been emptied, all sludge will have been processed through the Defense Waste Processing Facility (DWPF) and shipped off site, and all salt solution will have been processed through the Salt Waste Processing Facility (SWPF) and disposed on-site as saltstone within the SDF. The plan additionally projects that by September 2032 all FTF and HTF tanks will have been grouted, DWPF and SWPF will have been cleaned, and the Glass Waste Storage Buildings (GWSB) and the canister shipping facility will have been decommissioned and closed.
- The current SRS Federal Facility Agreement (WSRC 2007d, Appendix E.3) projects that the last Remedial Action (RA) start will occur in September 2025 for the Savannah River and Floodplain Swamp Integrator Operable Unit.

The earliest completion dates outlined in the above plans for closure of the FTF and HTF tanks, remediation of all inactive waste sites, and decommissioning of all facilities are 2020, 2023, and 2025, respectively. Therefore the start of the AP, defined as the date when all disposal facilities have been closed, all facility D&D completed, and operations at the DOE site have terminated, was taken to be the year 2025. The assumed start of the AP was not based upon continuing and future missions, which may extend beyond 2025, such as the potentially continuing NNSA missions. However the radionuclide inventories associated with current NNSA missions and facilities, even if their operation is projected to extend beyond 2025, were included in the CA as appropriate.

Based upon the criteria outlined above, the AP was determined to be as follows:

- The AP, as required by DOE 435.1-1 (DOE 1999d), is a 1,000-year period beginning after year 2025 (i.e., earliest time that all SRS operations have been assumed to cease (i.e., after all disposal facilities and tanks have been closed, all RCRA/CERCLA remediation has been completed, all site D&D has been completed, and all DOE site operations, other than long-term stewardship, NNSA missions, and other future missions, have ceased)). The dose as a function of time and the maximum dose and the time of maximum dose to a hypothetical future member of the public was determined within the 1,000-year AP.
- As part of the sensitivity and uncertainty analysis the following were determined:
    - The maximum dose and the time of maximum dose over an alternative 10,000-year Assessment Period. This period is not mandated by DOE 435.1-1, rather these analyses provide information that is potentially of value for those facilities regulated by Section 3116 of the 2005 National Defense Authorization Act.
    - The maximum dose and the time of the maximum dose beyond the 1,000-year AP regardless of its duration (within reasonable limits of modeling capability), as recommended by the DOE PA/CA Format and Content Guide (DOE 1999c).

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5.0 ANALYSIS APPROACH, METHODOLOGY, AND OVERVIEW

In this section, an overview of the approach taken to carry out the required analyses for the Savannah River Site (SRS) Composite Analysis (CA) revision and the methods employed to implement this approach are described. Oversight, direction, and approval associated with production of this CA revision were provided through formation of the CA Steering and Core Teams, described in Section 5.1 below. In Section 5.2, the following series of CA planning documents, prepared to define the tasks required to produce the CA and formalize the methodology used to perform the tasks, are outlined:

- *Comprehensive SRS Composite Analysis Program Plan* (Phifer and Cook 2007)
- *Comprehensive SRS Composite Analysis Execution Plan* (Phifer and Saldivar 2008a)
- *Comprehensive SRS Composite Analysis Program Quality Assurance Plan* (Phifer and Saldivar 2008b)
- *Comprehensive SRS Composite Analysis Criteria and Comments Matrices* (Phifer et al. 2008a)

The various tasks required for production of the CA, performed by individual task teams, are outlined in Section 5.3. The tasks typically culminated in the issuance of a report, which was used wholly or in part as direct input to this CA document. The CA Steering and Core Teams provided oversight, direction, and approval of the work performed by the individual task teams.

The primary approach employed during development of this CA revision, regarding input data and information, was to use existing data and information, rather than develop new data and information. The only exception to this general rule was based upon the results of the original 1997 CA, which indicated that C14 associated with the Old Radioactive Waste Burial Ground (ORWBG) was a significant contributor to the 1997 CA dose. Based upon this 1997 CA observation, the development of site-specific C14 geochemical and C14 water-to-fish bioaccumulation factors was incorporated as part of this CA revision. However during development of this CA revision, areas of improvement in regards to the existing data and information were noted and are outlined within Section 11.0, Future Work.

5.1 COMPOSITE ANALYSIS STEERING AND CORE TEAMS

Due to the diverse technical areas encompassed by this CA revision and CA coverage including the entire site, development of the CA revision began with the formation of a CA Steering Team to steer the overall effort and a CA Core Team to oversee implementation of the tasks, in accordance with Task 4.1 of the *Comprehensive SRS Composite Analysis Program Plan* (Phifer and Cook 2007). The CA Steering Team consisted of representatives from the Department of Energy – Savannah River (DOE-SR), Savannah River National Laboratory (SRNL), and Regulatory Integration and Environmental Services (RI&ES). The CA Steering Team provided overall CA direction and decision-making on policy and strategic issues, provided CA related review and approval, provided interface with DOE-SR and the Department of Energy (DOE) Low-Level Waste Disposal Facility Federal Review Group (LFRG), and provided feedback, as necessary, to SRS organizations in order to apprise them of any potential issues arising during CA development, which could have impacted their area of responsibility.

The CA Core Team consisted of the DOE-SR, SRNL, and RI&ES CA Steering Team representatives, SRNL technical experts, and personnel representing SRS organizations with custodial authority over the bulk of the sites containing radionuclides (i.e., Soil and Groundwater Closure Projects [SGCP], Site Deactivation and Decommissioning [SDD], Waste Management Area Projects [WMAP], and LWO). The CA Core Team performed three primary functions. First they provided recommendations concerning overall CA policy and strategic issues. Second the team provided technical oversight, direction, and review of philosophy, concepts, assumptions, input parameters and data (especially facility-specific), task proposals, methods, screening, modeling, results, documents, etc. associated with individual tasks. Finally the CA Core Team included representatives from SGCP, SDD, WMAP, and LWO to serve as an information resource and provide coordination of reviews in their respective organizations. A listing of CA Core Team members is provided in Table 13-1. CA Core Team meetings were scheduled on a biweekly basis and held as necessary, typically once a month, to facilitate the CA Core Team performance of their functions. Meeting minutes were documented and maintained within the CA files.

5.2 COMPOSITE ANALYSIS PLANNING DOCUMENTS

The planning documents developed for the CA revision effort are described below. These documents define the tasks required to produce the CA and provide a means of formalizing the methodology used to perform the tasks.

5.2.1 Composite Analysis Program Plan (Phifer and Cook 2007)

The purpose of the CA Program Plan was to present the plan for revision of the SRS CA pursuant to the requirements of DOE Order 435.1 (DOE 1999a) and its supporting manuals and guides (DOE 1999b, 1999c, 1999d, 2001, 2006, and 2007). The CA Program Plan outlined the discrete tasks listed in Table 5-1 for performance in revising the CA. Many of the tasks outlined in Table 5-1 culminated in individual reports issued by SRNL. The reports were issued at the conclusion of each respective task and were used in whole or part as input to this revised CA document.

Table 5-1. CA Program Plan Tasks

Program Plan Section	Task
4.1	CA Steering and Core Teams
4.2	CA Requirements Matrix
4.2.1	Composite Analysis Document Outline
4.2.2	Quality Assurance & Quality Control
4.2.3	Carbon-14 Studies
4.2.4	SRS Site Description
4.2.5	ELLWF, SDF, FTF, HTF, & E-Area TRU Pad 1 Description[1] (including relationship to other SRS radionuclide source locations)
4.2.6	Related Documentation
4.2.7	Radionuclide Source Locations
4.2.8	Radionuclide Inventory
4.2.9	Points of Assessment
4.2.10	Assessment Period
4.2.11	Preliminary Transport Modeling Input Data
4.2.12	Transport Pathway and Exposure Scenario Screening
4.2.13	Radionuclide Screening
4.2.14	Transport Conceptual Model (initial)
4.2.15	Existing Transport Model Evaluation
4.2.16	Modeling Philosophy
4.2.17	Selection of Transport Modeling Codes
4.2.18	Radionuclide Source Release Rates
4.2.19	Conduct Initial Transport Modeling Simulations
4.2.20	Final Transport Modeling Input Data
4.2.21	Conduct Transport Modeling
4.2.22	Conduct Dose Calculations/Modeling
4.2.23	Present Input Data and Results using GIS[2] Tools
4.2.24	Sensitivity and Uncertainty Analyses
4.2.25	Performance Evaluation
4.2.26	Options Analysis
4.2.27	Compile Revised CA
4.2.28	CA Review and Approval

[1] ELLWF = E-Area Low Level Waste Facility; SDF = Saltstone Disposal Facility; FTF = F-Area Tank Farm; HTF = H-Area Tank Farm; TRU = Transuranic
[2] Geographic information system

5.2.2 Composite Analysis Execution Plan (Phifer and Saldivar 2008a)

The CA Execution Plan was prepared to outline the process through which the tasks listed in the CA Program Plan (see Section 5.2.1) were completed and the overall CA document was prepared for DOE LFRG review and approval. The execution plan prescribed the organization developed to produce the revised CA, which included the CA Steering Team (see Section 5.1), the CA Core Team (see Section 5.1), and the individual CA Task Teams (see Section 5.3). This plan also outlined the preparation, review, and approval process associated with the SRNL reports prepared in response to the program plan tasks outlined in Table 5-1. The review and approval of the CA reports for adequacy, completeness, and correctness was conducted in a graded approach, based primarily upon the technical complexity of the task as outlined in Section 6.1.1.2 (Document Control). Finally the execution plan presented a preliminary schedule developed for preparation of the revised CA.

5.2.3 Composite Analysis Quality Assurance Plan (Phifer and Saldivar 2008b)

The CA Quality Assurance Plan was prepared to define the quality assurance (QA) controls to be taken during production of the CA. Details of the QA plan as implemented for production of the CA are provided in Section 6.1.

5.2.4 Composite Analysis Criteria and Comments Matrices (Phifer et al. 2008a)

As a preliminary step to revision of the SRS CA, the criteria defined by the Department of Energy (DOE) as applicable to production of a CA and DOE comments received on the original 1997 CA were reviewed and matrices were developed to ensure that all relevant requirements were reviewed, evaluated for applicability, and considered in the revised CA, as appropriate. The following documents were evaluated for applicability:

- *Guidance for a Composite Analysis of the Impact of Interacting Source Terms on the Radiological Protection of the Public from Department of Energy Low-Level Waste Disposal Facilities* (DOE 1996)
- *Composite Analysis E-Area Vaults and Saltstone Disposal Facilities* (WSRC 1997)
- DOE M 435.1-1, *Radioactive Waste Management Manual*, Chapter IV, Low-Level Waste Requirements (DOE 1999b)
- *Format and Content Guide for U.S. Department of Energy Low-Level Waste Disposal Facility Performance Assessments and Composite Analyses* (DOE 1999d)
- *Low-Level Waste Disposal Facility Federal Review Group Manual* (DOE 2008).

During this evaluation, it was determined that the criteria listed in the final three documents convey the currently applicable and relevant criteria for CA preparation and are comprehensive (i.e., cover all the criteria that were listed in the other references reviewed). Additionally, it was determined that the CA criteria listed within the 1999 CA Format and Content Guide (DOE 1999d) had been superseded by the criteria listed within the 2008 LFRG Manual (DOE 2008). Therefore, only criteria from the following two sources, in order of priority, were included in the criteria matrix:

1. DOE M 435.1-1, *Radioactive Waste Management Manual*, Chapter IV, Low-Level Waste Requirements (IV.P. (3) Composite Analysis) (DOE 1999b)
2. *Low-Level Waste Disposal Facility Federal Review Group Manual*, Revision 3, June 2008, (Section 3.3 CA Review Criteria) (DOE 2008).

However, it was also determined that the CA Format and Content Guide (Part C: Composite Analysis Standard Format and Content) (DOE 1999d) contains clarifying statements that should also be considered during development of the CA.

Based upon this evaluation, tables outlining the criteria extracted from the Radioactive Waste Management Manual (DOE 1999b), LFRG Manual (DOE 2008), and the clarifying statements provided within the CA Format and Content Guide (DOE 1999d) were developed and documented within the CA Criteria and Comments Matrices report (Phifer et al. 2008a). The primary table developed from this evaluation is the CA Review Criteria Matrix, based upon DOE M 435.1 (DOE 1999b) and the 2008 LFRG Manual (DOE 2008). The CA Review Criteria Matrix is provided as Appendix B to this revised CA document. To facilitate the LFRG review, this Appendix B table provides a list of where the CA Core Team believes that the CA criteria listed in DOE M 435.1 and the 2008 LFRG Manual are met within this revised CA document. The other tables provided the CA criteria and clarifying statements from the CA Format and Content Guide (DOE 1999d) in a format cross-referenced to Table 5-1. CA Program Plan tasks to facilitate consideration of the criteria and clarifying statements by the individual Task Teams were considered while implementing the tasks.

Each of the following documents was evaluated for review comments received on the 1997 CA:

- *Final Report for the Savannah River Site Composite Analysis Review* (DOE 1998)
- *Addendum to the Composite Analysis for the E-Area Vaults and Saltstone Disposal Facilities,* Revision 0 (Cook et al. 1999)
- *Addendum to the Composite Analysis for the E-Area Vaults and Saltstone Disposal Facilities*, Revision 1 (Cook et al. 2002)
- *Resolution of Final Composite Analysis Review Team Comments* (Cook 2003)
- *Performance Assessment and Composite Analysis Maintenance Program* (WSRC 2007)

From the evaluation of the above documents, 127 individual review comments received on the 1997 CA were identified. These comments were then reviewed for applicability to the current CA revision and were categorized as not applicable, applicable and to be addressed during CA implementation, or to be addressed outside this CA revision. Based upon this categorization, a summary review comments matrix was prepared that provided the response in relation to the current CA revision for each of the 127 review comments, which were determined to be applicable to the current CA revision. The summary comments matrix was put into a format cross-referenced to the CA Program Plan tasks (see Table 5-1) to facilitate consideration of the comments by the individual Task Teams while implementing the tasks.

Subsequently, the following documents were evaluated for review comments.

- Low-Level Waste Disposal Facility Federal Review Group Annual Summary Review for the Savannah River Z Area (Saltstone); and
- Low-Level Waste Disposal Facility Federal Review Group Annual Summary Review for the Saltstone Disposal Facility, 9/17/2009.

None requiring action in the CA revision were found.

5.3 COMPOSITE ANALYSIS TASK IMPLEMENTATION

Individual Task Teams were formed to implement each task (Table 5-1) required to complete the CA and to report to the CA Core Team concerning the assigned task. Each Task Team conducted their work consistent with the CA Program Plan (Phifer and Cook 2007), CA Execution Plan (Phifer and Saldivar 2008a), CA QA Plan (Phifer and Saldivar 2008b), and the CA Criteria and Comments Matrices (Phifer et al. 2008a). Task Team leads were assigned overall responsibility for task completion and documentation. As described in Section 5.2.1, many of the Table 5-1 tasks culminated in individual reports issued by SRNL. The reports were issued at the conclusion of each respective task and were used in whole or part as input to this revised CA document. Information learned during the implementation of preceding tasks was considered in the planning and execution of subsequent tasks, with the CA Core Team providing integrating oversight and direction to the individual Task Teams.

The review and approval of the CA reports for adequacy, completeness, and correctness was conducted in a graded approach based primarily upon the technical complexity of the task (Phifer and Saldivar 2008b). The preparation, review, approval, and issuance of planning, background information, information compilation, and scoping reports (i.e., those that did not include data manipulation and/or compilation, testing, and/or modeling) involved the following:

- Task Team report preparation
- Task Team report presentation to CA Core Team
- CA Core Team report review and comment
- Task Team report revision and issuance

The preparation, review, approval, and issuance of technical reports, which included data manipulation and/or compilation, testing, and/or modeling, involved the following:

- Task Team task proposal preparation
- Task Team task proposal presentation to the CA Core Team
- CA Core Team task modification and approval to proceed as appropriate
- Task Team report preparation
- Technical Design Check of report and associated work per WSRC 2004b (see Section 6.4 for a description of the design check process)
- Task Team report presentation to CA Core Team
- CA Core Team report review and comment
- Task Team report revision and issuance

Table 5-2 provides a listing of each of the CA Program Plan tasks and the resulting documentation produced by the Task Teams to complete the tasks. Section 13.0 provides a list of CA preparers along with a short biography for each and a statement regarding the contribution of each.

Table 5-2. CA Program Plan Tasks and Resulting Documentation

Program Plan Section	Task	Resulting Documentation
4.1	CA Steering and Core Teams	CA Core Team meeting minutes
4.2	CA Requirements Matrix	*Comprehensive SRS Composite Analysis Criteria and Comments Matrices* (Phifer et al. 2008a)
4.2.1	Composite Analysis Document Outline	Draft document outline prepared and revised as needed to guide CA document development
4.2.2	Quality Assurance & Quality Control	*Comprehensive SRS Composite Analysis Program Quality Assurance Plan* (Phifer and Saldivar 2008b)
4.2.3	Carbon-14 Studies	• *Systems Model of Carbon Dynamics in Four Mile Branch on the Savannah River Site* (Hinton et al. 2009) • *Carbon-14 Geochemistry at Savannah River Site* (Roberts and Kaplan 2008)
4.2.4	SRS Site Description	*Background Information for the Savannah River Site's Composite Analysis* (Phifer et al. 2008b)
4.2.5	ELLWF, SDF, FTF, HTF, & E-Area TRU Pad 1 Description (including relationship to other SRS radionuclide source locations)	
4.2.6	Related Documentation	Prepared as a direct insert to the CA document
4.2.7	Radionuclide Source Locations	*Inventory of Residual Radioactive Material at the Projected Savannah River Site End State* (Hiergesell et al. 2008)
4.2.8	Radionuclide Inventory	
4.2.9	Points of Assessment	*Comprehensive SRS Composite Analysis Program Points of Assessment and Assessment Period* (Phifer and Wilhite 2008)
4.2.10	Assessment Period	

Table 5-2. CA Program Plan Tasks and Resulting Documentation--continued

Program Plan Section	Task	Resulting Documentation
4.2.12	Transport Pathway and Exposure Scenario Screening	*Comprehensive SRS Composite Analysis Program Transport and Exposure Pathway Screening* (Wilhite and Phifer 2008)
4.2.13	Radionuclide Screening	*Radionuclide Screening Model for the Savannah River Site's Composite Analysis* (Taylor et al. 2008)
4.2.14	Transport Conceptual Model (initial)	*Conceptual Model for the Savannah River Site's Composite Analysis* (Phifer et al. 2008c)
4.2.15	Existing Transport Model Evaluation	
4.2.16	Modeling Philosophy	
4.2.17	Selection of Transport Modeling Codes	
4.2.11	Preliminary Transport Modeling Input Data	• *Savannah River Site Composite Analysis: Aquifer Flow Path Parameters* (Hamm et al. 2009) • *Savannah River Site Surface Water Flow Rates for Composite Analysis Dose Calculations* (Jones 2009) • *Distribution Coefficients (K_ds), K_d Distributions, and Cellulose Degradation Product Correction Factors for the Composite Analysis* (McDowell-Boyer and Kaplan 2009) • *Savannah River Site Vadose Zone Data for Composite Analysis Calculations* (Noffsinger et al. 2009) • *Material Property, Infiltration, and Saturation Estimates and Distributions* (Phifer and Dixon 2009) • *Savannah River Site Composite Analysis: Dose Module* (Phifer et al. 2009) • *Distribution of the 1,000-Year Mean of Background Infiltration in SRS Soils* (Shine 2008) • *Distributions of the Means of Stream Flows in the Savannah River Site Vicinity over One-Year and 1,000-Year* (Shine 2009) • *Radionuclide Screening Model for the Savannah River Site's Composite Analysis* (Taylor et al. 2008)
4.2.20	Final Transport Modeling Input Data	
4.2.19	Conduct Initial Transport Modeling Simulations	• *Savannah River Site Composite Analysis: Base Case Deterministic Calculations* (Smith et al. 2009a) • *Savannah River Site Composite Analysis: Dose Module* (Phifer et al. 2009)
4.2.18	Radionuclide Source Release Rates	
4.2.21	Conduct Transport Modeling	
4.2.22	Conduct Dose Calculations/Modeling	
4.2.23	Present Input Data and Results using GIS Tools	*Savannah River Site Composite Analysis: Aquifer Flow Path Parameters* (Hamm et al. 2009)
4.2.24	Sensitivity and Uncertainty Analyses	*Savannah River Site Composite Analysis: Sensitivity and Uncertainty Calculations* (Smith et al. 2009b)
4.2.25	Performance Evaluation	Prepared as a direct insert to the CA document
4.2.26	Options Analysis	Prepared as a direct insert to the CA document
4.2.27	Compile Revised CA	CA Document itself
4.2.28	CA Review and Approval	NA

N/A = not applicable

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6.0 QUALITY ASSURANCE AND QUALITY CONTROL

The Comprehensive SRS Composite Analysis Program Quality Assurance Plan (Phifer and Saldivar 2008a) outlines the quality assurance (QA) controls that were taken during production of the Savannah River Site (SRS) Composite Analysis (CA). This plan was approved by a representative of Savannah River National Laboratory (SRNL) Quality Engineering, who performed the Cognizant Quality Function (CQF) for the CA. The discussions in Section 6.1 are taken largely from that document. Portions of the QA associated with the CA-associated C14 geochemical sorption studies (Roberts and Kaplan 2008) and Dose Module (Phifer et al. 2009) development were addressed separately from the QA controls outlined in Phifer and Saldivar 2008a. The pertinent additional QA information for these activities is provided in Sections 6.2 and 6.3, respectively. Finally, in Section 6.4, an overview of the technical design check process and a listing of the documents produced as a result of this process are provided.

6.1 COMPREHENSIVE COMPOSITE ANALYSIS QUALITY ASSURANCE PLAN

Production of the CA primarily involved the compilation of existing information and data (with the exception of the C14 studies) for use in modeling efforts to determine the projected dose to a hypothetical future member of the public. The CA effort is classified as a Research and Development (R&D) task activity per Quality Assurance Manual (1Q) Procedure 2-3 (WSRC 2003), since the CA is required as part of compliance with U.S. Department of Energy (DOE) Order 435.1 (DOE 1999) and will be transmitted to the DOE Low-Level Waste Disposal Facility Federal Review Group (LFRG) for review and approval.

6.1.1 Quality Assurance Controls

The applicability of various QA requirements outlined within Quality Assurance Procedure (QAP) 2-3, Control of Research and Development Activities (WSRC 2003), to the CA tasks, other than the C14 geochemistry studies, are outlined within Table 6-1.

Table 6-1. CA QA Matrix

WSRC 1Q Section	SRNL[1] Implementing Procedures	Applicability to CA (Yes/No)
Organization	1Q, Procedure 1-1, Organization	**Yes** [2]
	1Q, Procedure 1-2, Stop Work	**Yes** [3]
	L1, Procedure 1.02, SRNL Organization	**Yes** [2]
	L1, Procedure 1.32, Read and Sign	**Yes** [2]
	L1, Procedure 7.16, Laboratory Notebooks and Logbooks	No
Quality Assurance Program	1Q, Procedure 2-1, Quality Assurance Program	**Yes** [2]
	L1, Procedure 8.02, SRNL QA Program Implementation and Clarification	**Yes**
	1Q, Procedure 2-2, Personnel Training & Qualification	**Yes** [2]
	1Q, Procedure 2-3, Control of Research & Development Activities	**Yes**
	L1, Procedure 7.10, Control of Technical Work	**Yes**
	1Q, Procedure 2-4, Auditor/Lead Auditor Qualification & Certification	No
	1Q, Procedure 2-5, Training, Qualification and Certification of Inspection Personnel	No
	1Q, Procedure 2-7, QA Program Requirements for Analytical Measurement Systems	No
Design Control	1Q, Procedure 3-1, Design Control	No
	L1, Procedure 7.10, Control of Technical Work	No

Table 6-1. CA QA Matrix--continued

WSRC 1Q Section	SRNL[1] Implementing Procedures	Applicability to CA (Yes/No)
Procurement Document Control	1Q, Procedure 4-1, Procurement Document Control	No
	7B, Procurement Management Manual	No
	3E, Procurement Specification Procedure Manual	No
	E7, Procedure 3.10, Determination of Quality Requirements for Procured Items	No
Instructions, Procedures, and Drawings	1Q, Procedure 5-1, Instructions, Procedures, and Drawings	**Yes** [3]
	L1, Procedure 1.01, SRNL Procedure Administration (Unreviewed Safety Question Required)	**Yes** [3]
	E7, Procedure 2.30, Drawings	No
Document Control	1Q, Procedure 6-1, Document Control	**Yes**
Control of Purchased Items and Services	1Q, Procedure 7-2, Control of Purchased Items and Services	No
	1Q, Procedure 7-3, Commercial Grade Item Dedication	No
	7B, Procurement Management Manual	No
	3E, Procurement Specification Procedure Manual	No
Identification and Control of Items	1Q, Procedure 8-1, Identification and Control of Items	No
Control of Processes	1Q, Procedure 9-1, Control of Processes	No
	1Q, Procedure 9-2, Control of Nondestructive Examination	No
	1Q, Procedure 9-3, Control of Welding and Other Joining Processes	No
	1Q, Procedure 9-4, Work Processes	No
Inspection	1Q, Procedure 10-1, Inspection	No
	L1, Procedure 8.10, Inspection	No
Test Control	1Q, Procedure 11-1, Test Control	No
Control of Measuring and Test Equipment	1Q, Procedure 12-1, Control of Measuring and Test Equipment	No
	1Q, Procedure 12-2, Control of Installed Process Instrumentation	No
	1Q, Procedure 12-3, Control and Calibration of Radiation Monitoring Equipment.	No
Packaging, Handling, Shipping, and Storage	1Q, Procedure 13-1, Packaging, Handling, Shipping, and Storage	No
Inspection, Test, and Operating Status	1Q, Procedure 14-1, Inspection, Test, and Operating Status	No
Control of Nonconforming Items and Activities	1Q, Procedure 15-1, Control of Nonconforming Items	**Yes** [3]
	1B, MRP 4.23, Site Tracking Analysis and Reporting	**Yes** [3]
Corrective Action System	1-01, MP 5.35, Corrective Action Program	**Yes** [3]
	1Q, Procedure 16-3, Corrective Action Program	**Yes** [3]
Quality Assurance Records	1Q, Procedure 17-1, Quality Assurance Records Management	**Yes**
	L1, Procedure 7.16, Laboratory Notebooks and Logbooks	No
	1Q, Procedure 18-2, Quality Assurance Surveillance	No
	1Q, Procedure 18-3, Quality Assurance External Audits	No
	1Q, Procedure 18-4, Management Assessments	**Yes** [2]
	1Q, Procedure 18-6, Quality Assurance Internal Audits	No
	1Q, Procedure 18-7, Quality Assurance Supplier Surveillance	No
	12Q, Procedure SA-1, Self-Assessment by Individual Assessment Units	**Yes** [2]
Quality Improvement	1Q, Procedure 19-2, Quality Improvement	**Yes** [3]
Software Quality Assurance	1Q, Procedure 20-1, Software Quality Assurance	**Yes**
Environmental Quality Assurance	1Q, Procedure 21-1, Quality Assurance Requirements for the Collection and Evaluation of Environmental Data	No

[1] Savannah River National Laboratory (SRNL) superseded Savannah River Technology Center (SRTC) in 2004.
[2] These applicable QA requirements are part of the overall SRS QA program and are implemented external to production of the CA. Therefore, information regarding these items is not provided herein.
[3] These applicable QA requirements are existing SRS QA control requirements that were not required to be implemented for production of the CA. Therefore information regarding these items is not provided herein.

The following QA requirements are directly applicable to production of the CA and are therefore discussed in detail herein:

- Manual 1Q, Procedure 1-1, Organization (WSRC 2005a)
- Manual 1Q, Procedure 2-3, Control of Research & Development Activities (WSRC 2003), SRNL Manual L1, Procedure 7.10, Control of Technical Work (SRNL 2002), and SRNL Manual L1, Procedure 8.02, SRTC QA Program Implementation and Clarification (SRNL 2005) (see Table 1 for the applicable portions of these requirements)
- Manual 1Q, Procedure 6-1, Document Control (WSRC 2004a) and Manual 1Q, Procedure 17-1, Quality Assurance Records Management (WSRC 2005b)
- Manual 1Q, Procedure 20-1, Software Quality Assurance (WSRC 2007)

The Radionuclide Source Locations and Radionuclide Inventory tasks (Sections 4.2.7 and 4.2.8, respectively, of the CA Program Plan, Phifer and Cook 2007) form the heart of the CA. Therefore the QA for these tasks is also discussed in detail herein.

The following sections address the organization, document control, and software quality assurance associated with CA production. Additionally the specific QA associated with Radionuclide Source Locations and Inventory is also discussed in a separate section below.

6.1.1.1 Organization (WSRC 2005a)

The organization established for production of the CA is outlined in detail within the CA Execution Plan (Phifer and Saldivar 2008b). The organization consists of the following:

- <u>CA Steering Team</u> The CA Steering Team consisted of representatives from the Department of Energy – Savannah River (DOE-SR), SRNL, and Site Regulatory Integration and Planning (SRIP; essentially a representative organization from Liquid Waste Operations (LWO)). The CA Steering Team performed the following functions:
  - Provided overall CA direction and decision-making on policy and strategic issues,
  - Provided CA related review and approval,
  - Provided interface with DOE-SR and the DOE LFRG, and
  - Provided feedback, as necessary, to SRS organizations in order to apprise them of any potential issues arising during CA development which could have impacted their area of responsibility.

- CA Core Team The CA Core Team consisted of the DOE-SR, SRNL, and SRIP CA Steering Team representatives, SRNL technical experts, and personnel representing SRS organizations with custodial authority over the bulk of the sites containing radionuclides (i.e., Soil and Groundwater Closure Projects [SGCP], Site Deactivation and Decommissioning [SDD], Waste Management Area Projects [WMAP], and LWO). The CA Core Team performed the following functions:
  - Recommendations concerning overall CA policy and strategic issues,
  - Technical oversight, direction, and review of philosophy, concepts, assumptions, input parameters and data (especially facility-specific), task proposals, methods, screening, modeling, results, document, etc. associated with individual tasks, and
  - SGCP, SDD, WMAP, and LWO representation to serve as an information resource and provide coordination of reviews in their respective organizations.
- Task Teams Individual Task Teams were formed to implement each task required to complete the CA as outlined within Section 4.0 of the CA Program Plan (Phifer and Cook 2007) and to report to the CA Core Team concerning the assigned task.

Each individual was responsible for the quality of his or her work. The Task Team leads were responsible for the overall quality of their assigned task(s). The CA Core Team and the CA Steering Team were responsible for the overall quality of the entire CA effort. The CQF was responsible for supporting the CA Core Team and the CA Steering Team, as necessary, to help ensure that appropriate measures were taken to ensure the overall quality of the entire CA effort.

6.1.1.2 Document Control (WSRC 2004a; WSRC 2005b)

Many of the CA tasks outlined within Section 4.0 of the CA Program Plan (Phifer and Cook 2007) culminated in the issuance of reports. The review and approval of the CA reports for adequacy, completeness, and correctness was conducted in a graded approach based primarily upon the technical complexity of the task (Phifer and Saldivar 2008b). The preparation, review, approval, and issuance of planning, background information, information compilation, and scoping reports (i.e., those that did not include data manipulation and/or compilation, testing, and/or modeling) involved the following:

- Task Team report preparation
- Task Team report presentation to CA Core Team
- CA Core Team report review and comment
- Task Team report revision and issuance

Table 6-2 provides a listing of the CA planning, background information, information compilation, and scoping reports that were approved as outlined above.

The preparation, review, approval, and issuance of technical reports, which included data manipulation and/or compilation, testing, and/or modeling, involved the following:

- Task Team task proposal preparation
- Task Team task proposal presentation to the CA Core Team
- CA Core Team task modification and approval to proceed as appropriate
- Task Team report preparation
- Technical Design Check of report and associated work per WSRC 2004b (see Section 6.4 for a description of the design check process)
- Task Team report presentation to CA Core Team
- CA Core Team report review and comment
- Task Team report revision and issuance

Table 6-3 provides a listing of the CA technical reports, which included data manipulation and/or compilation, testing, and/or modeling, that were approved as outlined above.

One of the primary differences in the preparation, review, approval, and issuance of technical reports, which included data manipulation and/or compilation, testing, and/or modeling, over that of reports for planning, background information, information compilation, and scoping is that technical reports underwent a document technical design check per WSRC 2004b (see 6.4). For technical reports consisting primarily of data manipulation and/or compilation for use as model inputs, the primary purpose of the document technical design check was to ensure that the information and data were appropriately referenced to justify its use, that all data manipulation was correct and appropriate, and that all such information and data were properly transcribed. For technical reports consisting primarily of testing, the primary purpose of the document technical design check was to ensure that the testing was conducted appropriately and that data was properly transcribed. For technical reports consisting primarily of modeling, the primary purpose of the document technical design check was to ensure that the correct information and data were used and was transcribed correctly, to ensure that the model was constructed properly, and to ensure that modeling runs are traceable and reproducible.

Additionally all reports produced for the CA were assigned a unique document number through Records Management, underwent review and approval for external release, were entered into the site's Records Management and Document Control systems, and were marked for permanent record retention.

Table 6-2. CA Planning, Background Information, Information Compilation, and Scoping Reports

CA Report Title	CA Report Citation
Comprehensive SRS Composite Analysis Program Plan	Phifer and Cook 2007
Comprehensive SRS Composite Analysis Execution Plan	Phifer and Saldivar 2008b
Comprehensive SRS Composite Analysis Program Quality Assurance Plan	Phifer and Saldivar 2008a
Comprehensive SRS Composite Analysis Criteria and Comments Matrices	Phifer et al. 2008a
Background Information for the Savannah River Site's Composite Analysis	Phifer et al. 2008b
Comprehensive SRS Composite Analysis Program Points of Assessment and Assessment Period	Phifer and Wilhite 2008
Comprehensive SRS Composite Analysis Program Transport and Exposure Pathway Screening	Wilhite and Phifer 2008
Conceptual Model for the Savannah River Site's Composite Analysis	Phifer et al. 2008c
Data Management Plan for Archiving Composite Analysis Data and Models	Zamecnik 2010

Table 6-3. CA Technical Reports

CA Report Title	CA Report Citation
Radionuclide Screening Model for the Savannah River Site's Composite Analysis	Taylor et al. 2008
Inventory of Residual Radioactive Material at the Projected Savannah River Site End State	Hiergesell et al. 2008
Radiological Inventory of Site Deactivation & Decommissioning (D&D) Facilities at Savannah River Site	WSRC 2008
Carbon-14 Geochemistry at Savannah River Site	Roberts and Kaplan 2008
Distribution of the 1,000-Year Mean of Background Infiltration in SRS Soils	Shine 2008
Savannah River Site Surface Water Flow Rates for Composite Analysis Dose Calculations	Jones 2009a
Distribution Coefficients (K_ds), K_d Distributions, and Cellulose Degradation Product Correction Factors for the Composite Analysis	McDowell-Boyer and Kaplan 2009 a & b
Savannah River Site Vadose Zone Data for Composite Analysis Calculations	Noffsinger et al. 2009
Systems Model of Carbon Dynamics in Four Mile Branch on the Savannah River Site	Hinton et al. 2009a & b
Distributions of the Means of Stream Flows in the Savannah River Site Vicinity over One-Year and 1,000-Year	Shine 2009
Material Property, Infiltration, and Saturation Estimates and Distributions	Phifer and Dixon 2009
Savannah River Site Composite Analysis: Aquifer Flow Path Parameters	Hamm et al. 2009
Savannah River Site Composite Analysis: Dose Module	Phifer et al. 2009
Savannah River Site Composite Analysis: Base Case Deterministic Calculations	Smith et al. 2009a
Savannah River Site Composite Analysis: Sensitivity and Uncertainty Calculations	Smith et al. 2009b

6.1.1.3 Software Quality Assurance (WSRC 2007)

The GoldSim™ Version 9.60 SP4 (Service Pack 4) code (GTG 2007) was used to produce a model to determine the projected dose to a hypothetical future member of the public for this CA effort. The GoldSim™ code conforms to the requirements of WSRC 1Q, Quality Assurance Manual, Procedure 20-1, Software Quality Assurance (QAP 20-1, WSRC 2007). For use in the production of the CA, it is classified as Level "C" software or higher per QAP 20-1 Attachment 1, which classifies "software applications used to comply with regulatory laws, environmental permits or regulations and/or commitments to compliance" as Level "C" software. This is appropriate since the CA is required as part of compliance with DOE Order 435.1 (DOE 1999) and will be transmitted to the DOE LFRG for review and approval. One of the primary requirements for Level "C" software and above is an associated Software Quality Assurance Plan (SQAP) that addresses the life cycle requirements of the model. The GoldSim™ code (GTG 2007) utilized for production of the CA already has an associated SQAP, which was produced in association with Performance Assessment (PA) applications at the SRS. This plan is applicable to the CA applications of GoldSim™, and is documented by Swingle (2006). Documentation of the acceptance testing required by the CA SQAP for GoldSim™ Version 9.60, Service Pack 4, used for production of the CA, is provided by Swingle (2010).

No other software requiring a SQAP was utilized directly for production of this CA. However information developed from previous modeling efforts was utilized and appropriately referenced as outline in Section 8.1.2.2 and Section 9.2.1.

6.1.1.4 Radionuclide Source Locations and Inventory QA

The QA associated with the Radionuclide Source Locations and Radionuclide Inventory tasks (Tasks 4.2.7 and 4.2.8, respectively, of the CA Program Plan, Phifer and Cook 2007) was primarily divided into the following categories:

- References: Documentation that is adequate to justify the source locations and radionuclide inventories was appropriately referenced and cross-checked. Radionuclide inventories for source locations that were extrapolated from that of other source locations or other information such as Safety Analysis Reports (SARs) were appropriately referenced and the extrapolation was demonstrated to tend "conservative" (i.e., to overstate the inventory).

- Inventory Data Hierarchy: The following hierarchy was adhered to in developing the inventory estimates:
  - Existing documented inventories were utilized where available.
  - Inventories were developed from existing sampling and analysis data, where available.
  - Inventories were developed from the inventories of other similar facilities/waste sites.
  - Inventories[1] were developed from the operational inventories provided within SARs.

  In general this inventory estimate hierarchy also represents a qualitative measure of the data uncertainty, with the use of existing documented inventories considered most certain and inventories developed from SARs considered least certain.
- Facility Representative Concurrence: Inventories developed for SGCP, SDD, WMAP, and LWO source locations (i.e., bulk of the sites containing radionuclides) received concurrence from the associated facility custodial representatives on the CA Core Team.
- Document Technical Design Check: The source location and radionuclide inventory report (Hiergesell et al. 2008) underwent a design check per the SRNL Technical Report Design Check Guidelines (WSRC 2004b). The design check included verification of the following: 1) that information and data were appropriately referenced to justify its use; 2) that all such information and data were properly transcribed; 3) that assumptions and correctness of calculations associated with inventory extrapolations were appropriate.
- CA Core Team Review: The source location and radionuclide inventory report (Hiergesell et al. 2008) underwent a review by the CA Core Team.

6.1.2 Cognizant Quality Function

The CA Quality Assurance Plan (Phifer and Saldivar 2008a) was approved by a representative of SRNL Quality Engineering, who performed the CQF for the CA. The CQF was responsible for supporting the CA Core Team and the CA Steering Team, as necessary, to help ensure that appropriate measures were taken to ensure the overall quality of the entire CA effort. In order to keep the CQF informed of CA activities, copies of all draft CA reports were provided to the CQF during the CA Core Team report review process and all CA Core Team meeting minutes were also provided. No revisions to the CA Quality Assurance Plan (Phifer and Saldivar 2008a) were deemed necessary throughout the CA process.

[1] Information on the hazards of a facility during operation is provided within its SAR. Thus, SAR inventory information represents the inventory of radioactive material in the facility during operation. The CA is an analysis of the post-operational impacts and thus uses projected end state inventories. The SAR inventories were used by assuming that a certain fraction of the operational inventory would be retained in the closed facility as outlined in Section 8.1.3.2.

6.1.3 Records and CA Data Management

All reports produced for the CA (see Table 6-2 and Table 6-3) are considered records and were entered into the SRS Records Management and Document Control system as outlined in Section 6.1.1.2.

Configuration control and the archival of CA data and models are maintained through implementation of the CA Data Management Plan (Zamecnik 2010). All pertinent CA related documents, models, files, etc. are maintained within the access controlled SRS CA Archive so that the full documentation associated with the SRS CA is maintained.

6.2 CARBON-14 GEOCHEMISTRY STUDIES QUALITY ASSURANCE

A laboratory study was conducted at SRNL in support of the CA to measure sorption coefficients for C14 (Roberts and Kaplan 2008). The specific objectives of these studies were to quantify and understand how aqueous C14, as dissolved carbonate, sorbs to and desorbs from SRS sediments and cementitious materials. This study was conducted in conformance to the CA Quality Assurance Plan (Phifer and Saldivar 2008a). In particular a design check of the data and associated technical report was carried out in accordance with WSRC 2004b (see Sections 6.1.1.2 and 6.4). In addition to these QA measures, the geochemical studies were performed consistent with ASTM International (ASTM) D 4646, *Standard Test Method for 24-h Batch-Type Measurement of Contaminant Sorption by Soils and Sediments* (ASTM 2008). This ASTM document describes the laboratory apparatus and reagents to be used, the detailed procedure to be followed, and how results are interpreted.

6.3 DOSE MODULE QUALITY ASSURANCE

The dose module used to perform CA dose calculations was developed under a subcontract to Neptune and Company, Inc. using the GoldSim™ (GTG 2007) modeling platform. The dose module utilized for the CA is version 1.1. Neptune and Company, Inc. personnel also performed QA checking of the module itself and SRNL performed the design check of the input data utilized within the dose module in accordance with WSRC 2004b (see Sections 6.1.1.2 and 6.4). The Neptune and Company, Inc. QA Plan for the Dose Module is documented in *Software Quality Assurance Plan for Development of GoldSim™ Models* (Neptune 2009). This QA Plan addresses Software QA associated with use of the GoldSim™ model for implementing the dose module computations. This plan has elements that address configuration management, model data input selection and records, model testing, model backup, sensitivity analysis, error reporting and resolution, model validation and verification, benchmarking, and model review.

6.4 TECHNICAL DESIGN CHECK PROCESS

As outlined in Section 6.1.1.2, CA technical reports, which included data manipulation and/or compilation, testing, and/or modeling underwent a document technical design check per WSRC 2004b. This technical design check process was applied in the CA analysis to ensure that the information and data were appropriately referenced to justify its use, that all data manipulation was correct and appropriate, that all information and data were properly transcribed, that testing was conducted appropriately, that the correct information and data were used within the models, that assumptions were appropriate, and that modeling runs were traceable and reproducible. Design checks were carried out for modeling and data compilations that are described in the technical reports listed in Table 6-3. Table 6-4 provides a description of each of the six elements of the SRNL design check process (WSRC 2004b).

Table 6-4. Six Elements of the Design Check Process (from WSRC 2004b)

Element	Description
Analytical/experimental approach	Technical review should begin with review of analytical/ experimental approach as work plans are being developed by the PI. Performing this check in the planning stage should address major design issues and minimize the need for rework.
Mathematical check	Perform a 100% mathematical check of calculations unless a more limited check is agreed upon by PI or manager with the customer. A single 'calculation' can refer to a set of results from applying the same calculation repeatedly using different inputs as in a spreadsheet. In this instance, the calculation check should consist of: 1) checking to ensure that the correct equation is being employed 2) checking to ensure that the equation is properly loaded into the spreadsheet 3) performing a hand calculation to compare one or more sets of inputs to the equation with the spreadsheet answer(s).
Correct use of input	Ensure that information is properly incorporated into the report from references, experimental data contained in charts and graphs, inputs to models, etc. This requires checking the original source. PI should define critical inputs to the reviewer. A graded approach would be to check a subset of the inputs. If no errors are found, stop. If errors are found expand search to less critical or remaining inputs. Alternatively, return report to PI for further self-checking.
Justification for assumptions	All assumptions must be justified. In some cases a sensitivity analysis by the PI may be called for to assess impact where assumptions are highly uncertain and important to the result. Again a graded approach is needed to determine the level of effort required.
Reasonableness of output	Experience of the reviewer is important on this check. Questions that should be asked include; Do results make sense? Have results been applied appropriately and have correct conclusions been drawn from them? Review of the conclusions is especially important to ensure that the author adequately explains the limitations of the scope for the conclusions.
Cross-check for accuracy of transcription	Transcription errors are one of the most common types of errors found in technical reports. Reviewer should cross-check data tables and numbers disseminated throughout the text in the report. A graded approach would be to check a subset of the data. If no errors are found, stop. If errors are found expand your search to less critical or remaining parameters

The principles deemed important in the SRNL design check process are (WSRC 2004b):

- The manager and PI will interact with the customer on the requirements of the review starting with receipt of the customer's task request. Assess work in progress for safety and technical issues.
- The author has overall responsibility for report quality while the reviewer has primary responsibility for quality of the technical review.
- The extent of a technical review should be commensurate with the complexity of the analysis, importance to safety, relation to the facility baseline, regulatory implications, and similarity with previous analyses.
- The reviewer should be independent of the work being performed and capable of reconstructing the analysis/experiment on his/her own.
- The technical reviewer should maintain a questioning attitude throughout the review process.
- The reviewer should take advantage of experienced scientists, engineers, or managers in defining the approach to the technical review as needed.
- Two or more reviewers may be needed to satisfactorily cover the topic under review or to maintain the independence of the review process.
- The technical review should begin during the planning stage of research.
- The technical review process should be standardized to ensure consistency in the quality of reports being issued.
- Technical reviews should be documented.

When design checks were carried out on CA reports and activities, a report was issued, documenting the design check process and the results. A list of these design check reports is provided in Table 6-5 below.

Table 6-5. CA Design Check Reports

Report / Activity	Design Check Report Issued
Radionuclide Screening Model for the Savannah River Site's Composite Analysis (Taylor et al. 2008)	• *Design Check for Composite Analysis radionuclide screening parameter Excel Workbook named "GoldSim™ dose data" and the related appendix to the document: Radionuclide Screening Model for the Savannah River Site's Composite Analysis (SRNS-STI-2008-00117, rev. 0) Appendix A, Documentation of Parameters for All-Pathways Screening Analysis for the Savannah River Site Composite Analysis* (Phifer and McDowell-Boyer 2009a) • *Design Check of GoldSim™ CA Screening Model* (Taylor 2009a)
Inventory of Residual Radioactive Material at the Projected Savannah River Site End State (Hiergesell et al. 2008), and *Radiological Inventory of Site Deactivation & Decommissioning (D&D) Facilities at Savannah River Site* (WSRC 2008)	*Summary of Design Checks for Computations Performed to Establish Residual End-State Radioactive Material Inventories for Facilities at the SRS* (Hiergesell 2008)

Table 6-5. CA Design Check Reports--continued

Report / Activity	Design Check Report Issued
Carbon-14 Geochemistry at Savannah River Site (Roberts and Kaplan 2008)	*Design Check for Carbon-14 Geochemistry at Savannah River Site, SRNS-STI-2008-00445* (Roberts 2009)
Distribution of the 1,000-year Mean of Background Infiltration in SRS Soils (Shine 2008)	*Design Check for Distribution of the 1,000-Year Mean of Background Infiltration in SRS Soils*, *SRNL-L5100-2008-00012* (Phifer 2009a)
Savannah River Site Surface Water Flow Rates for Composite Analysis Dose Calculations (Jones 2009a)	*Design Check for Savannah River Surface Water Flow Rates for Composite Analysis Dose Calculations, SRNL-STI-2009-00077* (Jones 2009b)
Distribution Coefficients (K_ds), K_d Distributions, and Cellulose Degradation Product Correction Factors for the Composite Analysis, (McDowell-Boyer and Kaplan 2009a, 2009b, including Excel file)	*Design Check for Composite Analysis K^d/CDP Excel Worksheets and the documents: Distribution Coefficients (K_ds), K_d Distributions, and Cellulose Degradation Product Correction Factors for the Composite Analysis, SRNL-STI-2009-00150, Rev. 0 and Rev. 1* (Phifer and McDowell-Boyer 2009b)
Savannah River Site Vadose Zone Data for Composite Analysis Calculations (Noffsinger et al. 2009)	*Design Check for Savannah River Vadose Zone Data for Composite Analysis Calculations* (Phifer 2009b)
Systems Model of Carbon Dynamics in Four Mile Branch on the Savannah River Site (Hinton et al. 2009a, 2009b)	*Design Check for "Systems Model of Carbon Dynamics in Four Mile Branch on the Savannah River Site," SRNL-STI-2009-00178, REVISION 1* (Kaplan 2009)
*Distributions of the Means of Stream Flows in the Savannah River Site Vicinity over One-Year and 1,000-Year (*Shine 2009)	*Design Check for Distributions of the Means of Stream Flows in the Savannah River Site Vicinity over One-Year and 1,000-Year, SRNL-L5100-2009-00009* (Phifer 2009c)
Material Property, Infiltration, and Saturation Estimates and Distributions for the Composite Analysis (Phifer and Dixon 2009)	*Design Check for Material Property, Infiltration, and Saturation Estimates and Distributions for the Composite Analysis,* SRNL-STI-2009-00316 (Phifer 2009d)
Savannah River Site Composite Analysis: Aquifer Flow Path Parameters (Hamm, Smith, and Phifer 2009)	*Design Review on Aquifer Flow Path Parameters Used in GoldSim™ Based Savannah River Site Composite Analysis Model* (Hamm 2009)
Savannah River Site Composite Analysis: Dose Module (Phifer et al. 2009)	• Neptune and Company, Inc. personnel performed QA checking of the module itself in accordance with the Neptune and Company, Inc. QA Plan entitled, "*Software Quality Assurance Plan for Development of GoldSim™ Models"* (Neptune 2009) • *Design Check for Savannah River Site Composite Analysis: Dose Module,* SRNL-STI-2009-00424, rev. 0, Input Data (Phifer 2009e)
Savannah River Site Composite Analysis: Base Case Deterministic Calculations (Smith et al. 2009a)	• *Design Check of GoldSim™ model used to perform generic Composite Analysis dose calculations* (Smith 2009c) • *Design Check of input data to GoldSim™ model used to perform generic Composite Analysis dose Calculations* (Smith 2009d) • *Review of GoldSim™ CA Model* (Taylor 2009b) • *Summary of Design Checks for CA GoldSim™ Source*, *Release Model Investigations* (Hiergesell et al. 2009)

Savannah River Site Composite Analysis: Sensitivity and Uncertainty Calculations (Smith et al. 2009b)	*Summary of Design Checks for CA Sensitivity and Uncertainty Investigations* (Hiergesell et al. 2009b)

6.5 CA DOCUMENT REVIEWS

As noted in Section 5.1, the preparation of the Composite Analysis was overseen by a Steering Team and a Core Team. The Core Team provided technical oversight, direction, and review of philosophy, concepts, assumptions, input parameters, and data, task proposals, methods, screening, modeling, results, documents, etc. associated with individual tasks. When the CA document was prepared, the Core Team reviewed the first draft, Revision A. Comments provided by the Core Team were incorporated into the next draft, Revision B, as appropriate.

Revision B of the CA document was reviewed by the Department of Energy – Savannah River Operations Office (DOE-SR). DOE-SR assembled a team to review and comment on Revision B. Details of the review team, the review process, the team comments and the disposition of the comments is included in Appendix G. Comments provided by the DOE-SR review team were incorporated into the next draft, Revision 0 – Draft for DOE LFRG Review.

Revision 0 – Draft for DOE LFRG Review was reviewed by the DOE headquarters LFRG Review Team. Details of the review team, the review process, and the team observations are documented in Carilli and Golian (2010). The disposition of the review team observations is provided in Appendix H. Observations provided by the LFRG Review Team were incorporated into this final report, Revision 0, as outlined in Appendix H.

6.6 REFERENCES

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7.0 SCREENING ANALYSIS

Screening analyses were conducted in the Composite Analysis (CA) to focus the dose analysis on the significant pathways of transport and exposure, as well as on the significant radionuclides. These analyses fulfilled the objectives of Tasks 4.2.11, 4.2.12, and 4.2.13 of the Composite Analysis Program Plan (Phifer and Cook 2007). The objective of Task 4.2.11 is to develop preliminary transport modeling input data (hydraulic, geochemical, radionuclide, etc.), which was accomplished in the process of applying the screening model described in Section 7.2. Transport and exposure pathway screening was first carried out, and is described in Section 7.1, in fulfillment of the objectives of Task 4.2.12 of the Program Plan. Once the significant transport and exposure pathways were identified, a screening model was developed to identify radionuclides that are potentially significant contributors to dose, thus allowing for elimination from further consideration in the CA those radionuclides that are insignificant dose contributors, in fulfillment of the objectives of Task 4.2.13 of the Program Plan. This screening model, including the radionuclide screening results, is described in Section 7.2.

7.1 TRANSPORT PATHWAY AND EXPOSURE SCENARIO SCREENING

The purpose of this section is to identify potential pathways for public exposure at every Point of Assessment (POA) (identified in Section 4.2) to radionuclides potentially released from sources of residual radioactive material at the Savannah River Site (SRS), and to justify eliminating some of these pathways from further consideration within the CA. The results are used to evaluate doses to the public potentially received as a result of radionuclides released at SRS. The following discussion was taken largely from Wilhite and Phifer (2008).

7.1.1 Transport Pathway Identification

Radionuclides released to the geosphere have the potential of reaching humans through numerous pathways. Potential pathways for a residual radionuclide source (e.g., subsurface radionuclides in closed waste units and in-situ decommissioned buildings) are indicated in Figure 7-1. Each pathway is briefly defined in the following list (WSRC 2008b):

(1) Leaching - migration of radionuclides from the source by a combination of dissolution, diffusion, and advection.
(2) Gaseous Diffusion - upward migration of gaseous radionuclides from the source by diffusion through the caps and surface soils to the atmosphere.
(3) Irrigation - contamination of surface soil by radionuclides that have reached groundwater which is subsequently used for irrigation.
(4) Deposition - contamination of surface water by radionuclides that have reached the atmosphere; represents deposition of particulate associated radionuclides or gaseous species partitioning at the air-water interface.
(5) Volatilization - partitioning of volatile radionuclide species present in surface water into air above the water body.
(6) Discharge - discharge of radionuclides present in groundwater into surface water.

(7) Recharge - movement of radionuclides into the groundwater from contaminated surface water.
(8) Irrigation - contamination of surface soil by radionuclides that have reached surface water that is being subsequently used for irrigation.
(9) Washload - contamination of surface water by soil containing radionuclides as a result of erosion by rain or irrigation water.
(10) Deposition - contamination of surface soil by radionuclides that have reached the atmosphere and have become associated with airborne particulate matter.
(11) Resuspension - resuspension of soil-associated radionuclides as a result of wind erosion.
(12) Biointrusion - contamination of surface soil by soil-associated radionuclides that are brought to the surface from the vicinity of the source by burrowing animals, such as rodents or ants, or by intruding plant roots.
(13) Deposition - deposition of radionuclides in surface water that have partitioned onto suspended sediment.
(14) Resuspension - resuspension of particulate-borne radionuclides in the sediment of surface water as a result of hydrodynamic forces at the sediment-water interface.
(15) Immersion - contamination of aquatic plants by radionuclides in surface water attributable to the immersion of the plants in the contaminated water.
(16) Immersion - human exposure to radionuclides as a result of immersion in contaminated surface water.
(17) Ingestion - human exposure to radionuclides as a result of ingestion of radionuclides present in surface water.
(18) Ingestion - contamination of terrestrial animals from their ingestion of radionuclides in surface water. Note that terrestrial animals include birds.
(19) Ingestion - contamination of terrestrial animals from their ingestion of radionuclides in groundwater.
(20) Irrigation - contamination of terrestrial plants as a result of irrigation with surface water containing radionuclides.
(21) Irrigation - contamination of terrestrial plants as a result of irrigation with groundwater containing radionuclides.
(22) Decomposition - contamination of surface soil as a result of decomposition of terrestrial plants in the soil.
(23) Root uptake - contamination of terrestrial plants by uptake through roots of soil water containing radionuclides.
(24) Deposition - deposition of airborne radionuclides onto terrestrial plant surfaces.
(25) Ingestion - ingestion of radionuclides by grazing animals as a result of contaminated soil ingestion.
(26) Ingestion - ingestion of radionuclide-containing vegetation by terrestrial animals.
(27) Decomposition - contamination of surface soil as a result of decomposition of terrestrial animals in the soil.
(28) Washoff - contamination of surface soil as a result of washoff of externally contaminated terrestrial animals.
(29) Resuspension - resuspension of surficial radionuclides on terrestrial animals to the atmosphere.

(30) Resuspension - resuspension of surficial radionuclides on terrestrial plants to the atmosphere.
(31) Inhalation - contamination of terrestrial animals as a result of inhalation of radionuclides in the atmosphere.
(32) Deposition - surface contamination of terrestrial animals via deposition of particulate-borne radionuclides in the atmosphere.
(33) Ingestion - contamination of terrestrial animals as a result of their ingestion of aquatic animals.
(34) Decomposition - contamination of surface water sediment as a result of decomposition of aquatic plants in the sediment.
(35) Decomposition - contamination of surface water sediment as a result of decomposition of aquatic animals in the sediment.
(36) Surface contact - surface contamination of aquatic animals as a result of contact with contaminated sediment.
(37) Root uptake - contamination of aquatic flora via radionuclide uptake through roots.
(38) Immersion - contamination of aquatic animals as a result of immersion in surface water containing radionuclides.
(39) Ingestion - contamination of aquatic animals as a result of their ingestion of aquatic plants containing radionuclides.
(40) Ingestion - human exposure to radionuclides as a result of ingestion of contaminated aquatic flora.
(41) Ingestion - human exposure to radionuclides as a result of ingestion of contaminated groundwater.
(42) Inhalation - human exposure to radionuclides as a result of inhalation of airborne radionuclides.
(43) Immersion - human exposure to radionuclides as a result of immersion in contaminated air.
(44) Ingestion - human exposure to radionuclides as a result of ingestion of contaminated terrestrial animals.
(45) Ingestion - human exposure to radionuclides as a result of ingestion of contaminated terrestrial plants.
(46) Ingestion - human exposure to radionuclides as a result of ingestion of contaminated aquatic animals containing radionuclides.
(47) Washoff - contamination of surface soil below vegetation due to rain-induced surface washoff.
(48) Direct exposure to surface water sediment - external radiation from sediment.
(49) Excretion - transfer of contamination to soil by excretion from terrestrial animals.
(50) Direct exposure to soil - external radiation from soil.

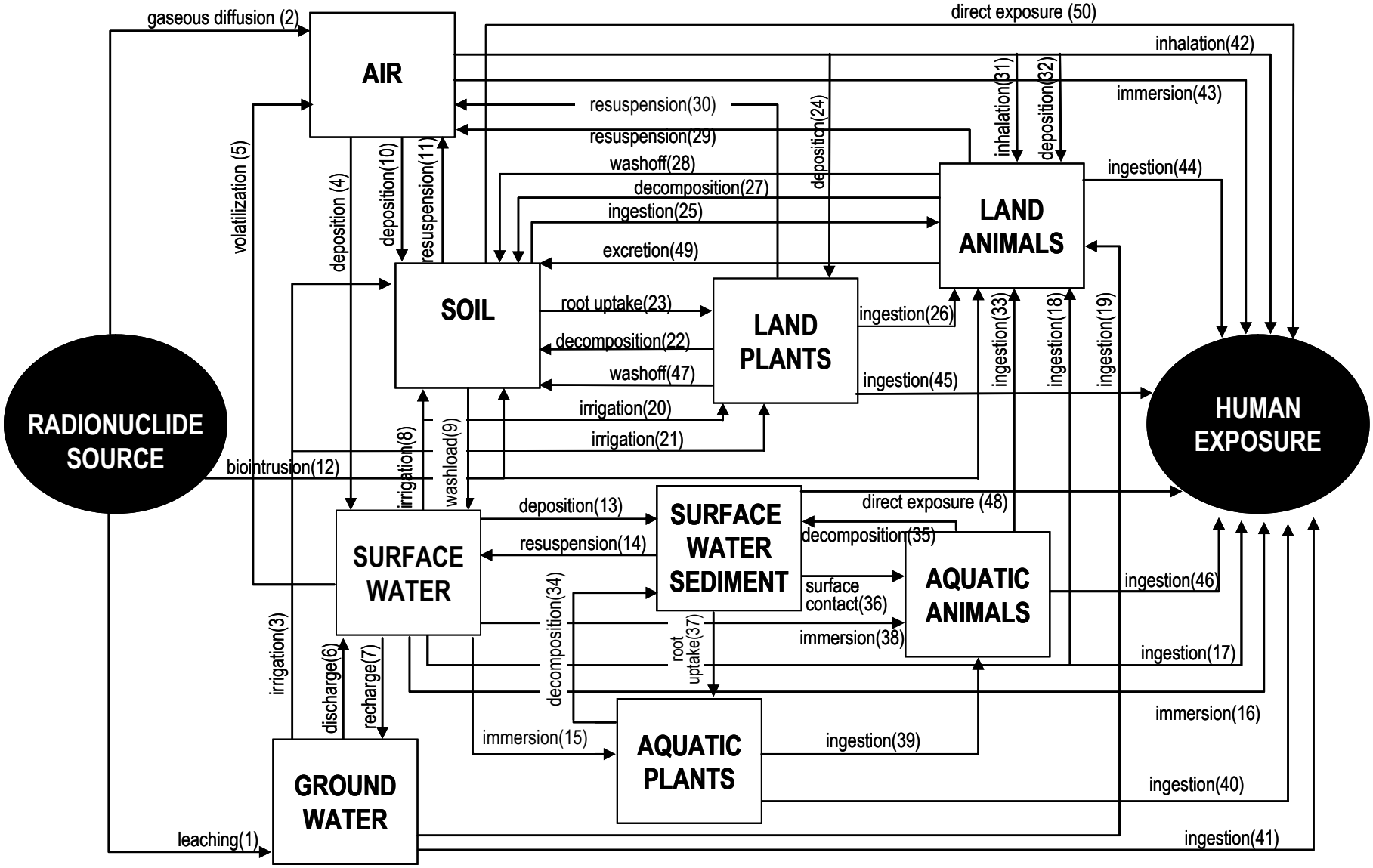


Figure 7-1. Potential Pathways to Human Exposure from Undisturbed Residual Radioactive Material

7.1.2 Transport Pathway Screening

The pathway list is generic in nature, and the significance of each pathway must be evaluated on a site-specific and CA-specific basis to develop transport and exposure models. Many pathways may be removed from consideration for the CA because of a negligible contribution to public exposure.

For the SRS, leaching and transport of radionuclides to the groundwater (pathway (1)) is the predominant means that radionuclides may be subsequently transported in the environment. Thus, this pathway must be addressed in developing a transport model. However, for the CA, the use of contaminated groundwater by the public is precluded by SRS land use expectations (Phifer and Wilhite 2008). All potential points of assessment involve the use of contaminated surface water rather than groundwater. Thus, use of contaminated groundwater is not considered in the CA, which excludes pathways (3), (19), (21), and (41). Irrigation with contaminated surface water may lead to contamination of agricultural crops and animals (pathways (8), (20), (23), (25) and (26)), which must be included in the exposure models.

Human exposure may occur as a result of direct human ingestion of contaminated surface water (pathway (17)), or as a result of consumption of contaminated food supplies (pathways (44) and (45)). Direct consumption of surface water by land animals (pathway (18)) is considered in the dose analysis, but indirect transport pathways between soil, plants and animals (pathways (22), (27), (28), and (47)) are not considered because they are judged to be insignificant. Two of these pathways involve washoff of external contamination from land animals (28) and plants (47) to surface soil. In the dose analysis, all irrigation water is assumed to be applied directly to the soil, so that none is lost by first contacting plants or animals (plants are also assumed to be exposed directly to irrigation water to account for external contamination of plant surfaces that are then ingested by humans). The other two pathways involve transfer of radionuclides to surface soil by decomposition of plants (22) and animals (27). Allowing animal carcasses to decompose in the garden soil is not a realistic pathway. Although some parts of plants may be left in the ground to decompose, it is judged that this is a minor addition to the surface soil compared to that from irrigation. Transfer of contamination from animals (including birds) to soil by excretion (pathway (49)) is not considered because it is judged to be insignificant. Additionally, human exposure may occur via external radiation from surface water sediment (pathway (48)) or soil (pathway (50)); these pathways are included in the set of exposure pathways (Section 7.1.3).

Discharge of contaminated groundwater to surface water (pathway (6)) may result in contamination of the aquatic ecosystem including the water body itself, sediment, and aquatic plants and animals (pathways (13), (14), (15), (34), (35), (36), (37), (38), and (39)). However, recharge of contaminated surface water to groundwater (pathway (7)) is conservatively not considered to enhance concentrations of radionuclides in surface water. Volatilization of radionuclides from contaminated surface water into the air (pathway (5)) is judged to be insignificant. Volatilization of tritium, the most potentially volatile radionuclide likely to be in a source considered in the CA, from a radionuclide source is shown in the paragraphs below to be insignificant; thus, volatilization from a secondary source such as surface water is expected to be insignificant. Also, contamination of surface water by contaminated soil as a result of erosion (pathway (9)) is judged to be insignificant, because buildup of radionuclides in surface soil would only result from radionuclides with high sorption potential. These radionuclides would not partition readily into the surface water if introduced as a result of erosive events. Ingestion of contaminated surface water by terrestrial animals (pathway (18)) may lead to human exposure. However, ingestion of aquatic animals by land animals (pathway (33)) is judged to be insignificant. Human exposure may occur as a result of immersion in contaminated surface water during recreational activities such as swimming (pathway (16)) or direct ingestion of surface water (pathway (17)).

Consumption of contaminated aquatic plants (pathway (40)) was not considered because there is no indication that aquatic plants present in potentially-contaminated surface water in the vicinity of the SRS are consumed by humans. However, consumption of aquatic animals (i.e., fish) (pathway (46)) is considered.

Volatilization and transport of radionuclides in the gaseous phase to the ground surface and, subsequently, the atmosphere (pathway (2)) is another means that radionuclides could be transported in the environment. However, tritium monitoring data for the SRS were considered to put the volatilization pathway into perspective. In the original CA (WSRC 1997a), tritium monitoring data for 1995 were used. In 1995, the SRS released 55,000 Ci of tritium oxide from all areas of the site, with an estimated dose of 0.06 mrem to the maximally-exposed individual at the site boundary (Table 7-1). Most of the tritium oxide (42,000 Ci) came from the General Separations Area (GSA). The 42,000 Ci released in 1995 from the GSA is greater than the estimated residual inventory for the F-Area and H-Area facilities (WSRC 1997a). Therefore, the atmospheric dose that would be calculated for the residual inventory in these facilities would be less than 0.06 mrem/yr, even if the entire residual inventory were released in one year.

The estimated inventory of tritium in the Old Radioactive Waste Burial Ground (ORWBG), the Mixed Waste Management Facility (MWMF) and the E-Area Low-Level Waste Facility (ELLWF) is 6.9×10^6 Ci, (WSRC 1997a) about 125 times the total amount released from all of SRS in 1995. Using a simple ratio and assuming all tritium oxide, if this entire inventory were released in one year, the calculated dose to the maximally-exposed individual would be 7.5 mrem/yr (because the 0.06 mrem/yr dose was based on the 55,000 Ci SRS release) (WSRC 1997a).

Though several million curies of tritium have been disposed in E Area since the mid 1950s, the air monitoring results are so low that E Area is classified as having no air emission sources. Because of decay, infiltrating water sweeping through the vadose zone, and release to the groundwater, little, if any, tritium activity is released to the atmosphere.

As shown in Table 7-1, annual atmospheric releases of tritium and estimated dose to a member of the public from the releases continue to be on the order of those observed in 1995. Thus, transport pathways involving release of radionuclides to the atmosphere from a source (pathway (2)), deposition from the atmosphere (pathways (4), (10), (32)) and inhalation (pathway (42)) or immersion (pathway (43)) are excluded. However, exposure to humans from radionuclides resuspended in air as a result of gardening activities (pathways (42) and (43)) are included in the dose analysis.

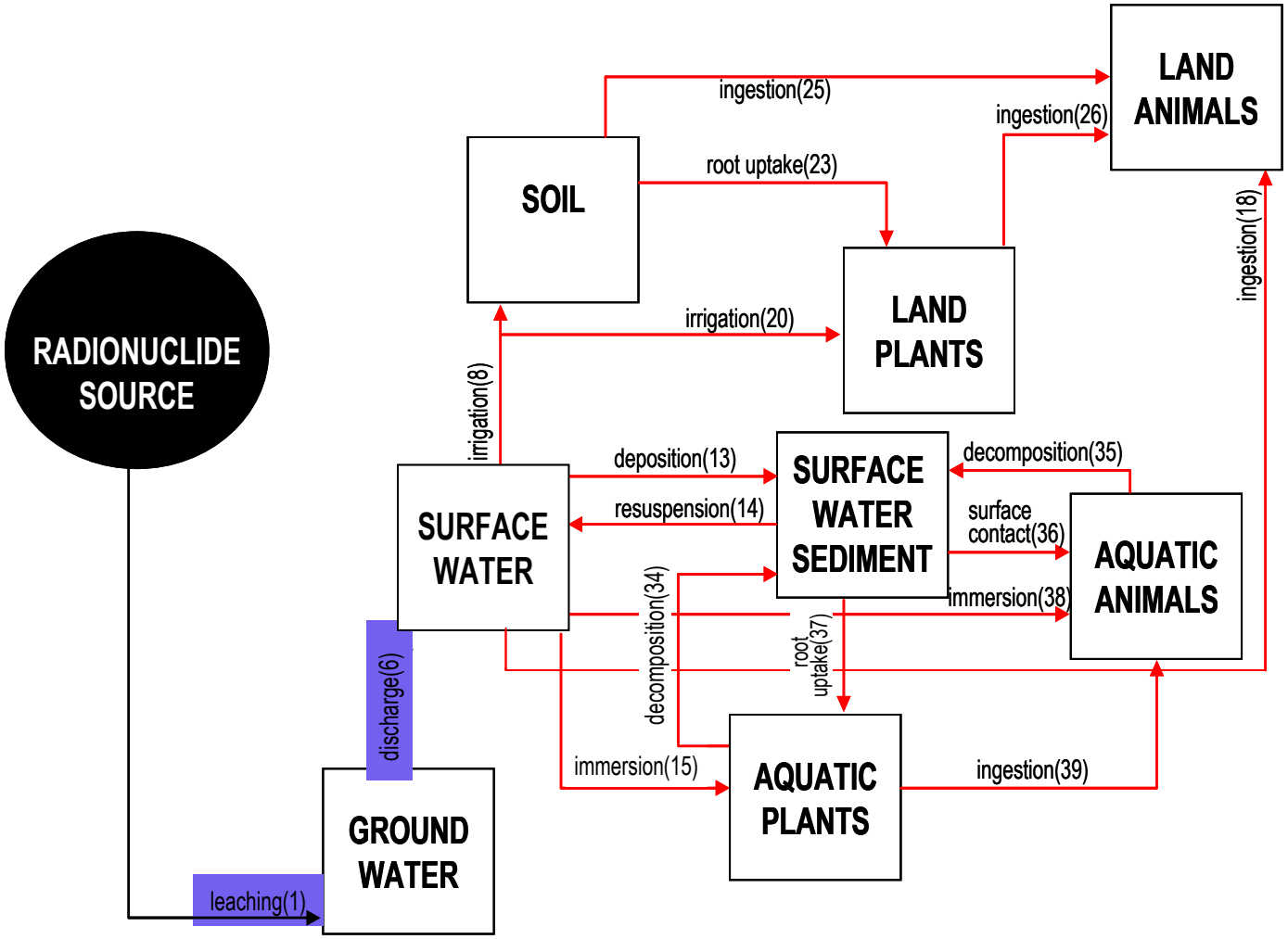
Deposition of radionuclides from the atmosphere to surface soil (pathway (10)) and resuspension of contaminated soil particles (pathway (11)) are considered in the dose analysis as is deposition from the atmosphere onto land plants (pathway (24)). Inhalation (pathway (31)) by and deposition (pathway (32)) onto land animals and resuspension from land plants (pathway (30)) and land animals (pathway (29)) are not considered because they are judged to be insignificant. As shown above, volatilization as a primary pathway is not significant for the CA. Therefore, these secondary pathways involving volatilization would also be insignificant.

Finally, contamination of surface soil over a source as a result of biointrusion of burrowing animals or plant roots (pathway (12)) must be addressed. It is acknowledged that biointrusion is a potentially significant pathway of contamination of surface soil over a source, as is concluded in a study by McKenzie et al. (1983). For the humid southeast, where ground surface and soil moisture limit resuspension of soil, biointrusion is likely to result in contamination of soils over the source, but probably not significant contamination off-site. Thus, the Biointrusion pathway (pathway (12)) is not considered further. Extensive experience at SRS with closed waste sites shows no radioactive material being brought to the surface of the site.

Table 7-1. Tritium Release to the Atmosphere at SRS and Resulting Dose.

Year	Total H-3 Release, Ci	H-3 Oxide Release, Ci	Dose at SRS Boundary, mrem	Reference
1995	97,000	55,000	0.06	WSRC 1995
1996	55,300	40,100	0.05	WSRC 1996
1997	58,000	39,100	0.05	WSRC 1997b
1998	82,700	58,600	0.07	WSRC 1998
1999	51,600	33,900	0.06	WSRC 1999
2000	44,800	32,400	0.06	WSRC 2000
2001	47,400	33,000	0.05	WSRC 2001
2002	47,300	32,100	0.06	WSRC 2002
2003	50,800	31,700	0.07	WSRC 2004
2004	61,300	42,000	0.06	WSRC 2005
2005	40,800	32,700	0.05	WSRC 2006
2006	34,600	30,900	0.11	WSRC 2007

In summary, only two pathways are considered to be of primary consequence for transport of radionuclides from the disposed waste into the environment. These are leaching of the source resulting in contamination of groundwater local to the sources (pathway (1)) and transport via groundwater to surface streams (pathway (6)). These pathways are highlighted in blue on Figure 7-2. Subsequent transport pathways are delineated by red lines on Figure 7-2. Of the original 50 pathways identified in Figure 7-1, only 17 are potentially significant. These pathways will be assessed in the CA radionuclide transport modeling.



The primary transport pathways are highlighted in blue, secondary transport pathways are shown by red lines.

Figure 7-2. Transport Pathways from Residual Radioactive Material

7.1.3 Exposure Pathway Screening

For calculating exposure to humans, pathways resulting in contamination of agricultural crops and animals as a result of irrigation with contaminated surface water (pathways (8), (23), (25), (26), (44), and (45)) and deposition or inhalation from the atmosphere (pathways (24), (42), and (43)) as a result of gardening activities, as well as direct ingestion of contaminated surface water (46), and external radiation from surface water sediment (pathway (48) and soil (pathway 50)) will be considered in the dose analysis for the CA. These pathways are shown on Figure 7-3.

Because the CA points of assessment are assumed to be at the mouths of the SRS streams and in the Savannah River (Phifer and Wilhite 2008), exposure scenarios involving contact with, and use of, contaminated surface water (i.e., stream or river water) were considered. Two exposure scenarios were judged to bound exposures, a recreational scenario and a residential scenario. The recreational scenario involves activities such as boating, swimming, fishing, shoreline exposure, and inadvertent ingestion of surface water. The residential scenario involves use of contaminated surface water for drinking and other household uses, and for irrigation. The irrigation results in production of garden crops (e.g., vegetables) and animal products (e.g., milk, meat). Consumption of vegetables also results in consumption of soil associated with the vegetables.

These two exposure scenarios are consistent with those assessed in recent SRS Performance Assessments (PAs); in the PA for the ELLWF (WSRC 2008b), for the F-Area Tank Farm (FTF, WSRC 2008a), and for the Saltstone Disposal Facility (LWO 2009). These exposure scenarios are also consistent with those assessed in the original SRS CA (WSRC 1997a).

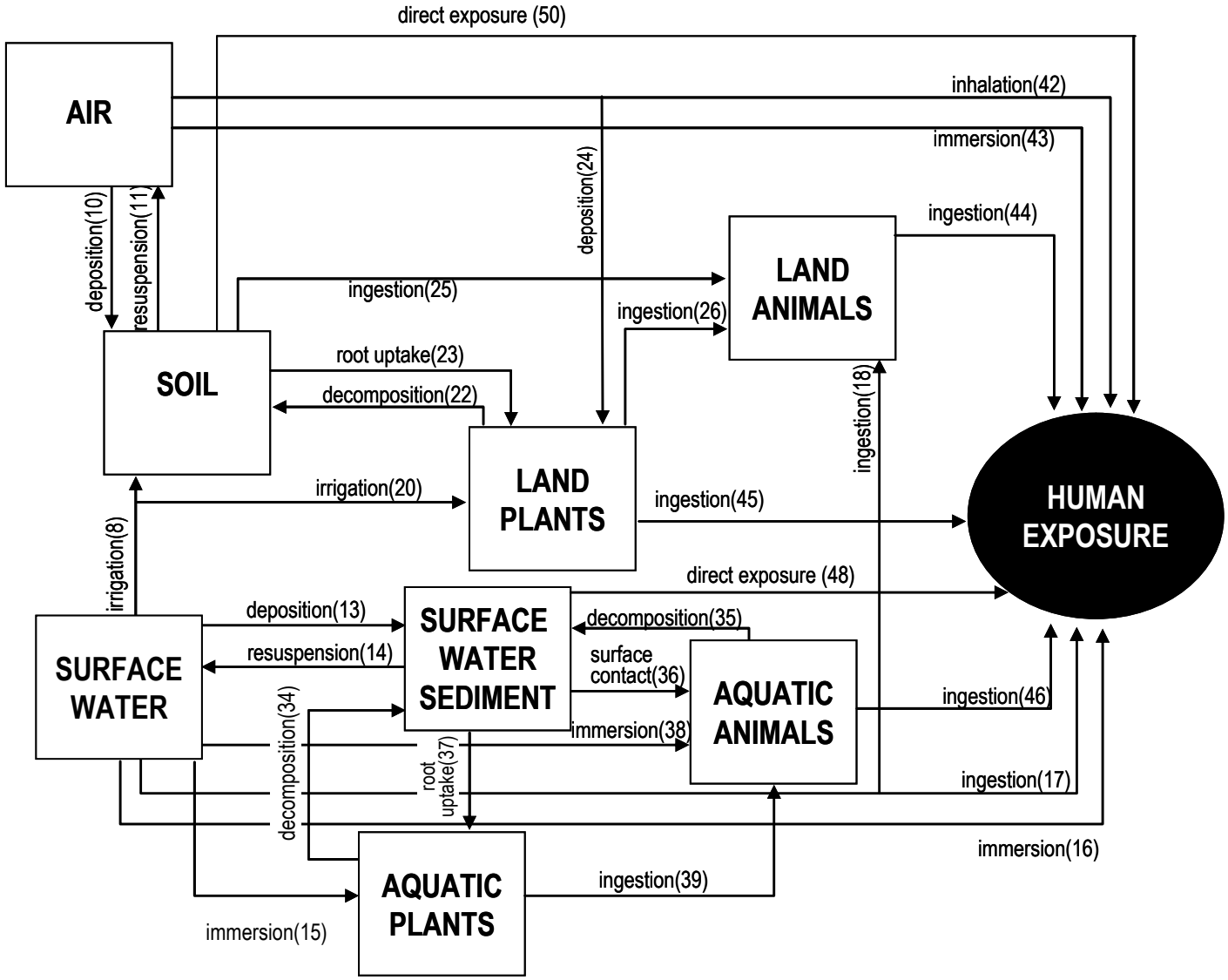


Figure 7-3. Pathways to Human Exposure from Radioactive Material Transported Through the Environment from an Undisturbed Source

7.2 RADIONUCLIDE SCREENING

A crucial step to be performed early in the CA effort was the determination of radionuclides of most interest. Although potentially, there are thousands of radionuclides to consider, performing screening allows for the number of radionuclides considered for the fate and transport analysis to be reduced to a reasonable number, thus eliminating from further consideration radionuclides that are insignificant contributors to the total dose. This section discusses the radionuclide screening methodology applied in the CA and the screening results. The source of preliminary modeling input data for this model is also described. The following discussion was taken largely from Taylor et al. (2008).

7.2.1 Approach to Radionuclide Screening

The approach taken in this radionuclide screening analysis is an extension of the National Council on Radiation Protection and Measurements (NCRP) methodology as described in NCRP 1996. The reference describes a simplified approach for calculating the dose to a receptor from groundwater. The tools that we have available at Savannah River National Laboratory (SRNL) make it possible for us to do a more rigorous analysis while adhering to the spirit of the NCRP's methodology. The screening analysis starts with an arbitrarily large number of curies of a radionuclide (i.e., 1E+07 Ci) directly in the ground (i.e., no credit is taken for the waste form, such as a concrete slab) which is then transported through representative unsaturated and saturated regions to a surface water body. The POAs used for screening are assumed to be at a stream mouth entering the Savannah River so transport will occur in that representative stream (Phifer and Wilhite 2008). The dose is calculated based on the stream mouth's radionuclide concentration and does not consider mixing in the river. The dose is then compared to the desired screening criterion to determine which radionuclides warrant further, detailed analysis (i.e., those radionuclides whose calculated dose exceeds the screening criterion).

7.2.2 Radionuclide Screening Methodology

This section provides details of implementation of the approach outlined in Section 7.2.1.

7.2.2.1 Points of Assessment

As outlined within Phifer and Wilhite 2008, the primary POAs for this CA are taken as the mouths of Upper Three Runs Creek and Fourmile Branch at the Savannah River and within the Savannah River (Section 4.2). This definition of the POAs for the CA forms the basis for conducting the radionuclide screening with the POA assumed at a stream mouth entering the Savannah River.

7.2.2.2 Transport and Exposure Pathways

As described in Section 7.1.2, the transport pathways to be analyzed in the CA are the leaching of radionuclides from the residual radioactive materials and transport through the unsaturated zone to the saturated zone and, then, transport through the saturated zone to surface streams. Radionuclides are transported via surface streams to the points of assessment. Also, as described in Section 7.1.3, the exposure pathways to be assessed in the CA are those involving use of surface water. Both recreational (e.g., boating, fishing) and residential (e.g., gardening, drinking) scenarios related to these pathways are assessed.

7.2.2.3 Model Implementation

The CA screening model was implemented in the GoldSim™ programming environment; specifically Version 9.60 SP4 (Service Pack 4) (GTG 2007). The model (i.e., CA Screening 2.002.gsm) takes advantage of previous modeling work. The dose module and material properties were developed for the FTF PA (WSRC 2008a). These were modified to account for the additional radionuclides and the simplified flow path to be analyzed in the screening. Discussions of implementation refer to the GoldSim™ implementation.

7.2.2.4 Conceptual Model

The conceptual model for the CA radionuclide screening is very simple. Each source is represented as radionuclides in a soil matrix (i.e., no credit is taken for waste form, such as concrete). The source receives water as infiltration. Radionuclides partition between the soil and the soil pore water in accordance with an element-specific distribution (or sorption or partition) coefficient (K_d). Transport is simulated through the unsaturated (i.e., vadose) zone to the water table, through the water table to the outcrop at a surface stream, and through the surface stream to the stream mouth at the Savannah River. Reactions such as sorption are not considered in the stream; the stream serves only to dilute the radionuclide concentration. The stream flow rates used to dilute the radionuclide concentrations are flow rates associated with SRS streams rather than the Savannah River (see Table 7-2 and Table 9-17 for a comparison).

7.2.2.5 Transport

A simple transport flow model is used for this analysis. All flows are assumed to be steady-state; therefore, the model uses Pipe elements. Inlet and outlet flows are specified for each of the elements based on representative values to be discussed below. Figure 7-4 shows the transport container of the model. A river flow path is included, but the reported results are from the *Stream* pipe element output, which represents radionuclide concentrations at the mouth of the stream (i.e., dilution from SRS streams but prior to dilution in the Savannah River).

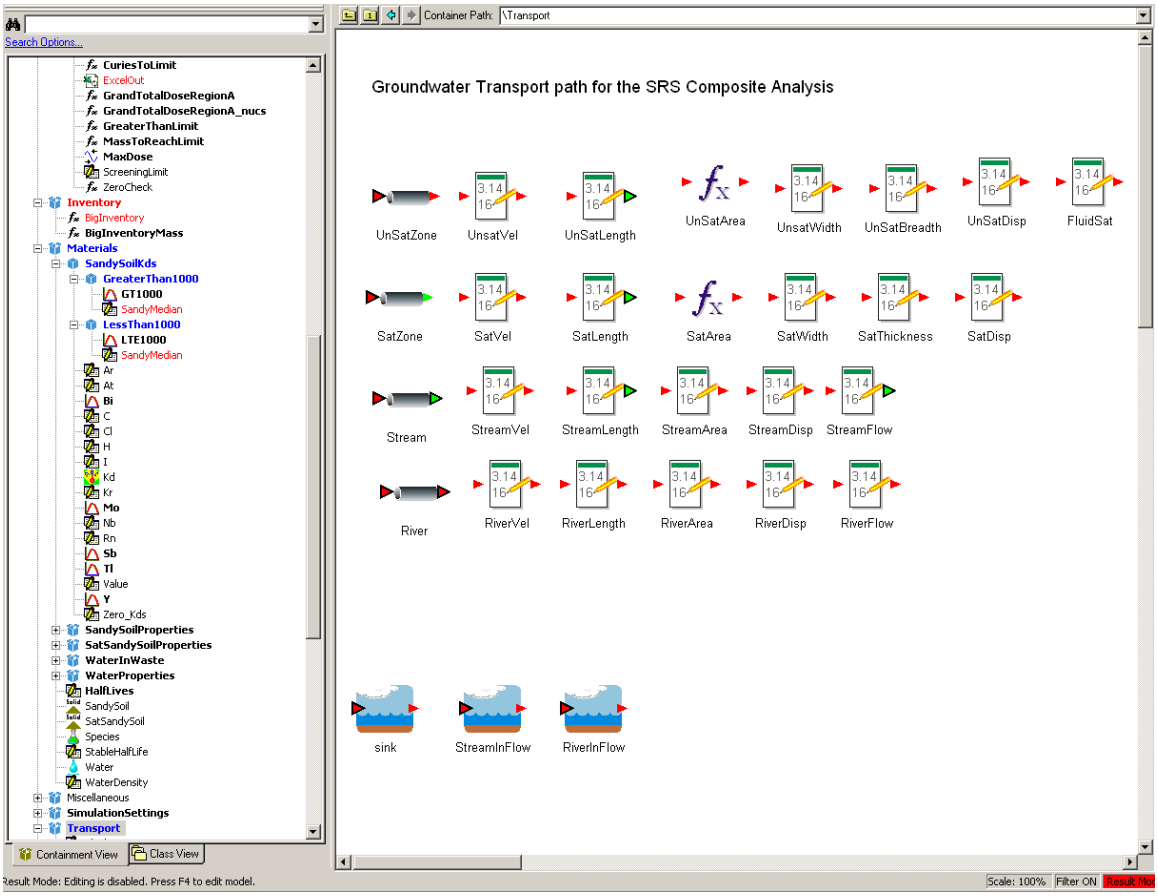


Figure 7-4. Transport Container

7.2.2.5.1 Infiltration and Unsaturated Zone Flow

A representative infiltration flow of 15 inches/yr was chosen for this analysis, based upon background infiltration studies (Phifer et al. 2007). This is typical of flow rates used in other analyses, such as the FTF PA (WSRC 2008a), when no cover is assumed. The unsaturated zone thickness (i.e., the depth to groundwater) was selected to range between 10 feet and 30 feet, which are typical values for many subsurface sources at the SRS.

7.2.2.5.2 Saturated Zone Flow

The saturated zone flow velocity was selected to be 150 feet per year, which is representative of values used in the ELLWF and FTF PAs. The saturated zone flow length (i.e., the distance from the radionuclide source to the outcrop into a surface stream) was varied between 100 and 762 meters, which are typical for many sources.

7.2.2.5.3 Stream Flow

A study was conducted to determine reasonable, conservative flow rates for the streams on the SRS. The results of that study, based on data from United States Geological Survey (USGS) gauging station data, are shown in Table 7-2. As will be shown in Section 7.2.3, the results are basically insensitive to these flow rate values.

Table 7-2. Conservative Stream Flow Rates

Stream	USGS Gage Station	Stream flow (cfs)	Time Frame
Upper Three Runs at Road A [1]	02197315	175	1992-1996
Fourmile Creek A12 (near D-Area) [1]	02197344	10	1992-1996
Pen Branch A13 (near the Savannah River Swamp; see Figure 3-11) [1]	02197348	25	1992-1996
Steel Creek at Highway 125 [1]	021973565	50	1992-1996
Lower Three Runs at Martin, SC [1]	02197415	40	2000-2002
Savannah River at Jackson, SC [2]	02197320	3,220	1972-1995

[1] WSRC 1999b
[2] Wike et al. 2006

7.2.2.5.4 Distribution Coefficients (K_d)

The K_ds used in the GoldSim™ screening model are element-specific and are shown in Table 7-3. The values used are documented in Appendix A of Taylor et al. (2008). The selection of values was made based on the following priority. The Kaplan (2006) report is considered the best available source of values for the SRS CA screening model. This report provides an in-depth review of sorption coefficients appropriate for various materials encountered by radionuclides in subsurface waters. The values selected from the Kaplan report are those for sandy sediment, as they tend to be lower than the clayey sediment K_d values, and thus provide a more conservative estimate of water concentration (i.e., less sorption is assumed).

The source of K_ds considered second-best for use in the CA screening model are those recommended for screening calculations in NCRP Report No. 123 (NCRP 1996). These values are based on a 1992 compilation by Kennedy and Strenge (1992). Values were selected from the NCRP report when not available in Kaplan (2006). Elements for which K_ds were not available in Kaplan (2006) or NCRP (1996) were assigned a default value of zero. For radioisotopes of these elements, sorption is assumed to be negligible, and transport is thus maximized.

Table 7-3. Element-Specific Kds used in the CA Screening Model

Element	Kd (ml/g)	Source[a]	Element	Kd (ml/g)	Source[a]	Element	Kd (ml/g)	Source[a]
Ac	1100	Kaplan 2006	Gd	1100	Kaplan 2006	Pt	0	Default
Ag	90	NCRP 1996	Ge	0	Default	Pu	270	Kaplan 2006
Al	0	Default	H	0	Kaplan 2006	Ra	5	Kaplan 2006
Am	1100	Kaplan 2006	Hf	0	Default	Rb	50	Kaplan 2006
Ar	0	Kaplan 2006	Hg	19	NCRP 1996	Re	0.1	Kaplan 2006
As	110	NCRP 1996	Ho	240	NCRP 1996	Rh	52	NCRP 1996
At	0	Kaplan 2006	I	0	Kaplan 2006	Rn	0	Kaplan 2006
Au	30	NCRP 1996	In	390	NCRP 1996	Ru	55	NCRP 1996
Ba	5	Kaplan 2006	Ir	91	NCRP 1996	S	14	NCRP 1996
Be	240	NCRP 1996	K	18	NCRP 1996	Sb	45	NCRP 1996
Bi	120	NCRP 1996	Kr	0	Kaplan 2006	Sc	310	NCRP 1996
Bk	1100	Kaplan 2006	La	1200	NCRP 1996	Se	1000	Kaplan 2006
Br	14	NCRP 1996	Lu	0	Default	Si	0	Default
C	0	Kaplan 2006	Md	0	Default	Sm	1100	Kaplan 2006
Ca	8.9	NCRP 1996	Mg	0	Default	Sn	2000	Kaplan 2006
Cd	40	NCRP 1996	Mn	50	NCRP 1996	Sr	5	Kaplan 2006
Ce	500	NCRP 1996	Mo	10	NCRP 1996	Ta	0	Default
Cf	1100	Kaplan 2006	N	0	Default	Tb	240	NCRP 1996
Cl	0	Kaplan 2006	Na	76	NCRP 1996	Tc	0.1	Kaplan 2006
Cm	1100	Kaplan 2006	Nb	0	Kaplan 2006	Te	1000	Kaplan 2006
Co	7	Kaplan 2006	Nd	240	NCRP 1996	Th	900	Kaplan 2006
Cr	30	NCRP 1996	Ne	0	Default	Ti	0	Default
Cs	50	Kaplan 2006	Ni	7	Kaplan 2006	Tl	390	NCRP 1996
Cu	30	NCRP 1996	Np	0.6	Kaplan 2006	Tm	0	Default
Dy	0	Default	O	0	Default	U	200	Kaplan 2006
Er	0	Default	Os	190	NCRP 1996	V	0	Default
Es	0	Default	P	8.9	NCRP 1996	W	100	NCRP 1996
Eu	1100	Kaplan 2006	Pa	0.6	Kaplan 2006	Xe	0	NCRP 1996
F	87	NCRP 1996	Pb	2000	Kaplan 2006	Y	190	NCRP 1996
Fe	160	NCRP 1996	Pd	52	NCRP 1996	Yb	0	Default
Fm	0	Default	Pm	240	NCRP 1996	Zn	200	NCRP 1996
Fr	50	Kaplan 2006	Po	2000	Kaplan 2006	Zr	900	Kaplan 2006
Ga	0	Default	Pr	240	NCRP 1996			

[a] A default value of zero was conservatively selected for the screening model for those elements not listed in Kaplan (2006) or the NCRP (1996)

7.2.2.5.5 Source Term

The source was assumed to be an area 100 meters by 10 meters (i.e., 1,000 m^2). The source term was implemented as a Cumulative Input to the *UnSatZone* pipe element. Each radionuclide was treated as a parent and each parent had its complete decay chain (as appropriate). An initial inventory of ten-million curies of each radionuclide was chosen for this screening analysis. This value was used for radionuclide screening in the ELLWF PA (WSRC 2008b). Preliminary radionuclide inventory data for a number of sources have been developed and are shown in Table 7-4. These inventories represent the sum of the inventories of all radionuclides in each source. The largest inventories are about six and five million curies for the ORWBG and the MWMF, respectively. All of the other sources are less than one-million curies. Thus, the assumed inventory of ten-million curies of each radionuclide will bound all the sources to be assessed in the CA. The data in Table 7-4 is not used further in the CA.

Table 7-4. Preliminary CA Source Data

Facility	Total Activity (Ci)	Facility	Total Activity (Ci)
ORWBG	5.81E+06	211-F	5.98E+02
MWMF	4.67E+06	F-Retention Basin	1.02E+02
H-Area Seepage Basins	3.66E+05	Old F-Seepage Basin	2.67E+01
F-Area Seepage Basins	3.46E+05	MIPSL[a]	1.50E+01
H-Area Tank Farm	2.76E+05	SRL Seepage Basins	1.70E+00
F-Area Tank Farm	2.69E+05	420-D	9.32E-01
Tims Branch and Steeds Pond	4.35E+04	421-2D	1.21E-01
FIPSL[a]	4.34E+03	420-2D	1.04E-01
HIPSL[a]	2.92E+03	421-D	3.32E-02
M-Area HWMF[b]	2.08E+03	772-D	1.53E-02
H-Area Retention Basins	1.55E+03	Ford Bldg. Seepage Basin	6.00E-03

[a] MIPSL = M-Area Inactive Process Sewer Line, FIPSL = F-Area Inactive Process Sewer Line, HIPSL = H-Area Inactive Process Sewer Line
[b] HWMF = Hazardous Waste Management Facility

A total of 849 radionuclides were initially considered. The list of 849 radionuclides was derived from the 826 radionuclides listed in the NCRP Report No. 123, Vol. 1 (NCRP 1996), and an additional 23 radionuclides that were identified from other radionuclide listings (not necessarily radionuclides pertinent to activities at the SRS) in the High-Level Waste Tank Farm Closure screening analysis (Hamm 2006). The additional 23 radionuclides are the following: Ar42, As79, Bi208, Ce142, Cf247, Es255, Fe52m, Ga74, Gd150, Nb92, Nd144, Os186, Pb204, Po209, Po212m, Rh105m, Se79m, Sm148, Sm149, Te118, Tl210, V50, and Yb165. Of these, eighteen have half-lives that are either very short (less than one year) or very long (>10E+15 yr) such that the radionuclides are essentially stable. The remaining five radionuclides (Ar42, Bi208, Gd150, Nb92, and Po209) have half-lives ranging from 33 yr to 3.5E+07 yr. There is no evidence that these radionuclides are produced in quantities that would be significant with respect to dose limits at the SRS; thus, these radionuclides are excluded from further consideration. The remaining 826 radionuclides, which are considered as parents in the screening model, are documented, along with their requisite data, in Appendix A of Taylor et al. (2008). Radionuclide decay data (e.g., half-lives, branching fractions) are taken from Tuli (2005).

7.2.2.6 Dose

The dose of interest for this screening analysis is the dose to a hypothetical member of the public (MOP) who resides at the mouth of an SRS stream (Section 7.2.2.1). The MOP is assumed to engage in recreational activities (e.g., boating, fishing) in and adjacent to the stream mouth (see Section 7.1.3) which is the exposure scenario assessed in the original SRS CA (WSRC 1997a). The MOP is also assumed to engage in farming activities (e.g., raising and consuming vegetables, meat, milk) using stream water for irrigation and for household activities (Section 7.1.3). These assumptions are very conservative, because farming by the public can only be conducted on the Georgia side of the Savannah River and therefore water for farming would not be water from the mouth of an SRS stream prior to dilution with the Savannah River on the South Carolina side of the river, but rather, water from the Savannah River itself with its increased dilution. The screening model computes the individual radionuclide dose to the MOP for the given inventory (i.e., 1E+07 Ci) for each parent radionuclide (i.e., it is not a cumulative dose from all radionuclides).

7.2.2.6.1 <u>Radionuclide Concentration used for Dose</u>

The radionuclide concentration used in the dose calculation is the concentration at the outlet of the GoldSim™ *Stream* element (Figure 7-4). This corresponds to the concentration at the stream mouth prior to mixing with river water.

7.2.2.6.2 Dose Pathways

The only dose pathway considered in this screening is the dose to a MOP from the use of contaminated surface water (Section 7.1.2 and Wilhite and Phifer 2008). Details of the dose calculations are documented in the FTF PA (WSRC 2008a). The following figures (i.e., Figure 7-5 through Figure 7-9) provide a graphical representation of the calculation in GoldSim™ and provide a summary of the sub-pathways considered. All of the parameters for the dose calculation (e.g., dose conversion factors, soil-to-plant transfer factors) are documented in Appendix A of Taylor et al. (2008).

Doses are calculated assuming stream water is used for recreational and domestic purposes. The following exposure pathways involving the use of contaminated surface water (from the applicable stream) for recreational use are assumed to occur:

- Direct irradiation during recreational activities (e.g., swimming, fishing) from stream water
- Dermal contact with stream water during recreational activities (e.g., swimming, fishing)
- Incidental ingestion and inhalation of stream water during recreational activities
- Ingestion of fish from the stream water

The LADTAP_Regions Container in Figure 7-6 addresses the calculational requirements for these recreational pathways. Figure 7-7 shows the various calculational elements of the LADTAP_Regions Container.

The following exposure pathways involving the use of surface water (from the applicable stream) for domestic purposes are assumed to occur:

- Direct ingestion of stream water
- Ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that drink stream water
- Ingestion of vegetables grown in garden soil irrigated with stream water
- Ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that eat fodder from pasture irrigated with stream water
- Ingestion and inhalation of stream water while showering
- Inhalation of stream water used for irrigation
- Inhalation of dust from the soil that was irrigated with stream water
- Ingestion of soil that was irrigated with surface water
- Direct radiation exposure from radionuclides deposited on the soil that was irrigated with stream water

The IRRIDOSE Container in Figure 7-6 addresses the calculational requirements for the domestic pathways associated with ingestion and direct exposure. Figure 7-8 shows the various calculational elements of the IRRIDOSE Container. The INHALATION Container in Figure 7-6 addresses the calculational requirements for the domestic pathways associated with inhalation, and Figure 7-9 shows the various calculational elements of the INHALATION Container.

The GoldSim™ *ExcelOut* element (Figure 7-6) is used to transfer the total dose for each radionuclide to an Excel workbook. A macro in the workbook parses the data and selects all radionuclides which exceed the specified screening criterion (i.e., those that are not screened out) and writes them to a summary sheet.

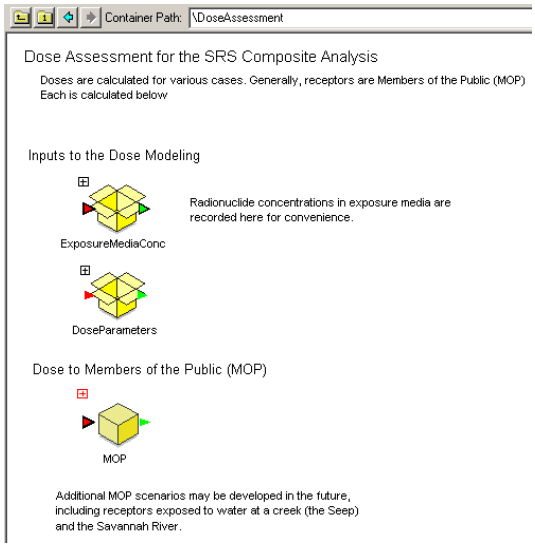


Figure 7-5. Top Level of the Dose Calculation *DoseAssessment*

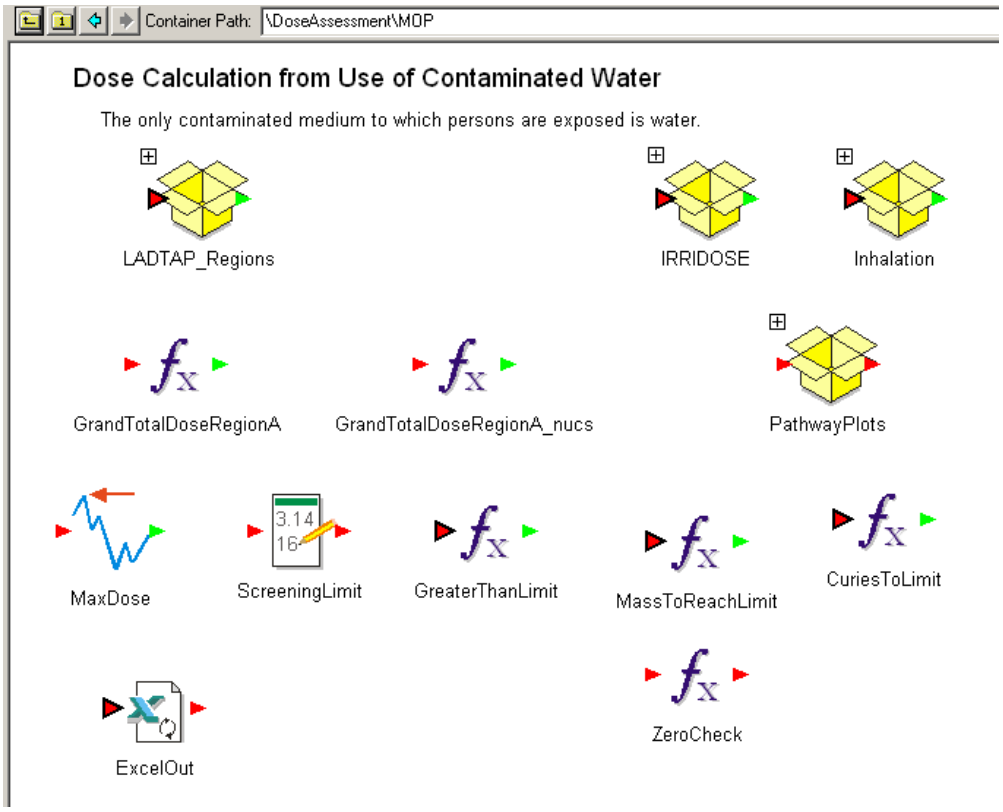


Figure 7-6. *MOP* Container

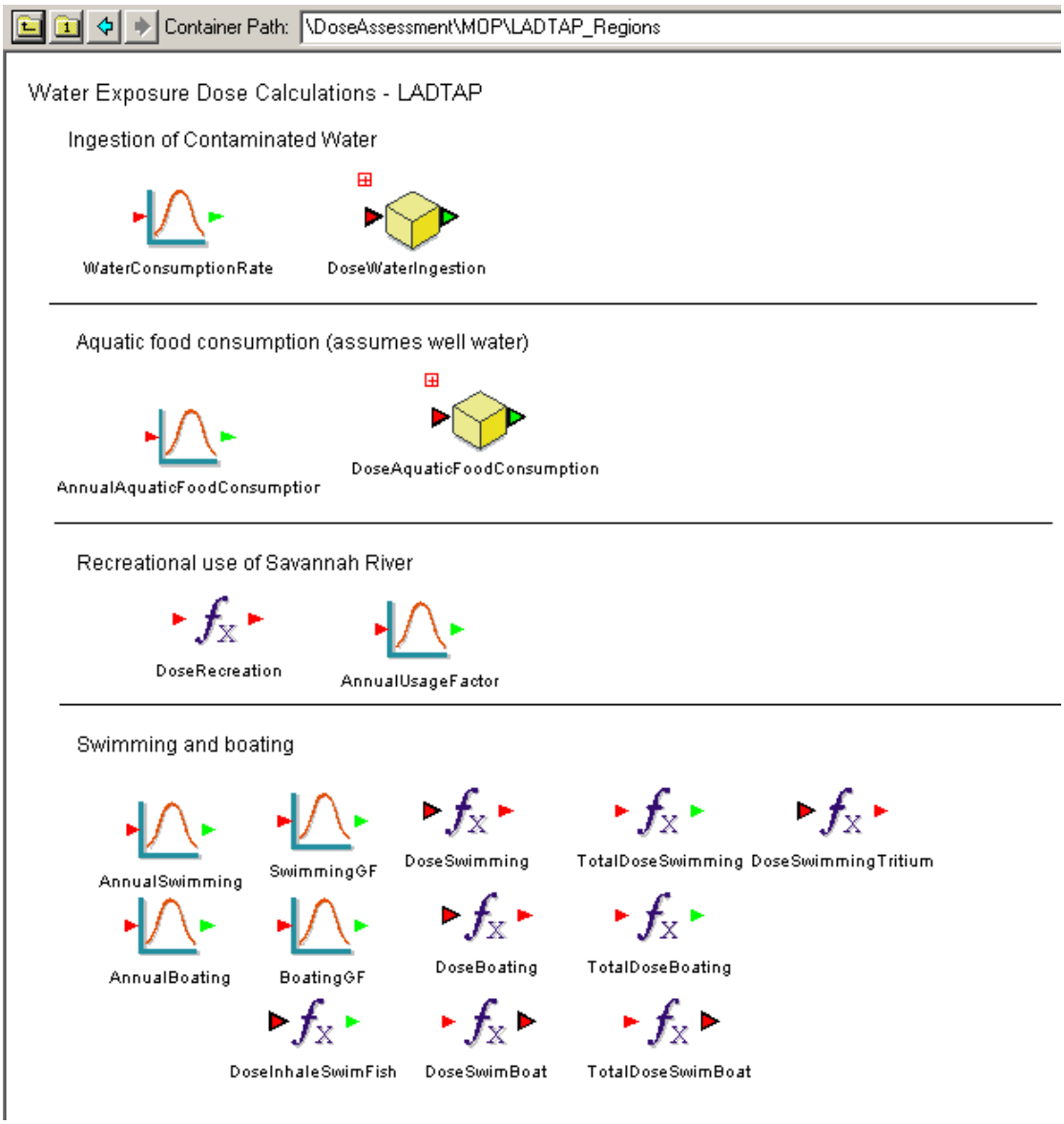


Figure 7-7. ***LADTAP_Regions*** **Container**

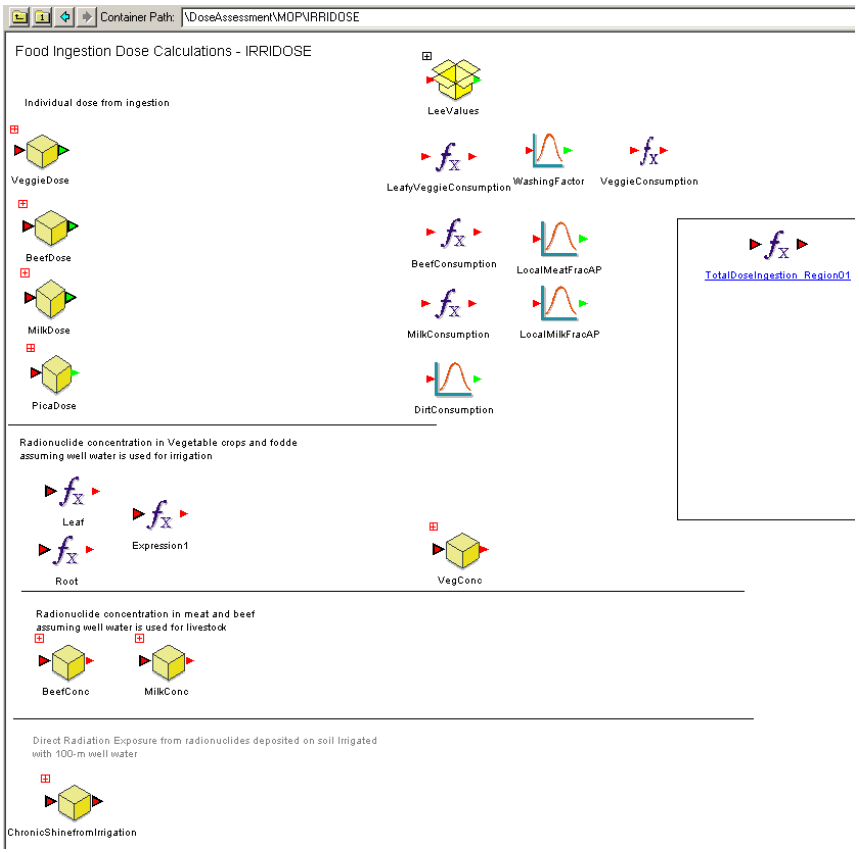


Figure 7-8. ***IRRIDOSE*** **Container**

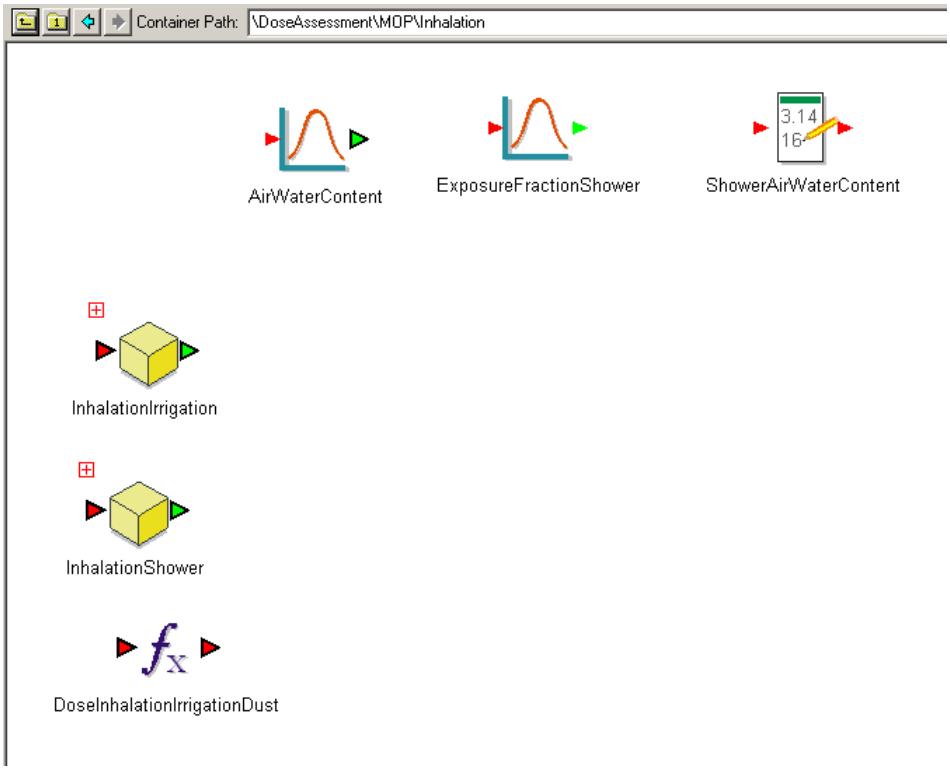


Figure 7-9. ***Inhalation*** **Container**

The physical exposure parameters (e.g., irrigation rates, weathering constants, crop yields, etc.), consumption and human exposure parameters (e.g., water ingestion rate), and bioaccumulation factors used in evaluating dose by the pathways addressed in the screening model are documented in Appendix A of Taylor et al. (2008) and discussed in sections 7.2.2.6.3, 7.2.2.6.4and 7.2.2.6.5 respectively.

7.2.2.6.3 Physical Exposure Parameters

Physical parameters required for evaluating the all-pathways screening dose for the CA are time periods that soil (buildup time), crops and pasture are exposed to irrigation; vegetable crop yields; local production fractions; areal density of soil; rate of irrigation for a garden; fraction of the year that crops are irrigated; weathering removal rate constant for crops, fractional retention of deposition on leaves; and the washing factor. Many of these parameters are generic in nature and are based on industry recommendations.

A summary of productivity and physical parameters used, along with values and distributions, are listed in Table 7-5. These values are consistent with those provided in Lee and Coffield (2008). The distribution type assumed in the GoldSim™ application is given in the table. A brief description of the basis for each parameter is provided below, summarized from Lee and Coffield (2008).

7.2.2.6.3.1 Exposure Time to Irrigation

For soil exposure time period to irrigation (buildup), Jannik and Dixon (2006) recommend 40 yrs to indicate the life time of a facility releasing radionuclides and ½ of a MEI lifetime assuming the MEI is exposed at that location for their lifetime. For the intruder and public scenario, it is assumed that the irrigation and harvesting of vegetables occur during the first year of residence, yielding the 183 d updated value. An updated maximum value of 365 d is recommended here assuming the entire MEI lifetime. However, if the exposure continues throughout the period of residence, a lifetime exposure of 30 yrs would yield a 5458 d maximum.

Exposure time to irrigation represents the period of time crop, leafy vegetable or pasture grass is exposed during the growing season. NRC (1977) recommends exposure times of 720 hours (30 days) for pasture grass and 1440 hours (60 days) for crop/vegetation. Simpkins and Hamby (2003) recommend a normal distribution for pasture and crop exposure times with a mean of 70 d and standard deviation of 7 d.

7.2.2.6.3.2 Crop Yields

Hamby (1991a) surveyed 21 county extension agents in Georgia and South Carolina to estimate the average mass, in kilograms, of vegetation harvested in a typical square meter of garden or farmland within a 50-mile radius of SRS. Crop yields in kg/m^2 were estimated for leafy vegetation (cabbage, lettuce, and spinach) and other aboveground vegetables (broccoli, cauliflower, green peas, lima beans, and sweet corn). Average agricultural, garden, and pasture grass productivity for farms in the 50-mile region is estimated to be 0.7 kg/m^2, 0.2 kg/m^2, and 1.8 kg/m^2, respectively. Because the garden productivity was estimated to be an order of magnitude lower than the NRC default (NRC 1977), Hamby (1991a) assumed the garden productivity to be equal to agricultural productivity. This report recommends use of the site-specific value of 0.7 kg/m^2 as the expected value for garden productivity, and the 0.2 kg/m^2 should be considered in the uncertainty range. Simpkins and Hamby (2003) recommend a lognormal distribution for crop yields with a geometric mean of 0.6 and geometric standard deviation of 1.4.

7.2.2.6.3.3 Fraction of Local Foodstuff

The current assumption of the fractions of vegetables, milk, and meat intake that is from a local garden were based on NRC (1977), professional judgment, and Yu et al. (2001) which considers the 0.5 fraction of vegetable intake to be a maximum value. Table 13-71 of the EPA Exposure Factors Handbook (EPA 1997) provides regional values for vegetables, milk, and meat intake fractions and scenario-specific values. This report recommends use of the values provided in EPA (1997) for households with gardens who raise animals for an all pathways analysis and those for households who farm for an intruder analysis.

7.2.2.6.3.4 Areal Density of Soil

The areal soil density is estimated by applying the assumed 0.15 m garden depth to the 1600 kg/m3 soil density yielding a 240 kg/m^2 areal density. Simpkins and Hamby (2003) recommend a normal distribution with a standard deviation of 17 kg/m^2.

7.2.2.6.3.5 Soil Density

The soil density of 1,600 kg/m^3 is based on a report by Hamby (1993a).

7.2.2.6.3.6 Atmospheric Mass Loading of Soil while Working in the Garden

The atmospheric mass loading of soil (kg/m^3) is based on an estimate provided in Revision 0 of the E-Area PA (McDowell-Boyer et al. 2000). The range in values is based on those provided in the RESRAD code (Yu et al., 2001).

7.2.2.6.3.7 Depth of Garden

The garden depth of 15 cm is based on a report by Hamby (1993a).

7.2.2.6.3.8 Garden Irrigation Rate

The rate of garden irrigation is based on an assumption of 1 in/wk (0.36 cm/d) (Lee and Coffield 2008, Hamby 1993b). For a 1 m^2 area, this is equivalent to 3.6 L/d/m^2. The maximum value assumes 0.5 cm/d irrigation

7.2.2.6.3.9 Fractional Crop Irrigation

The fraction of the year that crops are irrigated is assumed to be 0.2 (Lee and Coffield 2008). The maximum value is assumed to be 0.25.

7.2.2.6.3.10 Crop Weathering Constant

The weathering constant for crops contaminated directly via irrigation water is based on a 14-d half life for removal of activity from plants (Lee and Coffield 2008), which is consistent with NRC's (1977) reported value of 0.0021 hr^{-1}.

7.2.2.6.3.11 Fractional Retention on Leaves

The fractional retention of radionuclides deposited on leaves from irrigation is assumed to be consistent with the value NRC (1977) gives for all particulate radionuclides other than iodine. NRC (1977) gives a value of 1 for iodine. Therefore, the recommended maximum value of 0.25 should be increased to 1 for iodine.

7.2.2.6.3.12 Washing Factor

The washing factor applies to leafy crops only. It represents the fraction of superficially deposited radionuclides that are *not* washed off in preparation before ingestion. It is conservatively assumed to be unity here.

7.2.2.6.3.13 Water Content of Air

The water content of air in the garden during irrigation is assumed to be 10 g/m^3. This value corresponds to the water content associated with the 30-year annual average temperature (64.4º F) and relative humidity (68.7 %) for the SRS region (Kabela and Hunter 2007). In the shower, the water content of air is assumed to be 41 g/m^3 (Rittmann 2004).

Table 7-5. Crop Exposure Times and Productivity, and Physical Exposure Parameters for GoldSim™ All-Pathways Screening in CA (from Lee and Coffield, 2008 unless otherwise noted)

Parameter	Value	Minimum	Maximum	Distribution Applied
Soil exposure time period to irrigation (d) (Buildup time in soil)	183	60	365	Triangular
Pasture exposure time to irrigation (d)	30	30	90	Normal (SD = 7)
Vegetable crop exposure times to irrigation (d)	70	60	90	Normal (SD = 7)
Vegetable crop yield (kg/m^2)	0.7	0.2	4	Lognormal (Geom SD = 1.4)
Fraction of Foodstuff Produced Locally:				
Vegetables	0.173	0	0.5	Triangular
Meat	0.306	0	0.5	Triangular
Milk	0.207	0	0.5	Triangular
Areal density of soil (kg/m^2)	240	180	270	Normal (SD = 17)
Soil Density (kg/m^3)	1,600	1,350	1,600	Triangular
Atmospheric mass loading of soil while working in garden (k/m^3)	1.00E-07	1.00E-08	3.00E-07	Triangular
Depth of garden (cm)	15	15	61	Triangular
Garden irrigation rate (L/d/m^2)	3.6	2.08	5.5	Triangular
Fraction of the year that crops are irrigated	0.2	0.2	0.25	Triangular
Crop weathering constant (/d)	0.0495	0.03	0.0495	Triangular
Fractional retention of deposition on leaves	0.25	0.2	0.25	Triangular
Washing factor	1[a]		1	Discrete
Water content of air in garden (g/m^3)	10[b]			Discrete
Water content of air in shower (g/m^3)	41[c]			Discrete

[a] Conservatively assumed to be one (i.e., no effect of washing);[b] Kabela and Hunter (2007);
[c] Rittmann (2004)

7.2.2.6.4 Consumption and Human Exposure Parameters

Consumption rates currently used at SRS and updated parameter values required for evaluating dose for the various exposure pathways are listed in Table 7-6. Site-specific exposure parameters are provided when available.

Recommendations here attempt to advise the use of expected or average rates. Values evaluated from other sources and used at other DOE facilities are summarized below and provided in more detail in Lee and Coffield (2008). Where available, probability distributions and associated parameters taken from the literature are provided in the text.

With the exception of the last seven parameters in Table 7-6, the values and ranges reported are consistent with Lee and Coffield (2008). The source of the last seven parameters is Jannik and Dixon (2006).

7.2.2.6.4.1 Inhalation Rate

The inhalation rate in NRC (1977) is 8000 m^3/yr and is derived from data provided in ICRP-23 (ICRP 1974). According to EPA (1997), the mean adult female and male breathing rate for long-term exposures are 4113 m^3/yr (11.3 m^3/d) and 5548 m^3/yr (15.2 m^3/d), respectively. A value of 5548 m^3/yr inhalation rate is recommended for the screening analysis.

7.2.2.6.4.2 Water Ingestion

Ingestion of water is a key usage factor for the all-pathway analysis. The rate of contaminated water consumption can vary depending on assumed access to the water supply. The assumption could be made that total water intake comes from the community water supply. However, in the absence of site- and/or regional- specific surveys, national estimates are appropriate.

The RESRAD (Yu et al. 2001) 511 L/yr (1.4 L/d) average water ingestion rate updated for use in the all pathway analysis is based on EPA surveys published in the early 1990s. The 730 L/yr (2 L/d) water ingestion rate is taken from Hamby (1992) and is based on NRC (1977) rates for the MEI. The average rate for ingestion of drinking water listed in those sources is 370 L/yr (1 L/d). These publications consider indirect ingestion of water but do not consider whether or not the water is bottled or comes from a community or commercial source.

EPA (2004) estimates per capita ingestion of water using data from the combined 1994, 1995, 1996, and 1998 Continuing Survey of Food Intakes by Individuals (CSFII), conducted by the United States Department of Agriculture (USDA). This publication considers indirect ingestion of water from food with water added at the final phase of food preparation and reports water consumption from community water, bottled water, water from other sources, missing source, and total water. Summary data found in the Executive Summary (pages vii-viii) provide a 337 L/yr water ingestion rate.

7.2.2.6.4.3 Soil Ingestion

The 0.037 kg/yr (100 mg/d) ingestion rate is the value recommended for a child in EPA (1997). This document recommends 50 mg/d (0.019 kg/yr) for an adult in industrial settings but 100 mg/d for agricultural scenarios.

7.2.2.6.4.4 Foodstuff Consumption

Vegetable, milk, and beef consumption rates are taken from Hamby (1992). These values are based on county specific statistics provided by the counties within the states of South Carolina and Georgia that fall within a 50-mile radius of SRS. This report recommends continued use of average values from Hamby (1992) for screening analysis.

As a comparison, Simpkins and Hamby (2003) recommend a lognormal distribution for food consumption rates with a 19 kg/yr geometric mean and 2 kg/yr geometric standard deviation for leafy vegetables consumption; a 150 kg/yr geometric mean and 2 kg/yr geometric standard deviation for consumption of other vegetables; a 90 kg/yr geometric mean and 1.8 kg/yr geometric standard deviation for beef consumption; and a 140 kg/yr geometric mean and 2.2 kg/yr geometric standard deviation for consumption of milk.

7.2.2.6.4.5 Beef and Milk Cow Consumption Rates

Beef and milk cow annual consumption of fodder and water are taken from Hamby (1991a and 1992). Hamby (1993b) recommends a standard deviation of 8 kg/d and 11 kg/d for the fodder beef and milk cow consumption, respectively. Water ingestion rates for beef and milk cows come from Hamby (1993b).

7.2.2.6.4.6 Exposure Times

Recreational usage exposure times are taken from Hamby (1991a and 1992). According to Hamby (1991a and 1992), NRC recommends increasing averages by 50% to get maximum values. For the inhalation exposure pathways, Oztunali et al. (1981) determined that the inadvertent intruder person spends 1% of the year working in a garden (~100 h/yr). This report maintains the 0.01 time spent working in garden for the screening analysis. A 10 min/d showering exposure time was taken from EPA (1997).

The fraction of the year that cattle spend in pasture is based on a site-specific survey documented in Hamby (1991a). Simpkins and Hamby (2003) recommend a uniform distribution for cattle exposure fractions with a range from 75 to 100 percent.

7.2.2.6.4.7 Food Transport Times

The recommended amount of time that it takes for transport of vegetables from harvest to the consumer is taken from Yu et al. (2001). Simpkins and Hamby (1993) recommend a normal distribution for the vegetable transport time with a mean of 1 d (leafy) and 6 d (other vegetables) and 0.1 d (leafy) and 0.6 d (other vegetables) standard deviations. Milk and beef transport times are taken from Hamby (1991a). Simpkins and Hamby (1993) recommends lognormal distribution for milk and beef with 3 d mean and 1.5 d geometric standard deviation for milk and 6 d mean and 1.4 d geometric standard deviation for beef.

7.2.2.6.4.8 Other Exposure Parameters

References supporting the remaining exposure parameters listed in Table 7-6 are cited in Jannik and Dixon (2006). The shoreline width factor is the default value from NRC (1977). The elapsed time before consumption of water is a scenario- or site-specific parameter, which can be adjusted to account for holdup prior to consumption. It is conservatively assumed to be zero for the screening analysis. Hamby (1991b) is cited for the swimming and boating geometry factors, representing 4π and 2π geometry, respectively. As such, shielding provided by the boat's hull is not considered. Hamby (1991b) is also cited for the exposure time of the sediment or soil to contaminated water. The water-to-sediment transfer coefficient came from the LADTAP II documentation (Simson and McGill, 1980). Hamby (1991b) is again cited for the water absorption rate, which allows accounting for the tritium dose via skin absorption.

Table 7-6. Recommended Exposure Times and Consumption Rates (from Lee and Coffield, 2008) used in Screening Analysis

Parameter	Value	Minimum	Maximum	Distribution Applied
Inhalation rate (m^3/y)	5,548	1,267	11,600	Normal[a]
Consumption Rate				
Water (L/y)	337	184	730	Triangular[a]
Soil (kg/yr)	0.0365 [a]	0.0008	0.05	Triangular[a]
Leafy vegetable (kg/y)	21	18	43	Lognormal[a] (Geom SD = 2)
Other vegetable (kg/y)	163	90	276	Lognormal[a] (Geom SD = 2)
Meat (kg/y)	43	26	81	Lognormal[a] (Geom SD = 1.8)
Finfish (kg/y)	9 [b]	2.2	19	Triangular[a]
Seafood (kg/y)	0 [c]	0	5	Triangular[a]
Milk (L/yr)	120	73.7	230	Lognormal[a] (Geom SD = 2.2)
Fodder - Beef cattle (kg/d)	36	27	50	Normal[a] (SD = 8)
Fodder - Milk cattle (kg/d)	52	36	55	Normal[a] (SD = 11)
Fraction of milk-cow intake from pasture	0.56	0.5	1	Triangular[a]
Fraction of beef-cow intake from pasture	0.75	0.5	1	Triangular[a]
Water (beef cow) (L/d)	28	28	50	Triangular[a]
Water (milk cow) (L/d)	50	50	60	Triangular[a]
Exposure Time				
Shoreline exposure (hr/yr)	23	11	35	Triangular[a]
Swimming exposure (hr/yr)	8.9	8.9	13	Triangular[a]
Boating exposure (hr/yr)	21	9.1	31.5	Triangular[a]
Exposure Fraction shower	10 min/d	10 min/d	30 min/d	Triangular[a]
Fraction of year working in garden	0.01	0.01	0.08	Triangular[a]
Fraction of year cattle on pasture	1	0.75	1	Uniform[a]
Transport Time and Other Exposure Parameters				
Vegetable transport time (d)	6	6	14	Normal[a] (SD = 1)
Feed-milk-man transport time (d)	3	1	4	Lognormal[a] (Geom SD = 1.5)
Time from slaughter to consumption (d)	6	6	20	Lognormal[a] (Geom SD = 1.4)
Shoreline width factor	0.3[b]			Discrete
Elapsed time before consumption of water (d)	0[b]			Discrete
Swimming geometry factor	1[b]			Discrete
Boating geometry factor	0.5[b]			Discrete
Exposure time soil to water (yr)	40[b]			Discrete
Transfer coefficient water to sediment(L/m^2-d)	100[b]			Discrete
Water absorption (ml/hr)	35[b]			Discrete

[a] Distribution shape taken from WSRC (2008c); [b] From Jannik and Dixon (2006); [c]Value of 2 should be used if the dose is to a person downriver (i.e., Savannah area)

7.2.2.6.5 Bioaccumulation Factors

Bioaccumulation factors considered in the CA screening model include soil-to-vegetable (also known as soil-to-plant ratios, plant-to-soil ratios), feed-to-milk, feed-to-beef and water-to-fish transfer factors. Soil-to-vegetable transfer factors determine the fraction of an element that is drawn from the soil into the edible plant. Feed-to-milk transfer factors represent the nuclide fraction transferred from fodder to milk. Feed-to-meat transfer factors represent the nuclide-specific fraction transferred from fodder to beef. Water-to-fish transfer factors are the equilibrium ratios between concentration in aquatic foods and concentration in water.

The values of the bioaccumulation factors used in this analysis were taken from Lee and Coffield (2008). The values from the Lee and Coffield (2008) report were based on a recent comprehensive literature review, and thus values are updated to include the latest available information. In general, the values from more recent compilations are recommended, rather than those in older publications. Site-specific values are recommended when available, but the most recent compilation by Staven et al. (2003) is considered the preferred secondary source of values in the absence of site-specific values.

7.2.2.6.5.1 Soil-to-Vegetable Transfer Factors

The soil-to-vegetable transfer factors used in the screening analysis are given in Table 7-7. With the exception of NCRP (1996), soil-to-vegetable transfer factors listed in the references cited in Lee and Coffield (2008) are based on dry weight. These factors are converted for fresh vegetable consumption using a site-specific regional conversion factor (Lee and Coffield 2008).

7.2.2.6.5.2 Feed-to-Milk Transfer Factors

Feed-to-milk transfer factors represent the fraction of daily elemental intake in feed (Bq/d) transferred to a liter of milk (Bq/L). The values used in the screening analysis for the CA are listed in Table 7-7 and are largely taken from Lee and Coffield (2008), which documents the literature sources of these values. Feed-to-milk transfer factors for Ar, Fm, Kr, Md, Ne, O, and Xe are not given by Lee and Coffield (2008). Factors provided for these elements in Table 7-7 are taken from NCRP (1996).

7.2.2.6.5.3 Feed-to-Meat Transfer Factors

Feed-to-meat transfer factors represent the fraction of daily elemental intake in feed (Bq/d) which is transferred to a kg of meat (Bq/kg). The values used in the screening analysis for the CA are listed in Table 7-7 and are largely taken from Lee and Coffield (2008), which documents the literature sources of these values. Feed-to-meat transfer factors for Ar, He, Kr, Md, Ne, O, Rf, and Xe are not given by Lee and Coffield (2008). Factors provided for these elements in Table 7-7 are taken from NCRP (1996).

7.2.2.6.5.4 Water-to-Fish Transfer Factors

Element-specific water-to-fish bioaccumulation factors represent the fraction of contaminant concentration in water (Bq/L) that would be transferred to fish muscle (Bq/kg). SRS values assume freshwater fish. The values used in the screening analysis for the CA are listed in Table 7-7 and are largely taken from Lee and Coffield (2008), which documents the literature sources of these values. Water-to-fish transfer factors for Ar, B, Fm, Ha, Kr, Li, Lr, Md, Ne, No, Rf, Xe, and Yb are not given by Lee and Coffield (2008). Factors provided for these elements in Table 7-7 are taken from NCRP (1996).

Table 7-7. Elemental Soil-to-Vegetable Transfer Factors used in CA Screening Analysis (from Lee and Coffield 2008)

Z	Element	Soil-to-Vegetable (kg/kg)[a]	Feed-to-Milk (d/L)[b]	Feed-to-Beef (d/kg)[c]	Water-to-Fish (L/kg)[d]
89	Ac	6.83E-05	2.00E-05	4.00E-04	2.50E+01
47	Ag	1.18E-02	1.58E-03	3.00E-03	5.00E+00
13	Al	1.27E-04	2.06E-04	1.50E-03	5.00E+02
95	Am	6.83E-05	1.50E-06	4.00E-05	3.00E+01
18	Ar	0	0[e]	0[e]	0[e]
33	As	1.17E-03	6.00E-05	2.00E-03	1.70E+03
85	At	2.93E-02	1.03E-02	1.00E-02	1.50E+01
79	Au	3.51E-03	5.50E-06	5.00E-03	3.30E+01
56	Ba	2.93E-03	4.80E-04	2.00E-04	4.00E+00
4	Be	2.93E-04	9.00E-07	1.00E-03	1.00E+02
83	Bi	9.75E-02	5.00E-04	4.00E-04	1.50E+01
97	Bk	1.00E-03	2.00E-06	2.50E-05	2.50E+01
35	Br	2.93E-01	2.00E-02	2.50E-02	4.00E+02
6	C	1.37E-01	1.20E-02	3.10E-02	5.00E+04
20	Ca	6.83E-02	3.00E-03	2.00E-03	4.00E+01
48	Cd	2.93E-02	1.00E-03	4.00E-04	2.00E+02
58	Ce	3.90E-03	3.00E-05	2.00E-05	3.00E+01
98	Cf	6.83E-05	1.50E-06	4.00E-05	2.50E+01
17	Cl	1.37E+01	1.70E-02	2.00E-02	5.00E+01
96	Cm	8.39E-05	2.00E-05	4.00E-05	3.00E+01
27	Co	1.31E-02	3.00E-04	1.00E-02	3.00E+02
24	Cr	8.78E-04	1.00E-05	9.00E-03	4.00E+00
55	Cs	9.00E-01	7.90E-03	5.00E-02	3.00E+03
29	Cu	4.88E-02	2.00E-03	9.00E-03	2.00E+02
66	Dy	3.90E-03	3.00E-05	2.00E-05	3.00E+01
68	Er	3.90E-03	3.00E-05	2.00E-05	3.00E+01
99	Es	1.00E-03	2.00E-06	2.50E-05	2.50E+01
63	Eu	3.90E-03	3.00E-05	2.00E-05	3.00E+01
9	F	1.17E-03	1.00E-03	1.50E-01	1.00E+01
26	Fe	9.75E-03	3.00E-05	2.00E-02	2.00E+02
100	Fm	2.00E-03	8.00E-06[e]	2.00E-04	1.00E+01[e]
87	Fr	5.85E-03	2.06E-02	2.50E-03	3.00E+01
31	Ga	7.80E-05	5.00E-05	5.00E-04	4.00E+02
64	Gd	3.90E-03	3.00E-05	2.00E-05	3.00E+01
32	Ge	1.56E-02	7.21E-02	7.00E-01	4.00E+03
1	H	4.80E+00	1.50E-02	1.2E-02	1.00E+00
72	Hf	1.95E-04	5.50E-07	3.16E-05	3.00E+02

Table 7-7. Elemental Soil-to-Vegetable Transfer Factors used in CA Screening Analysis (from Lee and Coffield 2008) -- continued

Z	Element	Soil-to-Vegetable (kg/kg)[a]	Feed-to-Milk (d/L)[b]	Feed-to-Beef (d/kg)[c]	Water-to-Fish (L/kg)[d]
80	Hg	3.90E-02	4.70E-04	2.50E-01	1.00E+03
67	Ho	3.90E-03	3.00E-05	3.00E-04	3.00E+01
53	I	7.80E-03	9.00E-03	4.00E-02	4.00E+01
49	In	7.80E-05	2.00E-04	8.00E-03	1.00E+04
77	Ir	2.93E-03	2.00E-06	1.50E-03	1.00E+01
19	K	1.07E-01	7.20E-03	2.00E-02	1.00E+03
36	Kr	0	0[e]	0[e]	0[e]
57	La	6.83E-05	2.00E-05	2.00E-03	3.00E+01
71	Lu	7.80E-04	2.06E-05	4.50E-03	2.50E+01
101	Md	2.00E-03	5.00E-06	2.00E-04[e]	1.00E+01[e]
12	Mg	1.07E-01	3.90E-03	2.00E-02	5.00E+01
25	Mn	3.90E-02	3.00E-05	5.00E-04	4.00E+02
42	Mo	1.56E-01	1.70E-03	1.00E-03	1.00E+01
7	N	3.50E-01	2.50E-02	7.50E-02	2.00E+05
11	Na	5.85E-02	1.60E-02	8.00E-02	2.00E+01
41	Nb	4.88E-03	3.20E-05	2.90E-04	3.00E+02
60	Nd	3.90E-03	3.00E-05	2.00E-05	3.00E+01
10	Ne	0	0[e]	0[e]	0[e]
28	Ni	1.17E-02	1.60E-02	5.00E-03	1.00E+02
93	Np	2.54E-03	5.00E-06	1.00E-03	3.00E+01[e]
8	O	6.00E-01	2.00E-02[e]	2.00E-01[e]	1.00E+00
76	Os	6.83E-04	5.00E-03	4.00E-01	1.00E+03
15	P	6.83E-01	1.60E-02	5.00E-02	5.00E+04
91	Pa	4.18E-04	5.00E-06	4.47E-04	1.00E+01
82	Pb	1.17E-03	2.60E-04	4.00E-04	3.00E+02
46	Pd	7.80E-03	1.00E-02	4.00E-03	1.00E+01
61	Pm	3.90E-03	3.00E-05	2.00E-05	3.00E+01
84	Po	1.37E-03	3.40E-04	5.00E-03	5.00E+01
59	Pr	3.90E-03	3.00E-05	2.00E-05	3.00E+01
78	Pt	4.88E-03	5.15E-03	4.00E-03	3.50E+01
94	Pu	2.15E-04	1.10E-06	1.00E-05	3.00E+01
88	Ra	4.64E-03	1.30E-03	9.00E-04	5.00E+01
37	Rb	1.76E-01	1.20E-02	1.00E-02	2.00E+03
75	Re	1.29E+00	1.50E-03	8.00E-03	1.20E+02
45	Rh	7.80E-03	1.00E-02	2.00E-03	1.00E+01
86	Rn	0	0	0	0
44	Ru	7.80E-03	3.30E-06	5.00E-02	1.00E+02
16	S	2.93E-01	1.60E-02	2.00E-01	8.00E+02
51	Sb	2.49E-03	2.50E-05	1.00E-03	1.00E+02
21	Sc	1.95E-04	5.00E-06	1.50E-02	1.00E+02

Table 7-7. Elemental Soil-to-Vegetable Transfer Factors used in CA Screening Analysis (from Lee and Coffield 2008) -- continued

Z	Element	Soil-to-Vegetable (kg/kg)[a]	Feed-to-Milk (d/L)[b]	Feed-to-Beef (d/kg)[c]	Water-to-Fish (L/kg)[d]
34	Se	5.14E-02	4.00E-03	1.50E-02	1.70E+02
14	Si	1.37E-02	2.00E-05	4.00E-05	2.00E+01
62	Sm	3.90E-03	3.00E-05	3.16E-04	3.00E+01
50	Sn	1.17E-03	1.00E-03	8.00E-02	3.00E+03
38	Sr	9.75E-02	2.80E-03	8.00E-03	6.00E+01
73	Ta	4.88E-03	4.10E-07	1.34E-05	3.00E+02
65	Tb	3.90E-03	3.00E-05	2.00E-05	3.00E+01
43	Tc	4.68E-02	1.87E-03	6.32E-03	2.00E+01
52	Te	1.20E-02	4.50E-04	7.00E-03	4.00E+02
90	Th	6.44E-05	5.00E-06	4.00E-05	1.00E+02
22	Ti	5.85E-04	7.53E-05	1.73E-04	1.00E+03
81	Tl	7.80E-05	2.00E-03	4.00E-02	1.00E+04
69	Tm	7.80E-04	2.06E-05	4.50E-03	2.50E+01[e]
92	U	2.34E-03	4.00E-04	3.00E-04	1.00E+01
23	V	5.85E-04	2.06E-05	2.50E-03	2.00E+02
74	W	5.00E-02	3.00E-04	4.00E-02	1.00E+01
54	Xe	0	0[e]	0[e]	0[e]
39	Y	1.95E-03	2.00E-05	1.00E-03	3.00E+01
70	Yb	7.80E-04	2.06E-05	4.00E-03	2.00E+02[e]
30	Zn	6.83E-02	1.00E-02	1.00E-01	3.50E+02
40	Zr	1.95E-04	5.50E-07	1.84E-04	3.00E+02

[a] Corresponding to mg/kg fresh weight vegetable per mg/kg soil (mg representing elemental quantity)

[b] Corresponding to mg/L milk per mg/d intake by dairy cow (mg representing elemental quantity)

[c] Corresponding to mg/kg meat per mg/d intake by beef cattle (mg representing elemental quantity)

[d] Corresponding to mg/kg fish tissue per mg/L water (mg representing elemental quantity)

[e] From NCRP (1996)

7.2.2.6.6 Dose Factors

7.2.2.6.6.1 Internal DCFs

Previous SRS PAs utilized the internal DCFs from Environmental Protection Agency (EPA) Federal Guidance Report 11, published in 1988 (EPA 1988). The International Commission on Radiological Protection (ICRP) published new internal DCFs based upon updated dosimetric models in ICRP Publication 72 in 1995 (ICRP 1995). The DOE has begun using the ICRP models for occupational exposure internal dose assessments at different sites including SRS and they are used for SRS safety basis calculations. Because the ICRP data is the most recent data available and is based on the most recent dosimetric models, SRS is proposing to use the ICRP Publication 72 DCFs for PA analyses (Lee and Coffield 2008), and thus they are used in the CA screening analysis. For DCFs not provided in ICRP Publication 72, internal DCFs (i.e., effective dose factors) from NCRP Report No. 123 were used (NCRP 1996).

The internal DCFs are reported in Table 7-1 of Appendix A in Taylor et al. 2008 for the 826 radionuclides considered in the CA screening analysis, in units of Sv/Bq. For ingestion of H3, the effective dose factor assuming all H3 is in the form of HTO is assumed, rather than organically-bound, as this is the more likely form of the majority of H3 in water. For inhalation DCFs, the most conservative absorption type from ICRP-72 was assumed, except for H3. For inhalation of H3, the effective dose factor assumed is higher than that for HTO vapor, and corresponds to a clearance rate of M (representing a "medium" clearance rate in ICRP 72). This is consistent with the value chosen for the Saltstone Disposal Facility and F-Area Tank Farm PAs (WSRC 2008b, 2008c).

7.2.2.6.6.2 External DCFs

External DCFs selected to represent dose from contaminated soil to a depth of 15 cm are taken from EPA Federal Guidance Report 12 (EPA 1993). The external DCFs represent the dose rate per unit of activity of soil contaminated to 15 cm, reported in Sv/s per Bq/m^3 of soil. External dose from radionuclides deposited on the ground surface (without mixing into the soil below) is represented by Federal Guidance Report 12 DCFs reported in units of Sv/s per Bq/m^2 of ground surface, reported as "Shore Shine Factors" in Table 7-2 of Appendix A in Taylor et al. 2008. External dose resulting from immersion in water is calculated using Federal Guidance Report 12 (EPA 1993) DCFs reported in units of Sv/s per Bq/m^3 of water. Values for these three types of external DCFs are given in Table 7-2 of Appendix A in Taylor et al. 2008.

7.2.2.6.7 Screening Dose Criterion

The performance measures for the CA are a dose limit of 100 mrem/yr and a dose constraint of 30 mrem/yr (DOE 1999). Consistent with the radionuclide screening approach for the ELLWF PA (WSRC 2008b), the radionuclide screening dose criterion for the CA is based on the 30 mrem/yr dose constraint. The radionuclide screening dose criterion chosen is one percent of the dose constraint (i.e., 0.3 mrem/yr) and is consistent with that used in the ELLWF PA.

7.2.3 Radionuclide Screening Results

The output from the GoldSim™ model, which contains the data for all 826 radionuclides, is written to an Excel workbook. That workbook runs a macro which parses the data, selects those data that meet the screening criterion (i.e., those radionuclides for which the dose exceeds the screening criterion of 0.3 mrem/yr), and writes the selected data to a summary sheet. Table 7-8 shows an example of the summary sheet. The data at the beginning of the sheet are used to differentiate runs. "Unsat Thickness" is the thickness of the unsaturated zone (i.e., the depth to groundwater). "Sat Length" is the distance from the inlet of the saturated zone to the point of discharge to the surface stream. "Curies" is the initial inventory in curies for each radionuclide. "Number of nuclides not screened out" identifies how many of the 826 radionuclides remain after a given set of conditions is run. The column "mrem/yr" is the dose at the stream mouth for a given set of conditions. All of these data are transmitted from the GoldSim™ model.

Radionuclide screening results are presented in Appendix B of Taylor et al. (2008) and are summarized in Table 7-9. Run 1 is considered the most representative set of conditions (i.e., Unsat Thickness=10 feet, Sat Length=762 m, and Stream Flow=10 ft^3/s) and will be used to determine which radionuclides should be analyzed in the CA. Run 1 results in 77 radionuclides not being screened out. The other cases were run to assess the impact of changing the screening conditions. Increasing the stream flow to 50 ft^3/s (Run 2) decreases the number of radionuclides not screened out to 72 (i.e., only five radionuclides less than Run 1). Decreasing the saturated zone flow length to 100 meters (Run 3) increases the number of radionuclides not screened out to 112 (only 35 radionuclides more than Run 1 for an extremely conservative saturated zone flow length). Increasing the unsaturated zone thickness to 30 feet (Run 4) decreases the number of radionuclides not screened out to 66 (only eleven radionuclides less than Run 1).

Table 7-8. Sample Output

Screening limit = 0.3 mrem/yr		
Unsat Thickness = 10 ft		
Sat Length = 762 m		
Stream Length = 1,000 m		
Stream Flow = 10 ft3/s		
Curies =	1.00E+07	
Nuclide	mrem/yr	Number of nuclides not screened
Ag108m	4.31E-01	7.70E+01
Al26	2.63E+06	
Am241	2.21E+02	
Am242m	2.22E+01	
Am243	8.78E+01	
Be10	5.95E+01	
Bi210m	1.41E+04	
Bk247	5.03E+00	
C14	3.92E+07	
Ca41	1.22E+05	
Cd113	1.21E+05	
Cf249	8.41E+00	
Cl36	5.18E+06	
Cm243	3.92E+00	
Cm244	4.22E+00	
Cm245	2.09E+02	
Cm246	6.51E+00	
Cm247	6.98E+00	
Cs135	1.01E+05	
Es254	3.28E+01	
Fe60	2.37E+05	
Ge68	3.56E+01	
H3	1.21E+04	
Hf172	1.59E+03	
Hf182	2.26E+06	
Hg194	3.01E+05	
I129	1.51E+07	
In115	6.29E+01	
K40	2.82E+05	
Lu173	1.29E+01	
Lu174	8.32E+02	
Lu174m	1.11E+02	
Lu176	2.19E+05	
Md258	7.55E+00	
Mn53	2.12E+02	
Mo93	5.88E+04	
Mo93m	1.28E+00	
Nb93m	2.62E+04	
Nb94	8.35E+05	
Ni59	1.08E+03	
Ni63	1.04E+02	

Table 7-8. Sample Output (Continued)

Nuclide	mrem/yr	Number of nuclides not screened
Np236a	2.25E+06	
Np237	1.43E+07	
Pa231	5.59E+07	
Pd107	3.91E+01	
Pt193	2.99E+03	
Pt193m	7.10E-01	
Pu237	8.24E-01	
Pu238	1.71E+01	
Pu239	3.36E+03	
Pu240	1.56E+03	
Pu241	7.04E+00	
Pu242	4.11E+03	
Pu244	7.11E+03	
Ra226	4.03E+06	
Rb87	4.79E+04	
Re186m	1.87E+06	
Re187	3.48E+03	
Si32	4.81E+06	
Sr90	4.28E+02	
Ta179	7.08E+01	
Ta180m	2.61E+04	
Tc97	8.49E+03	
Tc98	2.47E+05	
Tc99	7.87E+04	
Th230	4.43E+05	
Th231	4.99E+00	
Th232	9.12E-01	
Ti44	6.70E+06	
Tm171	3.68E+01	
U231	1.97E+01	
U233	1.35E+04	
U234	4.84E+04	
U235	6.79E+04	
U236	7.66E+03	
U238	8.03E+03	
Zr93	3.49E+04	

Table 7-9. Results of Screening Analysis Runs

Run	1	2	3	4
Unsaturated Zone Thickness (ft)	10	10	10	30
Saturated Zone Length (m)	762	762	100	762
Stream Flow (ft^3/s)	10	50	10	10
Curies per Radionuclide	1.0E+07	1.0E+07	1.0E+07	1.0E+07
Number of Radionuclides Not Screened out at 0.3 mrem/yr Screening Limit	77	72	112	66

Results of this CA screening are compared with results from the ELLWF and FTF PAs in Table 7-10. The first column of the table contains the list of radionuclides not screened out in this radionuclide screening from Run 1. The second and third columns show results from the ELLWF PA. The second column shows the 38 radionuclides that did not screen out and the third column shows the key (i.e., those for which disposal limits were developed) radionuclides for the all-pathways performance objective for the slit trench disposal units. The fourth and fifth columns show results from the FTF PA. The fourth column shows the 64 radionuclides selected for detailed analysis. The fifth column shows the key radionuclides (i.e., those with a peak individual water ingestion dose greater than 0.005 mrem/yr) in the FTF PA. Because the FTF PA credited the steel and concrete of the waste storage tank and the grout used to fill the tank, the ELLWF PA results for the slit trenches are more appropriate for comparison. The radionuclide screening results for Run 1 include all of the radionuclides considered to be important in the ELLWF PA except for Cf251, Cm248, Se79, and Sn126.

Table 7-10. CA Screening Compared with ELLWF and FTF PAs

CA Radionuclide Run 1	ELLWF Radionuclide [1]	ELLWF Key Radionuclide [2]	FTF PA Radionuclide [3]	FTF PA Key Radionuclide [4]
- -	- -	- -	Ac227	- -
Ag108m	- -	- -	- -	- -
Al26	- -	- -	Al26	- -
Am241	Am241	Am241	Am241	- -
Am242m	--	- -	Am242m	- -
Am243	Am243	Am243	Am243	- -
- -	- -	- -	Ba137m	- -
Be10	- -	- -	- -	- -
Bi210m	- -	- -	- -	- -
Bk247	- -	- -	- -	- -
- -	- -	- -	Bk249	- -
C14	C14	C14	C14	C14
Ca41	- -	- -	- -	- -
Cd113	- -	- -	- -	- -
- -	- -	- -	Ce144	- -
Cf249	Cf249	Cf249	Cf249	- -
- -	Cf251	Cf251	Cf251	- -
- -	- -	- -	Cf252	- -
Cl36	Cl36	Cl36	- -	- -
- -	- -	- -	Cm242	- -
Cm243	- -	- -	Cm243	- -
Cm244	Cm244	Cm244	Cm244	- -
Cm245	Cm245	Cm245	Cm245	- -
Cm246	Cm246	Cm246	- -	- -
Cm247	Cm247	Cm247	Cm247	- -
- -	Cm248	Cm248	Cm248	- -
- -	- -	- -	Co60	- -
- -	- -	- -	Cs134	- -
Cs135	- -	- -	Cs135	- -
- -	- -	- -	Cs137	- -
Es254	- -	- -	- -	- -
- -	- -	- -	Eu152	- -
- -	- -	- -	Eu154	- -
- -	- -	- -	Eu155	- -
Fe60	- -	- -	- -	- -
Ge68	- -	- -	- -	- -
H3	H3	H3	H3	- -
Hf172	- -	- -	- -	- -
Hf182	- -	- -	- -	- -
Hg194	- -	- -	- -	- -
I129	I129	I129	I129	- -
In115	- -	- -	- -	- -
K40	K40	K40	- -	- -
Lu173	- -	- -	- -	- -
Lu174	- -	- -	- -	- -

Table 7-10. CA Screening Compared with ELLWF and FTF PAs (Continued)

CA Radionuclide Run 1	ELLWF Radionuclide [1]	ELLWF Key Radionuclide [2]	FTF PA Radionuclide [3]	FTF PA Key Radionuclide [4]
Lu174m	- -	- -`	- -	- -
Lu176	- -	- -	- -	- -
Md258	- -	- -	- -	- -
Mn53	- -	- -	- -	- -
Mo93	Mo93	Mo93	- -	- -
Mo93m	- -	- -	- -	- -
- -	- -	- -	Na22	- -
Nb93m	- -	- -	- -	- -
Nb94	Nb94	Nb94	Nb94	- -
Ni59	Ni59	Ni59	Ni59	- -
Ni63	- -	- -	Ni63	- -
Np236a	- -	- -	- -	- -
Np237	Np237	Np237	Np237	Np237
Pa231	- -	- -	Pa231	Pa231
- -	- -	- -	- -	Pb210
Pd107	Pd107	Pd107	- -	- -
- -	- -	- -	Pm147	- -
- -	- -	- -	Pr144	- -
Pt193	- -	- -	- -	- -
Pt193m	- -	- -	- -	- -
Pu237	- -	- -	- -	- -
Pu238	Pu238	Pu238	Pu238	- -
Pu239	Pu239	Pu239	Pu239	Pu239
Pu240	Pu240	Pu240	Pu240	Pu240
Pu241	Pu241	Pu241	Pu241	- -
Pu242	Pu242	Pu242	Pu242	Pu242
Pu244	Pu244	Pu244	Pu244	- -
Ra226	Ra226	Ra226	Ra226	Ra226
	- -	- -	Ra228	- -
Rb87	- -	- -	- -	- -
Re186m	- -	- -	- -	- -
Re187	- -	- -	- -	- -
- -	- -	- -	Rh106	- -
- -	- -	- -	Ru106	- -
- -	- -	- -	Sb125	- -
- -	- -	- -	Sb126	- -
- -	- -	- -	Sb126m	- -
- -	Se79	Se79	Se79	- -
Si32	- -	- -	- -	- -
- -	- -	- -	Sm151	- -
- -	Sn126	- -	Sn126	- -
Sr90	Sr90	Sr90	Sr90	- -
Ta179	- -	- -	- -	- -
Ta180m	- -	- -	- -	- -
Tc97	- -	- -	- -	- -
Tc98	- -	- -	- -	- -
Tc99	Tc99	Tc99	Tc99	Tc99

Table 7-10. CA Screening Compared with ELLWF and FTF PAs (Continued)

CA Radionuclide Run 1	ELLWF Radionuclide [1]	ELLWF Key Radionuclide [2]	FTF PA Radionuclide [3]	FTF PA Key Radionuclide [4]
- -	- -	- -	Te125m	- -
- -	- -	- -	Th229	Th229
Th230	Th230	Th230	Th230	Th230
Th231	- -	- -	- -	- -
Th232	Th232	Th232	Th232	- -
Ti44	- -	- -	- -	- -
Tm171	- -	- -	- -	- -
U231	- -	- -	- -	- -
- -	- -	- -	U232	- -
U233	U233	U233	U233	U233
U234	U234	U234	U234	U234
U235	U235	U235	U235	- -
U236	U236	U236	U236	U236
U238	U238	U238	U238	U238
- -	- -	- -	Y90	- -
Zr93	Zr93	Zr93	- -	- -

[1] From Tables 4-3 and 4-4 of Part C of WSRC 2008b
[2] From Tables 7-10, 7-11, and 7-12 of WSRC 2008b
[3] From Table 4.2-5 of WSRC 2008a
[4] From Table 5.2-9 of WSRC 2008a

7.2.4 Secondary Screening

The list of 77 radionuclides derived from the CA radionuclide screening was reviewed to determine if some radionuclides should be removed from the list on the basis of unlikely production in SRS processes, etc. (Sigg and Lee 2008). This review concluded that 48 of the 77 radionuclides could be excluded from the CA due to a variety of considerations (e.g., no good production pathway). The results of this review are shown in Table 7-11 (Sigg and Lee 2008).

Table 7-11. CA Radionuclides versus Expected SRS Production

Nuclide	Delete			Fission Yield %	Comment
	Short Half-Life	Excessive Mass	Production Pathway		
Ag108m				2.28E-12	
Al26					Al27(n,2n) or Na23(alpha, n); Ref. 2.
Am241					
Am242m					
Am243					
Be10		X			Several metric tons of Be10 is unlikely
Bi210m					
Bk247			X		
C14					
Ca41					
Cd113		X		0.014	Excessive Mass

Table 7-11. CA Radionuclides versus Expected SRS Production (Continued)

Nuclide	Delete				
	Short Half-Life	Excessive Mass	Production Pathway	Fission Yield %	Comment
Cf249		X			Greatly exceeds tank inventories
Cl36			X		Long lived, Took measures to minimize Cl target
Cm243		X			Exceeds Cm production for Cf campaigns and tank inventories
Cm244		X			Exceeds Cm production for Cf campaigns and tank inventories
Cm245					
Cm246		X			Exceeds Cm production for Cf campaigns and tank inventories
Cm247		X			Exceeds Cm production for Cf campaigns and tank inventories
Cs135				6.54	< The 354 Ci total tank inventory
Es254	X		X		No good production pathway, short lived
Fe60			X		2n capture, short half-life after 1st capture (low production)
Ge68	X		X		No good production pathway, ~short lived
H3					
Hf172			X		No good production pathway
Hf182			X		No good production pathway, Long lived
Hg194			X		No good production pathway
I129				0.54	
In115		X		1.22E-02	Excessive Mass
K40		X			Excessive Isotope Mass and abundance (0.0117%) is low
Lu173	X		X	Shielded	Extremely low yield chain
Lu174				Shielded	Extremely low yield chain
Lu174m	X			Shielded	Extremely low yield chain
Lu176		X		Shielded	Extremely low yield chain
Md258			X		No good production pathway, short lived
Mn53		X	X		No good production pathway; Excessive Mass
Mo93			X	Shielded	Mo activation?
Mo93m			X	Shielded	6.35% Chain, short lived
Nb93m			X	Shielded	6.35% Chain; 1.5E6 year parent
Nb94			X	Shielded	Long Lived, greatly exceeds tank inventories
Ni59					Exceeds total tank inventories
Ni63					
Np236a			X		Long lived, (n,2n) on Np237
Np237					
Pa231					Long lived U235 daughter; Ci < tank inventory, Ref. 2
Pd107		X		0.146	Long lived (low activity)
Pt193					Pt Activation
Pt193m	X				Pt Activation, short lived
Pu237	X				Short lived
Pu238					
Pu239					
Pu240					
Pu241					
Pu242					
Pu244		X			Excessive Mass

Table 7-11. CA Radionuclides versus Expected SRS Production (Continued)

Nuclide	Delete			Fission Yield %	Comment
	Short Half-Life	Excessive Mass	Production Pathway		
Ra226					
Rb87		X		2.56	Extremely long lived, excessive mass
Re186m	X		X		Production low/unlikely
Re187		X			Extremely long lived, excessive mass
Si32			X		2n capture, first n short lived
Sr90				5.78	
Ta179	X		X		~Short lived, No good production pathway (n,2n)
Ta180m		X			Extremely long lived, Excessive mass
Tc97			X	Shielded	
Tc98			X	Shielded	
Tc99				6.11	
Th230		X			U234 daughter; not enough parent
Th231		X			Mass based on U235 first daughter equilibrium
Th232					
Ti44			X		No good production pathway
Tm171	X				Short lived
U231	X				Short lived
U233					Thorium Campaign
U234		X			Excessive Mass (> isotope with 0.0055% abundance)
U235		X			Excessive Mass (>20 metric tons)
U236		X			Excessive Mass (>6 metric tons), 235 capture
U238		X			Excessive Mass, (>1,000 metric tons)
Z93				6.35	

Five of the radionuclides marked for deletion in Table 7-11 are judged to be potentially significant and will be retained for the CA analyses. These are Mo93 (could be formed by neutron activation, a production pathway not considered in Sigg and Lee 2008), U234, U235, U236, and U238 (considerable quantities have been disposed in the ORWBG). This leaves 43 radionuclides that could be screened out on the basis of limited production.

7.2.5 Conclusions

Of the 77 radionuclides that did not screen out (Table 7-10), 43 should be screened out on the basis of limited production (i.e., secondary screening by Sigg and Lee 2008). However, ten of these which could be screened out by the secondary screening (i.e., Cf249, Cl36, Cm244, Cm246, Cm 247, K40, Nb94, Pd107, Pu244, and Th230) should not be screened out because they are key radionuclides in either the ELLWF or FTF PAs (Table 7-10). Also, four other radionuclides (i.e., Cm248, Pb210, Se79, and Th229), which were screened out by the primary screening in Run 1 will be retained because they are key radionuclides in either the ELLWF or FTF PAs (Table 7-10). In addition, Cs137 will be retained because it is a significant radionuclide in some of the stream bed sources.

This results in a set of 49 parent radionuclides to be analyzed in the CA. These parent radionuclides are depicted in Table 7-12.

Table 7-12. Parent Radionuclides to be Assessed in Detail in the SRS CA

Ag108m	H3	Pu241
Al26	I129	Pu242
Am241	K40	Pu244
Am242m	Lu174	Ra226
Am243	Mo93	Se79
Bi210m	Nb94	Sr90
C14	Ni59	Tc99
Ca41	Ni63	Th229
Cf249	Np237	Th230
Cl36	Pa231	Th232
Cm244	Pb210	U233
Cm245	Pd107	U234
Cm246	Pt193	U235
Cm247	Pu238	U236
Cm248	Pu239	U238
Cs135	Pu240	Zr93
Cs137		

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8.0 SOURCES AND INVENTORIES

This section summarizes the work conducted to accomplish Composite Analysis (CA) Tasks 4.2.7 and 4.2.8 in the Comprehensive SRS Composite Analysis Program Plan (Phifer and Cook 2007). The objectives of these tasks are to develop an updated radionuclide inventory for all locations at the Savannah River Site (SRS) that are expected to contain a radionuclide inventory when the SRS End State is reached. The SRS End State is expected to be achieved in the year 2025, with the exception of those facilities whose operational life is expected to extend beyond that date. This radionuclide inventory is required in order to perform the necessary analyses to demonstrate that the dose constraints spelled out in the CA guidance documents (Department of Energy (DOE) Order 435.1) will not be exceeded over the 1,000-year period of compliance. The supporting document for this section provides a more extensive description of the sources and facilities noted here (Hiergesell et al. 2008).

The focus of the CA is to identify those sources of residual radionuclides at SRS which could potentially interact with radionuclides transported from the low-level waste (LLW) disposal facilities, closed radioactive liquid waste storage tanks, and transuranic (TRU) waste disposals (i.e., facilities requiring a Performance Assessment (PA)) resulting in an additional contribution to the dose to the hypothetical member of the public beyond the expected dose attributable to those facilities. All of these PA facilities are located within the General Separations Areas (GSA), including E-, F-, H-, S- and Z-Areas.

To accomplish the objectives of CA Tasks 4.2.7 and 4.2.8, extensive consultation was conducted with the custodial organizations for individual facilities, waste tanks and waste sites to identify those which have had a process history associated with radionuclides. These organizations included Soil and Groundwater Closure Projects (SGCP), Liquid Waste Operations (LWO), Site Deactivation and Decommissioning (SDD), and Waste Management Area Project (WMAP). The following hierarchy was adhered to in developing the inventory estimates:

- Existing documented inventories were utilized where available
- Inventories were developed from existing sampling and analysis data, where available
- Inventories were developed from the inventories of other similar facilities/waste sites
- Inventories were developed from the operational inventories provided within Safety Analysis documents

An overriding philosophy in this effort is that the results of this effort are a “living” estimate, whereby improvements/refinements will be forthcoming in future CA updates if they are needed, and if new data should become available.

8.1 RADIONUCLIDE SOURCES

The objective of CA Task 4.2.7, "Radionuclide Source Locations" is to identify all radionuclide source locations over the entire SRS site that are projected to have a remaining radionuclide inventory after all DOE site operations have ceased and that could interact with the E-Area Low Level Waste Facility (ELLWF), Saltstone Disposal Facility (SDF), F- and H-Area Tank Farm (FTF and HTF) radioactive liquid waste storage tanks, F-Area Materials Storage (FAMS) and/or E-Area TRU Pad 1 at the CA Points of Assessment (POAs). This includes future facilities included in DOE long-range plans such as the Mixed Oxide Facility (MOX), Pit Disassembly and Conversion Facility (PDCF), Waste Solidification Building (WSB), Salt Waste Processing Facility (SWPF), etc. Documentation utilized to identify radionuclide source locations should include but not be limited to pertinent WMAP, LWO, Site Regulatory Integration and Planning (SRIP), SGCP, SDD, PA, CA, Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), Resource Conservation and Recovery Act (RCRA), National Environmental Policy Act (NEPA), and the Savannah River Federal Facilities Agreement (FFA) documentation.

Likewise, CA Task 4.2.8 is stated in the CA Program Plan, and documented in Phifer and Cook (2007). In brief, it is to determine the radionuclide inventory (each radionuclide and its quantity) for the ELLWF, SDF, FTF and HTF tanks, FAMS, E-Area TRU Pad 1, and all other SRS radionuclide source locations that have been determined to have a remaining radionuclide inventory after all DOE site operations have ceased and provide adequate documentation to justify the inventory. The radionuclide inventory shall consider both parents and daughters. The remaining radionuclide inventory after all DOE site operations have ceased will take into consideration planned CERCLA and RCRA remedial actions, planned Deactivation and Decommissioning (D&D), planned removal of waste from the FTF and HTF tanks, and other such actions intended to reduce the radionuclide inventory.

Documentation utilized to establish the radionuclide inventory should include, but not be limited to, pertinent WMAP, LWO, SRIP, SGCP, SDD, PA, CA, CERCLA, RCRA, NEPA, and FFA documentation. Where information on the radionuclide inventory is limited and extrapolations of the inventory must be made, the extrapolation should tend to produce a reasonable representation unlikely to underestimate the resulting dose over the long term and the method of extrapolation and its justification must be provided. Additionally during development of the radionuclide inventory, information necessary for the development of radionuclide source release rates for each source location will be gathered and documented. Information gathered should include any information on the physical and geochemical conditions that could affect the release rates (such as site characteristics, presence of a closure cap, packaging or containment, waste form, solubility, distribution coefficient, etc.).

In developing the estimates of remaining inventory after DOE site operations have ceased, it is necessary to consider the long-term plans to dispose of nuclear material, waste, and contaminated facilities presently on the site. The goal of the SRS Environmental Management (EM) Cleanup Project and resulting *SRS End State Vision* (DOE 2005) is to dispose of all EM nuclear material and waste hazards permanently, decommission all EM facilities and remediate all SRS inactive waste units. The vast majority of EM nuclear material and waste hazards will be permanently removed from SRS and dispositioned offsite. Inactive waste units will be remediated by deploying an area-by-area closure and deletion strategy. By 2025, all inactive waste sites that pose an unacceptable risk to surface water or groundwater will be remediated, and any contaminated groundwater will be remediated or undergoing remediation.

8.1.1 SRS Facilities that May Retain Residual Radionuclides at SRS End State

The initial steps to accomplishing CA Tasks 4.2.7 and 4.2.8 involved identifying all facilities and structures at SRS which might possibly retain residual radioactive material at the time that SRS facilities, waste tanks and waste sites are projected to have attained their End States, in the year 2025. Of the long list of facilities and waste sites, many either never had any radionuclides at their localities or are expected to be completely de-inventoried of radionuclides when the End State is reached. Extensive consultation with the custodial organizations to determine which ones are expected to retain residual radioactive material at SRS End State was conducted as a part of this investigation.

Two major federal laws drive environmental cleanup: the RCRA and the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), or Superfund. SRS is meeting the integrated requirements of these two laws through a FFA among DOE, United States Environmental Protection Agency (EPA) Region 4, and the South Carolina Department of Health and Environmental Control (SCDHEC). The FFA, effective August 16, 1993, specifies how SRS will address contamination or potential contamination at waste units, surplus facilities, and operable units in accordance with RCRA and CERCLA requirements. Once DOE declares a facility surplus it is added to the list of facilities to be decommissioned in Appendix K-1 of the FFA. D&D, by DOE, of facilities listed in Appendix K-1 is coordinated with EPA and SCDHEC as outlined within the FFA. A total of 515 waste sites at SRS that will be closed in accordance with RCRA and CERCLA are also listed in Appendices C.1, C.2, C.3, and C.4 of the FFA.

The term "End State" refers to the status of a facility or waste site after decommissioning and closure activities are complete. There are two possible End State alternatives for SRS facilities: Demolition or In-Situ Disposal (ISD) (WSRC 2003). Demolition includes demolishing and removing the entire facility to grade, and decontaminating as necessary to meet established release criteria. There may be variations among individual residual conditions within this End State category. ISD is the preferred End State for some structurally robust facilities for which demolition would be both very expensive and unnecessary. In this case, radiological and other hazardous material is removed as appropriate and the facility or waste tank is decontaminated to a level that meets established criteria, and additional barriers are emplaced as necessary. For waste sites, the potential End States are "no further action" (NFA) needed/required or "long-term stewardship" (LTS). NFA is the preferred End State when, upon completion of the characterization or remediation process, certain sites are determined as needing no further remedial action.

8.1.1.1 Facilities

The focus of the CA is to identify those sources of residual radionuclides at SRS which could potentially interact with radionuclides transported from the LLW disposal facilities, closed radioactive liquid waste storage tanks, and TRU waste disposals (i.e., facilities requiring a PA) resulting in an additional contribution to the dose to the hypothetical member of the public beyond the expected dose attributable to those facilities. All of these PA facilities are located within the GSA (including E-, F-, H-, S- and Z-Areas).

As outlined within Phifer and Wilhite (2008), the POAs, where a hypothetical member of the public can come into contact with radionuclides emanating from the PA facilities, are at the mouths of SRS surface streams and within the Savannah River. The primary transport pathways from SRS radionuclide source locations to these POAs are through leaching of the source to the groundwater and subsequent discharge of groundwater to surface streams (Wilhite and Phifer 2008). As outlined within Wilhite and Phifer (2008), exposure scenarios involving contact with, and use of, contaminated surface water were considered. Two exposure scenarios were judged to bound exposures, a recreational scenario and a residential scenario. Based upon this construct, those SRS facilities and waste sites with a projected End State radionuclide inventory that could potentially interact with radionuclides transported from the PA facilities, and thus contribute to the dose to the hypothetical member of the public will be evaluated in this CA.

8.1.1.2 Waste Tanks

A total of 51 radioactive liquid waste storage tanks are located in the F- and H-Areas that will be dispositioned to an ISD End State. These waste tanks will be emptied and filled with grout for final disposition. The actions may include annulus cleaning, water washing, acid cleaning, and/or other technologies, as necessary. Once these actions have been accomplished and the waste has been removed, the waste tank is sampled, characterized, and modeled, and documents prepared to obtain approval from DOE and SCDHEC to grout the waste tank.

8.1.1.3 Waste Sites

A total of 515 Waste Sites are identified in the FFA (EPA, DOE, and SCDHEC 1994) to be addressed. The SRS focus is to investigate and remediate, if needed, releases of hazardous substances (including radionuclides) to minimize or eliminate potential risks to human health and the environment. As of 2003, the remaining SRS scope consists of 211 waste sites consolidated into 100 projects for which characterization or remediation had not been initiated. The organization's scope, regulatory requirements, and execution strategy are defined in SGCP baseline documents for each of its projects. There are a variety of types of waste sites, including seepage basins, abandoned process sewer lines, burial grounds, groundwater units and integrator units (watersheds containing stream sediments and floodplains, canals and ponds).

8.1.2 Identification of Facilities/Closure Units with a Potential Contribution to Dose

The goal of consultation with the SRS facility custodial organizations was to first identify which facilities and waste sites might retain some level of radioactivity when the End State is reached and then to determine which of these could potentially contribute to the dose to a hypothetical member of the public at POAs over the 1,000-year compliance period. The approach to narrowing this list is described in the following sections.

8.1.2.1 Contribution Pathway

The focus of the CA is to identify those sources of residual radionuclides at SRS which could potentially interact with radionuclides transported from the LLW disposal facilities, closed FTF and HTF tanks, and TRU waste disposals resulting in an additional contribution to the dose to the hypothetical member of the public beyond the expected dose attributable to those facilities. All of these waste disposal facilities are located within the GSA (including F-, H-, and E-Areas).

For the purposes of this CA, the dose to the hypothetical member of the public will be acquired through recreational and residential use of surface water which receives discharge from groundwater contaminant plumes emanating from SRS facilities that have been closed, or placed into their End State. The locations of the POAs are at the mouths of the SRS streams that discharge to the Savannah River and the Savannah River itself.

8.1.2.2 Primary Screening Analysis Tool

The Vadose Zone Contaminant Migration Multi-Layered (VZCOMML©) Soil Screening Program (Rucker 1999; Rucker 2004; Rucker 2007) is utilized by Site Deactivation and Decommissioning (SDD) to determine whether, after deactivation and decommissioning (D&D), the facility remains (i.e., concrete slabs, low grade trenches and sumps, and building foundations) could potentially impact groundwater directly underneath the facility. Other equivalent models, which could potentially be used for such screening, include the Residual Radioactive Materials Computer Code (RESRAD), the Seasonal Soil Compartment Model (SESOIL), Reactor Screening Software 1 (RST1) (WSRC 2006), and the Contaminant Migration Screening (CMScreen) model (WSRC 2006a). The groundwater directly underneath the facility is considered impacted if the results of the screening indicated that the facility will cause groundwater concentrations to exceed the Safe Drinking Water Act (SDWA) Maximum Contaminant Levels (MCL). Of the 273 facilities, which have already undergone D&D, only 38 had a radiological history and of these 38 only seven had the potential to impact groundwater, based upon the VZCOMML© determination. Results of these SDD determinations along with the associated references are provided within WSRC 2008 Appendix A for each of the 273 facilities, which have already undergone D&D. As such, this VZCOMML© screening was not conducted as part of this CA; rather it is existing information referenced by this CA and used to determine which D&Ded facilities should be included in the CA inventory.

Such previous screening is a significant evaluation for the purposes of this CA. The rationale for this is that if it can be demonstrated that the groundwater directly beneath a D&Ded facility does not exceed the MCL with respect to any of the residual radionuclides, then any contribution to the dose to the CA hypothetical member of the public would be many orders of magnitude less by the time the plume migrated to the nearest stream and was greatly diluted by the clean surface water flow. Therefore, any potential contribution to dose from such facilities would be insignificant. In such cases it is reasonable to dismiss these facilities from further consideration in this CA.

VZCOMML© is a conservative spreadsheet-based model that can be used as a preliminary tool to evaluate the potential for vadose zone contamination to impact groundwater. It is used to screen sources of radioactive material in soil and concrete slabs (for concrete slabs, the source is modeled as if it were in soil; i.e., no credit is taken for the concrete). The model simulates transport in the vadose zone by steady-state one-dimensional flow and represents average flow conditions over the period of interest. The analysis approach employed within the VZCOMML© suite is consistent with the approved Contaminant Migration Protocol of the Federal Facility Implementation Management Plan and U.S. Environmental Protection Agency Soil Screening Guidance (EPA 1996).

The spreadsheet model:

- Estimates the theoretical peak groundwater concentration for an analyte at the surface of the water table
- Predicts the time to maximum groundwater concentration at a down gradient receptor located at the edge of the waste unit/facility by calculation of travel time in the vadose zone and application of a dilution factor in the aquifer

Assumptions relating to the nature of the model architecture (including the source term) and the analytical equations are such that the estimated groundwater concentrations are conservative, that is groundwater concentrations are somewhat overestimated (Dixon et al. 2005, Dixon et al. 2005a, Dixon et al. 2005b, and Dixon et al. 2008). Because the CA assesses impacts in surface water bodies and because the VZCOMML© suite conservatively calculates contaminant concentrations in the groundwater immediately adjacent to the source of radioactive material, use of the VZCOMML© suite for screening sources in the CA is conservative.

All D&Ded facilities, which do not require further remedial actions such as capping in order to be screened as not impacting groundwater with VZCOMML©, are not considered further in the CA. However even if a facility was screened out by VZCOMML© screening, but remedial actions, other than removal, such as capping, were conducted in association with the facility, then such facilities are considered as sources within the CA. For example, D&Ded buildings 677-T and 678-T were screened out, but a closure cap was installed over these building slabs due to their proximity to adjacent waste sites, which required a closure cap; therefore the radionuclide inventory associated with these buildings was included as a CA source.

8.1.2.3 Facilities with Potential to Contribute to Offsite Dose

Extensive consultation was conducted with the custodial organizations for individual facilities, waste tanks and waste sites to identify those which have had a process history associated with radionuclides. These organizations included SGCP, LWO, SDD, and WMAP. These organizations have performed analyses to determine which of these facilities pose a risk to workers or members of the public. From the perspective of the CA, the particular risk that is of interest is whether the groundwater beneath each of the facilities is impacted at levels that exceed SDWA MCL. The rationale for this is that if a facility can be demonstrated to not impact the groundwater in excess of the MCL, then there is no risk of adding to the dose received by a general member of the public by intermingling with the releases from the SRS LLW disposal facilities. This logic was addressed in Section 8.1.2.2.

The closure organizations, SGCP and SDD, have performed these groundwater screening analyses for the facilities and waste sites within their jurisdiction. This information, along with the judgment of the custodial organizations was employed to eliminate those facilities with no potential to contribute to the hypothetical member of the public from further CA consideration and to list those facilities to target for acquiring radionuclide inventories. In addition, certain operational facilities that were established to handle or process significant quantities of radionuclides were also evaluated. Some of these facilities are still in active operation and may contain a significant inventory of certain radionuclides when they are demolished or placed in an ISD configuration at their End State.

This list of target facilities and waste sites is presented in Table 8-1. These are categorized by SRS operational area as Primary Facilities and when appropriate Sub-Unit facilities are also identified. The custodial organization of each unit is also listed.

Table 8-1. SRS Facilities Expected to Retain a Significant Radionuclide Inventory at the End State

SRS Area	Primary Facility	Sub-Unit Facilities	Custodial Organization[1]
A/M Areas	M-Area HWMF	- -	SGCP
	MIPSL	- -	SGCP
	SRL Seepage Basins	- -	SGCP
	SRNL Buildings	Building 773-A	Other[2]
		Building 776-A	Other
		Building 778-A	Other
	Tims Branch/Steed Pond	- -	SGCP
D-Area	Building 420-D	- -	SDD
	Building 420-2D	- -	SDD
	Building 421-D	- -	SDD
	Building 421-2D	- -	SDD
	Building 772-D	- -	SDD
E-Area	E-Area Disposal Facilities	CIG Trenches	WMAP
		Engineered Trenches	WMAP
		ILV	WMAP
		LAWV	WMAP
		Naval Reactor Pads (643-7E, 643-26E)	WMAP
		Slit Trenches (East)	WMAP
		Slit Trenches (Central)	WMAP
		Slit Trenches (West)	WMAP
		TRU Pad #1	WMAP
	MWMF	- -	SGCP
	ORWBG	- -	SGCP

Table 8-1. SRS Facilities Expected to Retain a Significant Radionuclide Inventory at the End State – continued

SRS Area	Primary Facility	Sub-Unit Facilities	Custodial Organization
F-Area	FAMS	- -	Other
	FIPSL	- -	SGCP
	F-Canyon and FB Line	FB Line	Other
		F-Canyon	Other
		F-Canyon Process Vessels	Other
	Canyon Auxiliary Facilities, Building 211-F	Waste Handling Vault Sludge Tanks Segregated Solvent Facility Acid Recovery Unit Water Handling Facility General Purpose Tanks General Purpose Evaporators	Other
	F-Retention Basin 281-3F	- -	SGCP
	F-Seepage Basins	- -	SGCP
	F-Area Tank Farm	22 Tanks: 1-8, 17-20, 25-47 FTF Ancillary Equipment	LWO
	Central Labs	Building 772-F	Other
		Building 772-1F	Other
		Building 772-4F	Other
	Old F-Area Seepage Basin	- -	SGCP
H-Area	H-Area Tank Farm	29 Tanks: 9-16, 21-24, 29-32, 35-43, 48-51	LWO
		HTF Ancillary Equipment	LWO
	HIPSL	- -	SGCP
	H-Retention Basins	281-1H, -2H, -8H	SGCP
	H-Area Seepage Basins	- -	SGCP
	HB-Line	- -	Other
	H-Canyon	- -	Other
	RBOF	- -	Other
	Sand Filter	- -	Other
	Tritium	Building 232-H	Other
		HANM Building	Other
		HAOM Building	Other
		Vault 217-H	Other
		Building 234-7H	Other
		Building 236-H	Other
		Building 237/238H	Other
	Tritium Extraction Facility	- -	Other
	Remote Handling Building	Building 264-2H	Other
		TPBARs Activation Products	Other

Table 8-1. SRS Facilities Expected to Retain a Significant Radionuclide Inventory at the End State – continued

SRS Area	Primary Facility	Sub-Unit Facilities	Custodial Organization
N-Area Reactors	Ford Bldg Seepage Basin	- -	SGCP
	Reactor Buildings	(P,R,C,K,L Reactor Buildings)	SDD/SGCP[3]
	Reactor Seepage Basins	(P,R,C,K,L Seepage Basins)	SGCP
	Reactor Cask Car Railroad Tracks	(P,R,C,K,L)	SGCP
	P-Reactor Outfalls P02 and P007	- -	SGCP
S-Area	DWPF	Building 221-S	LWO
		LPPP	LWO
		Sandfilter	LWO
TNX Area	TNX Burying Grounds	- -	SGCP
	Old TNX Seepage Basin	IPSL	SGCP
	TNX-Discharge Gully	- -	SGCP
	TNX Outfall Delta/Swamp	- -	SGCP
	Bldg. 677-T and 678-T	- -	SGCP
Z-Area	Saltstone Facility	Vault 1	LWO
		Vault 4	LWO
		Vault 2 Type Cells (64)	LWO
General Site	Integrator Op. Units	Fourmile Branch[4]	SGCP
		Lower Three Runs	SGCP
		Pen Branch	SGCP
		SR and Floodplain Swamp	SGCP
		Steel Creek	SGCP
		Upper Three Runs	SGCP
	GW Closure Units	(C-, D-, F-, H-, K-, L-(south), L-(north), P-, and R-Groundwater OUs)	SGCP
	Miscellaneous Spills	FTF Spills (Tanks 3 and 8 and 281-3F) HTF Spills (Tanks 9, 13, 16, and 37)	SGCP
SRS Future Facilities	SWPF	Alpha Absorption Tank-A	LWO
		Filter Feed Tank-A	LWO
		Filter Feed Tank-B	LWO
		Sludge Solid Receipt Tank (SSRT)	LWO
		Salt Solution Feed Tank (SSFT)	LWO
		Strip Effluent Hold Tank	LWO
	MOX	- -	Other

[1] During the process of producing the CA several organizational and DOE contractual changes have occurred. Organizational names used within the CA are generally those in effect when the CA was initiated. The following are some of the primary changes of importance to the CA:

- SGCP and SDD were combined to become Area Closure Project (ACP)
- WMAP became Solid Waste Management (SWM)
- LWO was awarded as a separate DOE contract to Savannah River Remediation (SRR)

[2] Other refers to custodial organizations other than SGCP, SDD, and WMAP, for which the facility inventory was derived from facility specific safety analysis documents.
[3] Nuclear Material Storage (NMS) Project is currently the custodian for C, K, and L reactor buildings; however SDD/SGCP will assume custodial duties for reactor remediation and deactivation and decommissioning.
[4] Within various SRS documents Fourmile Branch has also been denoted as Four Mile Branch, Four Mile Creek, and Fourmile Creek. Within this document the stream itself will be denoted Fourmile Branch and the associated Integrator Operable Unit (IOU) will be denoted Fourmile Branch IOU.

8.1.3 Radionuclide Inventory Compilation

8.1.3.1 General Considerations

An overriding philosophy in this effort is that the results of this effort are a "living" estimate, whereby improvements/refinements will be forthcoming in future CA updates if they are needed, and if new data should become available. The following hierarchy was adhered to in developing the inventory estimates provided:

- Existing documented inventories were utilized where available
- Inventories were developed from existing sampling and analysis data, where available
- Inventories were developed by interpolation from the inventories of other similar facilities/waste sites
- Inventories were developed by interpolation from the operational inventories provided within Safety Analysis documents

In addition, a conservative bias was implemented, where reasonable, in calculations to determine the End State radionuclide inventory for a given facility/waste site. In this context, conservative is taken to mean that the actual calculated residual radionuclide inventory is overestimated rather than underestimated.

More information on the specific approach implemented to generate the radionuclide inventories of each facility is provided in the section of the Inventory Results provided for each facility evaluated in this report. In addition, design checks of calculations performed to develop these inventories have been performed and documented.

8.1.3.2 Currently Operating and Future SRS Facilities

For future facilities, or facilities that are still in an operational mode, a method has been developed to estimate the residual inventory that might be left at the end of DOE operations. The method consists of using facilities for which safety documentation, both during operation and following deactivation, exists. The ratio of inventories provides an estimate of the factor by which the operational inventory might be reduced prior to reaching the End State. Two potential End States are considered, in situ disposal and demolition, consistent with the current End State Vision report.

Two facilities were used in this study, the F Canyon complex and the 321-M building. Operating limits for each were taken from Safety Analysis documentation from the time frame during which they were functional. Safety documentation was done on the F Canyon complex when deactivation work was halted (WSRC 2007). Deactivation work was halted before the planned deactivation activities were completed, which means that the residuals are greater than were originally planned and the calculated Reduction Factor is larger than would generally be expected. Comparison of the operating limits (WSRC 1996) with the inventory of the primary radionuclide processed in the facility, Pu239, remaining after deactivation of the F Canyon complex gives one estimate of the in situ disposal reduction factor (Table 8-2).

For 321-M, operational limits were taken from the 1991 version of the Nuclear Criticality Technical Standard for the 300 Area (WSRC 1991). A Hazard Classification report done following deactivation (Musall 2000) provided the basis for a second estimate of the in situ disposal reduction factor (Table 8-3). The results of verification sampling of the 321-M building slab following demolition (WSRC 2006b) were used to estimate the demolition End State (Table 8-4). Information for the primary radionuclide processed in the facility, U235, was used. The allowable inventory of the storage vaults was not included in these calculations, because not every operating facility has storage vaults and the calculation is a generic estimate.

Table 8-2. F Canyon In Situ Disposal Reduction Factors

Radionuclide	Operational Limit (Ci) (WSRC 1996)	Deactivation Residual (Ci) (WSRC 2007)	In Situ Reduction Factor
Pu239	1.56E+05	4.95E+03	3.18E-02

Table 8-3. 321-M In Situ Disposal Reduction Factors

Radionuclide	Operational Limit (Ci) (WSRC 1991)	Deactivation Residual (Ci) (Musall 2000)	In Situ Reduction Factor
U235	2.98E-01	8.99E-04	3.02E-03

Table 8-4. 321-M Demolition Reduction Factors

Radionuclide	Operational Limit (Ci)	Slab Residual (Ci) (WSRC 2006b)	Slab Reduction Factor
U235	2.98E-01	2.53E-04	8.48E-04

Based on these calculations, estimates for ISD and Demolition Reduction Factors are 1E-2 and 1E-3, respectively. A Reduction Factor versus the current inventory after partial deactivation of one should be used for the F Canyon Complex unless further deactivation is performed.

8.1.3.3 Short-lived Progeny of Cs137 and Sr90

For two isotopes commonly associated with SRS operations and closure units and frequently occurring at relatively high inventories, namely Cs137 and Sr90, significant inventories of their daughter isotopes, Ba137m (daughter of Cs137) and Y90 (daughter of Sr90) will always be present as well. These progeny have extremely short half-lives but will always be present because they exist in secular equilibrium with their parent isotopes. The half-lives of these two isotopes are 2.55 minutes and 64.1 hours for Ba137m and Y90, respectively. The policy of this CA is to account for the dose contributed by these daughter isotopes to the hypothetically exposed member of the public by assuming concentrations of these daughters in the streams and river based upon secular equilibrium with their respective parent. Therefore, because the dose contribution is accounted for and because the half-lives of the progeny are so short, the inventories of Ba137m and Y90 are not listed in the inventory information that is presented for each facility in this report.

8.2 RADIONUCLIDE SOURCE LOCATIONS AND INVENTORY

8.2.1 A/M Facilities

The A/M Facilities, which are projected to contain a radionuclide inventory when the SRS End State is reached, include the M-Area Hazardous Waste Management Facility (HWMF), the M-Area Inactive Process Sewer Line, the Savannah River Lab Seepage Basins, the Savannah River National Laboratory (SRNL) Buildings, and Tims Branch and Steed Pond.

8.2.1.1 M-Area HWMF

8.2.1.1.1 Background

The M-Area of the SRS is located near the northwest edge of SRS and contained manufacturing facilities for nuclear fuel and target components, offices and research facilities. The M-Area HWMF consists of an unlined surface impoundment (settling basin), a portion of an inactive process sewer line, drainage and seepage areas, and a Carolina Bay known as the Lost Lake. The manufacturing processes associated with the M-Area HWMF consisted of aluminum-forming and metal-finishing processes used to produce fuel and targets for the SRS reactors. Waste effluents were discharged from three production buildings and two support laboratories to a settling basin through an underground process sewer line. The waste effluents consisted of hydroxides, precipitates of aluminum, uranium, nickel, lead and other metals, solvents, acids, and caustics. The M-Area HWMF has been designated as a source-specific Operable Unit (OU) within the A/M Area Fundamental Study Area (WSRC 2007a).

Remedial actions for the M-Area HWMF, which included dewatering the basin, treating the basin liquid, waste consolidation, excavation of the sewer line and associated soils, and installation of a low-permeability cap, were completed in 1990, and the basin closure was certified in 1991.

8.2.1.1.2 Inventory and Basis for Estimate

The basis of the M-Area HWMF radionuclide inventory is the inventory reported in ERD (2001). This report lists the mass for all Uranium isotopes, which are assumed to be Highly Depleted Uranium (HDU). The relative percentages (by mass) of U238, U235 and U234 contained in HDU are assumed to be 9.98E-01, 2.0E-03 and 5.0E-06 for these isotopes, respectively. The total mass reported in ERD (2001) was apportioned between these uranium isotopes by these fractions and converted to curies using the specific activity of each isotope. The 2002 radionuclide inventory for the M-Area HWMF is provided in Table A-1.

8.2.1.2 M-Area Inactive Process Sewer Line

8.2.1.2.1 Background

The M-Area Inactive Process Sewer Line Operable Unit (MIPSL OU) comprises portions of the M-Area Settling Basin Inactive Process Sewer, including approximately 3,800 feet of underground piping (1,283 feet of the M-Area Inactive Process Sewer (MIPS) and 2,520 feet of 313-MIPS). From 1958 until early 1985, several M-Area facilities (313-M, 320-M, and 321-M) manufactured reactor fuel and target assemblies. Associated operations included support buildings, maintenance operations, laboratories, and infrastructure for managing waste. Effluents from M Area were transported through two separate networks of vitrified clay pipes. The MIPS network discharged waste to the M-Area Settling Basin, and the 313-MIPS network released waste to the A-014 Outfall, which flowed to a tributary of Tims Branch. For a time in 1982, the 313-MIPS process waters were temporarily diverted from Tims Branch to conjoin with MIPS process waters already flowing to the M-Area Settling Basin.

M-Area effluent wastes included chlorinated solvents (used for degreasing fuel and target assemblies), acids, caustics, heavy metals, and HDU. The scope of the MIPSL OU remedial action is limited to vadose zone soils. The related actions are in the construction phase at this time (WSRC 2007a).

8.2.1.2.2 Inventory and Basis for Estimate

The basis of the MIPSL radionuclide inventory is the inventory reported in ERD (2001). This report lists the mass for all uranium isotopes, which are assumed to be HDU. The relative percentages (by mass) of U238, U235 and U234 contained in HDU are assumed to be 9.98E-01, 2.0E-03 and 5.0E-06 for these isotopes, respectively. The total mass reported in ERD (2001) was apportioned among these uranium isotopes by these fractions and converted to curies using the specific activity of each isotope. The 2002 radionuclide inventory for the MIPSL is provided in Table A-2.

8.2.1.3 Savannah River Lab Seepage Basins

8.2.1.3.1 Background

The Savannah River Laboratory Seepage Basins (SRLSB) OU is located within the northwestern section of SRS, approximately 4,000 feet from the nearest SRS boundary and 4,500 feet from the nearest residence. The four basins lie within the northern portions of the A/M Area, northeast of Savannah River Ecology Laboratory (SREL) and southeast of the SRNL. The four basins were connected by a series of sequential overflow channels designed to receive wastewater by overflow from Basin 1 to Basin 4. Basins 1 and 2 (904-53G) were placed in operation in 1954. Basins 3 (904-54G) and 4 (904-55G) were added in 1958 and 1960, respectively. The basins were used from 1954 to 1982 to dispose of low-level radioactive liquid waste generated in the laboratories located in Building 735-A and 773-A. The basins and the process sewer line were taken out of operation in 1982 (WSRC 2007a).

Under a 1999 ROD, the most contaminated soil and pipeline were excavated and disposed at Envirocare of Utah, Inc.; the basins were backfilled to grade and revegetated; and institutional control of the covered basins is maintained to prevent unauthorized access.

8.2.1.3.2 Inventory and Basis for Estimate

The radionuclide inventory reported in ERD (2001) for the COCs was utilized as the best available estimate for use in this CA. COCs were identified following an evaluation of the potential impact to the groundwater. Because a low-permeability cap has not been placed over the basins, this list is adopted as the relevant list for this CA. This inventory represents the radionuclide inventory that remains in the SRLSB after the removal of contaminated soil for offsite shipment and disposal. The 2002 radionuclide inventory for the Savannah River Laboratory Seepage Basins is provided in Table A-3.

8.2.1.4 SRNL Buildings

8.2.1.4.1 Background Information

The SRNL buildings listed below are all located in the northwestern portion of the SRS.

8.2.1.4.1.1 Building 773-A – Main Laboratory

The Main Laboratory, Building 773-A, is the primary Research and Development (R&D) facility in the SRNL Technical Area. The building is classified as a nuclear, Hazard Category (HC)-2 facility. It contains offices, a technical library, chemical stores, chemical and radiochemical laboratories, shielded cell facilities, instrument development laboratories, machine and glass shops, gloveboxes, three dry cobalt irradiators, and ventilated radiological/chemical hoods. One section was dedicated to the development and testing of production reactor fuel and target assemblies. The SRS Solid Waste Assay Facility is presently located in 773-A. (WSRC 2007b)

8.2.1.4.1.2 Buildings 776-1A through 776-6A – Radioactive Liquid Waste Handling Facilities

Radioactive aqueous sample returns from Building 773-A are drained by gravity through pipes in the 904-A Process Trench to the Radioactive Liquid Waste (Sample Returns) Handling Facilities located on the east side of the SRNL Technical Area. These buildings are classified as a nuclear, HC-3 facility. Separate High Activity Drain (HAD) and Low Activity Drain (LAD) waste drains are provided. A spare drain is installed for each waste drain (WSRC 2007b).

8.2.1.4.1.3 Buildings 778-1A and 778-2A – Solid Waste Handling and Staging Facility

The 778-A Solid Waste Handling and Staging Facility (Buildings 778-1A and 778-2A) consists of various facilities where both low- and high-activity laboratory solid wastes generated from the SRNL Technical Area facilities are temporarily staged while awaiting final disposal. These buildings are classified as nuclear, HC-3 facilities (WSRC 2007b).

8.2.1.4.2 Inventory and Basis for Estimate

The inventory estimates developed for the nuclear facilities within the SRNL area were based on the operating facility inventories analyzed in the 2007 Documented Safety Analysis (DSA) (WSRC 2007b). An End State Reduction Factor of 1E-3 has been applied to approximate the fraction of the operating facility inventory that might remain with an End State configuration of "Demolish" (Table 4.3b, Appendix J, DOE 2005).

8.2.1.4.2.1 773-A

The DSA uses inventories for each room within the 773-A building, which handles radioactive material, based on the mission of that lab at the time the DSA was prepared (see Table 3.4.1-1 of the DSA). Because the inventories are used to set operating limits for the labs, they are bounding estimates. The estimates for all of the labs were summed to give building totals. A table from Apperson (1983) was used to calculate inventories of some radionuclides of importance to PA and CA analyses that are not of concern to Safety Analyses, such as C14, Tc99 and I129, based on the inventory estimates of Cs137. Table 3.4.1-3 in the DSA was used to convert inventories given as "Pu239/241" and "Pu242" into individual Pu and Am radionuclides. Building 773-A is an operating facility, which is not slated for closure until the assumed SRS end state date of 2025. Therefore the end state inventory estimated from the 2007 DSA is assumed to occur in 2025 at the building's end state. For the CA analysis, the residual radioactive material should be considered to be distributed within a concrete slab.

8.2.1.4.2.2 776-A Complex

The DSA uses inventories for the High Activity Waste tanks within the 776-A complex, which are used to store the output from the SRNL High Activity Drains. Because the inventories are used to set operating limits for the labs, they are bounding estimates. Table 3.4.1-3 in the DSA was used to convert inventories given as "Pu239/241" and "Pu242" into individual Pu and Am radionuclides. The 776-A Complex is an operating facility, which is not slated for closure until the assumed SRS end state date of 2025. Therefore the end state inventory estimated from the 2007 DSA is assumed to occur in 2025 at the complex's end state. For the CA analysis, the residual radioactive material should be considered to be distributed within a concrete slab.

8.2.1.4.2.3 778-A Waste Storage Area

The DSA uses inventories for the Waste Storage Area within the 778-A complex which are used to store drummed waste from the SRNL labs. Because the inventories are used to set operating limits for the labs, they are bounding estimates. Table 3.4.1-3 in the DSA was used to convert inventories given as "Pu239/241" and "Pu242" into individual Pu and Am radionuclides. The 778-A Waste Storage Area is an operating facility, which is not slated for closure until the assumed SRS end state date of 2025. Therefore the end state inventory estimated from the 2007 DSA is assumed to occur in 2025 at the area's end state. For the CA analysis the residual radioactive material should be considered to be distributed within a concrete slab. This area is a candidate to be removed from the CA analysis because the drummed waste will be disposed elsewhere and no radioactive material should escape the drums and remain as a residual contamination at 778-A.

The estimated 2025 radionuclide inventories for the SRNL Buildings 773-A, 776-A, and 778-A are provided in Table A-4, Table A-5, and Table A-6, respectively.

8.2.1.5 Tims Branch and Steed Pond

8.2.1.5.1 Background

Tims Branch is a tributary of Upper Three Runs Creek and is located in the northwest part of SRS, between the A/M Areas and Upper Three Runs Creek. Steed Pond is the result of an impoundment on Tims Branch. Both of these are evaluated separately from the Upper Three Runs Integrator Operable Unit and hence, a separate radionuclide inventory is developed for each. Tims Branch receives discharge from the A-01 and A-14 Outfalls.

8.2.1.5.2 Inventory and Basis for Estimate

The radionuclide inventory reported in ERD (2001) was utilized as the best available estimate for use in these facilities. This report lists a composite mass for all Uranium isotopes, which are assumed to be HDU since Tims Branch and Steed Pond received effluent through the MIPSL from M-Area (Section 8.2.1.2). The relative percentages (by mass) of U238, U235 and U234 contained in HDU are assumed to be 9.98E-01, 2.0E-03 and 5.0E-06 for these isotopes, respectively. The total mass reported in ERD (2001) was apportioned between these uranium isotopes by these fractions and converted to curies using the specific activity of each isotope. The 2002 radionuclide inventory associated with Steed Pond and Tims Branch is presented in Table A-7.

8.2.2 D-Area Facilities

D-Area is located about 1 mile from the Savannah River, near the west boundary of the SRS. Decommissioning of the D-Area facilities, where heavy water was reprocessed, involved demolition of the building super structures to the concrete slabs under the auspice of the National Environmental Policy Act categorical exclusion designation. The residual radionuclide contamination remaining in the concrete slabs was determined through a final field verification survey. The analytical results were documented in various, site-specific Decommissioning Project Final Reports (DPFRs).

8.2.2.1 420-D (Concentrator Building)

8.2.2.1.1 Background

The 420-D Concentrator Building built in 1953 was a 167 feet by 37 feet primarily open steel frame structure with enclosed control, pump, compressor, and electrical rooms. The Building 420-D processes initially concentrated 60% pure heavy water from the adjacent Girdler Sulfide Process Units to meet SRS reactor specifications. One of the two Building 420-D distillation processes was rededicated in 1965 to reconcentrating degraded moderator from SRS reactors.

Following the removal of contaminated components from Building 420-D, the structure was demolished to the foundation slab. The concrete slab remains in place and penetrations greater than two inches in diameter were grouted. A groundwater evaluation determined that volumetric tritium contamination of the remaining slab exceeds the MCL, as determined by the VZCOMML© screening analysis (Dixon and Flach 2006). All other contaminants were evaluated and determined not to represent a threat to the groundwater (WSRC 2008).

8.2.2.1.2 Inventory and Basis for Estimate

Several samples of the remaining materials (i.e., concrete slab) were analyzed to verify that the decommissioning and decontamination activities were complete. Slab remnants were divided into exposure areas (EAs) and each EA was evaluated to determine the total curies (Ci) of each detectable radionuclide based on the area of the EA and assuming that the results apply to the top six inches of the slab for tritium and the top two inches for other radionuclides. Appendix B in WSRC (2008) provides the analysis inputs for Building 420-D.

Tritium was detected in all of the building remnants that had a radiological history in D-Area by the final verification surveys. The amounts of tritium detected in building remnants for 420-D were determined to have potential impact to groundwater based on the analytical results and using VZCOMML©. Strategies were considered to remove the remnants but were abandoned because the removals would likely increase the probability of an impact and would just relocate the tritium contamination to another area at SRS. The decision was made to leave the slabs intact to act as an impervious barrier to retard any potential groundwater impact.

The final verification surveys divided the facilities into EAs. The EA for radiological assessments was approximately 1,000 square feet in most cases. The facility geometry may have dictated a larger or smaller area. The inventory was calculated by using the average of the analytical results for an EA. While only tritium was determined to pose a threat to the groundwater, the other radionuclides detected in analytical samples are also included to provide a more complete inventory. The 2006 radionuclide inventory for Building 420-D is provided in Table A-8.

8.2.2.2 420-2D (Rework Handling Facility)

8.2.2.2.1 Background

The Rework Handling Facility, Building 420-2D was constructed in 1974. Drums from the Rework Unit at Building 420-D, which purified heavy water (deuterium oxide), were emptied and filled there. It was a single-story, metal-sided structure with a metal roof, constructed on a concrete slab. Building 420-2D had a footprint area of approximately 400 square feet. The facility was in use until 1999.

Following the removal of contaminated components from Building 420-2D, the structure was demolished to the foundation slab. The concrete slab remains in place and penetrations greater than two inches in diameter were grouted. A groundwater evaluation determined that volumetric tritium contamination of the remaining slab exceeds the MCL, as determined by the VZCOMML© screening analysis (Dixon and Flach 2006). All other contaminants were evaluated and determined not to represent a threat to the groundwater (WSRC 2008).

8.2.2.2.2 Inventory and Basis for Estimate

Several samples of the remaining materials (i.e., concrete slab) were analyzed to verify that the decommissioning and decontamination activities were complete. Slab remnants were divided into EAs and each exposure area was evaluated to determine the total curies of each detectable radionuclide based on the area of the EA and assuming the results apply to the top six inches of the slab for tritium and two inches for other radionuclides. Appendix B in WSRC (2008) provides the analysis inputs for Building 420-2D.

Tritium was detected in all building remnants that had a radiological history in D-Area by the final verification surveys. The amounts of tritium detected in building remnants for 420-2D were determined to have potential impact to groundwater based on the analytical results and using VZCOMML©. Strategies were considered to remove the remnants but were abandoned because the removals would likely increase the probability of an impact and would just relocate the tritium contamination to another area at SRS. The decision was made to leave the slabs intact to act as an impervious barrier to retard any potential groundwater impact.

The final verification surveys divided the facilities into EAs. The EAs for radiological assessments were approximately 1,000 square feet in most cases. The facility geometry may have dictated a larger or smaller area. The inventory was calculated by using the average of the analytical results for an EA. While only tritium was determined to pose a threat to the groundwater, the other radionuclides detected in analytical samples are also included to provide a more complete inventory. The 2005 radionuclide inventory for building 420-2D is provided in Table A-9.

8.2.2.3 421-D (Finishing Building)

8.2.2.3.1 Background

The finishing building was used to increase the concentration of heavy water coming from the DW process from 98 percent to 99.8 percent or higher. The finishing building was a multi-level building with a structural steel frame covered with corrugated asbestos siding and roofing. Building 421-D had a footprint area of approximately 5,000 square feet (Reed and Swanson 2009).

Following the characterization and subsequent removal of contaminated components from Building 421-D, the structure was demolished to the foundation. The concrete slab remains in place and penetrations greater than two inches in diameter were grouted. A VZCOMML© model was developed for D-Area. The model identified specific contaminants that could impact groundwater within 1,000 years in D-Area (Dixon and Flach 2006). Of this list, tritium was the only COC identified by the VZCOMML© screening analysis (WSRC 2008).

8.2.2.3.2 Inventory and Basis for Estimate

Several samples of the remaining materials (i.e., concrete slab) were analyzed to verify that the decommissioning and decontamination activities were complete. Slab remnants were divided into EAs and each exposure area was evaluated to determine the total curies of each detectable radionuclide based on the area of the EA and assuming the results apply to the top six inches of the slab for tritium and two inches for other radionuclides. Appendix B in WSRC (2008) provides the analysis inputs for Building 421-D.

The final verification surveys divided the facilities into EAs. The EAs for radiological assessments were approximately 1,000 square feet in most cases. The facility geometry may have dictated a larger or smaller area. The inventory was calculated by using the average of the analytical results for an EA. The 2005 radionuclide inventory for building 421-D is provided in Table A-10.

8.2.2.4 421-2D (Moderator Handling and Storage)

8.2.2.4.1 Background

The Building 421-2D Moderator Handling and Storage Facility was built in 1953 to house an addition to the Building 421-D E Process (deuterium gas). In the 1960s, electrolytic cells were replaced with equipment for purifying degraded reactor moderator to meet the original heavy water specifications. Facility modifications added air monitoring, drum washing, drum testing, ion exchange columns, waste water collection, indoor drum storage, and secondary containment. Operations ceased in 1998 and Building 421-2D was transitioned to an inactive facility.

Following the removal of contaminated components from the process areas in and around Building 421-2D, the structure was demolished to the foundation. The concrete slab remains in place and penetrations greater than two inches in diameter were grouted (WSRC 2004).

A groundwater evaluation determined that volumetric tritium contamination of the remaining slab exceeds the VZCOMML© groundwater standards (Dixon and Flach 2006). All other contaminants were evaluated and determined not to represent a threat to the groundwater.

8.2.2.4.2 Inventory and Basis for Estimate

Several samples of the remaining materials (i.e., concrete slab) were analyzed to verify that the decommissioning and decontamination activities were complete. Slab remnants were divided into EAs and each exposure area was evaluated to determine the total curies of each detectable radionuclide based on the area of the EA and assuming the results apply to the top six inches of the slab for tritium and two inches for other radionuclides. Appendix B in WSRC (2008) provides the analysis inputs for Building 421-2D.

The final verification surveys divided the facilities into EAs. The EAs for radiological assessments were approximately 1,000 square feet in most cases. The facility geometry may have dictated a larger or smaller area. The inventory was calculated by using the average of the analytical results for an EA. While only tritium was determined to pose a threat to the groundwater, the other radionuclides detected in analytical samples are also included to provide a more complete inventory. The 2004 radionuclide inventory for building 421-2D is provided in Table A-11.

8.2.2.5 Building 772-D (Control Laboratory and Supervisor Office)

8.2.2.5.1 Background

The 772-D Control Laboratory and Supervisor Office was built in 1953 to provide sample analysis for heavy water production control, office space, and a cafeteria for the D-Area facilities. Facility modifications to install radiological controls allowed moderator chemistry control analyses in support of the Reactors and the D-Area moderator purification operations to be performed in the lab. Later various analyses in support of environmental programs were performed in the lab. Water Chemistry Lab operations were relocated to Building 772-F in 1998 and Building 772-D was transitioned to an inactive facility along with the other remaining heavy water facilities.

The above ground portion of the structure was demolished to grade elevation and all interfacing utilities were isolated, disconnected, and plugged. Floor openings were filled to achieve a uniform surface, and the surrounding soil that was disturbed during decommissioning was graded and seeded (WSRC 2008).

A groundwater evaluation determined that volumetric tritium contamination of the remaining slab exceeds the VZCOMML© groundwater standards (Dixon and Flach 2006). All other contaminants were evaluated and determined not to represent a threat to the groundwater (WSRC 2004a).

8.2.2.5.2 Inventory and Basis for Estimate

Several samples of the remaining materials (i.e., concrete slab) were analyzed to verify that the decommissioning and decontamination activities were complete. Slab remnants were divided into EAs and each EA was evaluated to determine the total curies of each detectable radionuclide based on the area of the EA and assuming that the results applied to the top six inches of the slab for tritium and the top two inches for other radionuclides. Appendix B in WSRC (2008) provides the analysis inputs for Building 772-D.

The final verification surveys divided the facilities into EAs. The EAs for radiological assessments were approximately 1,000 square feet in most cases. The facility geometry may have dictated a larger or smaller area. The inventory was calculated by using the average of the analytical results for an EA. While only tritium was determined to pose a threat to the groundwater, the other radionuclides detected in analytical samples are also included to provide a more complete inventory. The 2004 radionuclide inventory for building 772-D is provided in Table A-12.

8.2.3 E-Area Facilities

8.2.3.1 ELLWF Disposal Facilities

The background information provided herein in association with the ELLWF was derived from the ELLWF PA (WSRC 2008a). The ELLWF is located in the central region of the SRS known as the GSA. Radiological operations at the ELLWF began in 1994. The current ELLWF area developed for disposal consists of approximately 100 acres and a surrounding buffer zone that extends out to the 100-m point of compliance. It is an elbow-shaped, cleared area, which curves to the northwest, situated immediately north of the Mixed Waste Management Facility (MWMF). Disposal units within the footprint of the ELLWF include the Slit Trenches, Engineered Trenches, Components-in-Grout (CIG) Trenches, the Low Activity Waste Vault (LAWV), the Intermediate Level Vault (ILV), and the Naval Reactor Component Disposal Areas (NRCDAs). Final closure of the Slit Trenches, Engineered Trenches, CIG Trenches, LAWV, ILV, and NRCDAs will take place at final closure of the entire ELLWF at the end of the 100-year institutional control period (WSRC 2004b).

8.2.3.1.1 <u>Inventory and Basis for Estimate</u>

WMAP LLW Engineering developed projected disposal inventories at closure for each type of ELLWF unit on a footprint basis (footprints generally taken as a 157-foot by 656-foot area). The projected inventory at closure (i.e., 2025) was based upon disposal history and projections and a review of the PA limits. In the case of the closed NRDCA in 643-7E, actual disposed inventories are given. The United States Naval Reactor program was contacted and has confirmed the continued use of the projected inventories found in WSRC (2001) for the operational NRCDA in 643-26E.

The closure inventory estimates (Sink 2007) were developed using the radionuclide distribution received for each type of ELLWF disposal unit and the PA radionuclide limits developed for each disposal unit. Comparisons to waste volumes disposed to date vs. radionuclide inventories disposed to date for each unit were considered for averaging purposes to project closure inventories. In addition, key LLW Waste Acceptance Criteria (WAC) radionuclides were reviewed in detail for adjusting closure inventory estimates. Also, process knowledge of past disposals was used to evaluate key radionuclides where disposal campaigns have been completed for the site (i.e., M Area Glass waste forms, etc.). These methodologies produced upper-bound curie inventory estimates.

8.2.3.1.1.1 Components-in-Grout (CIG) Trenches

Components-in-Grout disposal units are below-grade earthen trenches, which have essentially vertical sides and contain grout encapsulated waste components. Two such CIG Trench units or footprints (157-foot-wide by 656-foot-long), designated CIG-1 and CIG- 2, are anticipated. Components to be disposed within the CIG Trenches consist of large radioactively contaminated equipment and other smaller waste forms such as B-25 boxes to fill the space around and above the large equipment. Components to date consist of tankers, radioactive sources in a concrete culvert filled with grout, SeaLands, B-25s, B-12s, flatbed trailers, tanks, high integrity containers, columns, etc. The waste components are surrounded by a minimum one foot of grout (WSRC 2008a).

The anticipated 2025 radionuclide inventory for the CIG Trenches is presented in Table A-13. The basis for this inventory is the use of a typical inventory (see Sink 2007) identified for a single CIG trench footprint that has been doubled to account for the presence of two CIG trench footprints. The aggregate inventory is what appears in Table A-13.

8.2.3.1.1.2 Engineered Trenches

The Engineered Trenches (ETs) are below-grade earthen, drive-in, disposal units. The excavated soil is stockpiled for later placement over disposed waste. Currently two ETs are located in the ELLWF. ET #1 is approximately 650 feet long by 150 feet wide and is designed to contain approximately 19,000 B-25 boxes of waste. ET #2 is approximately 656 feet long by 160 feet wide and is also designed to contain approximately 19,000 B-25 boxes of waste. ET #2 is essentially identical to ET #1 except that it does not contain a sump (WSRC 2008a); see Section 3.4.1.2.

The anticipated 2025 radionuclide inventory for the Engineered Trenches is presented in Table A-14. The basis for this inventory is the use of a typical inventory (see Sink 2007) identified for a single Engineered trench that has been doubled to account for the presence of two Engineered trenches within the ELLWF. The aggregate inventory is what appears in Table A-14.

8.2.3.1.1.3 Intermediate Level Vault

The IL Vault is a below-grade, reinforced concrete vault, consisting of two modules, which together encompass a 278.8-foot by 48.5-foot area. The Intermediate Level – Tritium (ILT) module contains two cells, with one cell (ILT Cell #1) containing 144, 20-inch diameter by 20-foot long vertical silos. The Intermediate Level – Non Tritium (ILNT) module contains seven identical cells. The area between the two modules provides manhole access to the subdrain system (WSRC 2008a).

The anticipated 2025 radionuclide inventory for the ILV is presented in Table A-15. The basis for this inventory is the inventory provided by Sink 2007 for the single ILV that the ELLWF is projected to contain.

8.2.3.1.1.4 Low Activity Waste Vaults

The LAW Vault is an above-grade, reinforced concrete vault, divided into three modules along its length, which contain four cells each. The modules share a common footer but have a 2-inch (5.08 cm) gap between their adjacent walls. The 12-cell total is designed to contain more than 12,000 B-25 boxes of waste (WSRC 2008a).

The anticipated 2025 radionuclide inventory for the LAW Vault is presented in Table A-16. The basis for this inventory is the inventory provided by Sink (2007) for the single LAW Vault that the ELLWF is projected to contain.

8.2.3.1.1.5 Naval Reactor Components Disposal Areas

NRCDAs are above-grade gravel pads for the disposal of shipping/disposal casks containing waste naval reactor (NR) components. Two NRCDAs are associated with the ELLWF. The 643-7E NRCDA contains 41 casks, is a trapezoidal area consisting of approximately 0.3 acres, and is closed to future receipts. It has an interim soil cover in place. The 643-26E NRCDA is currently in operation, is an irregularly shaped area consisting of approximately 1.4 acres, and is expected to receive up to 100 casks for disposal (WSRC 2008a).

The anticipated 2025 radionuclide inventory for the 643-7E NRCDA at closure is presented in Table A-17. The basis for this inventory is the closure inventory provided by Sink (2007).

The anticipated 2025 radionuclide inventory for the 643-26E NRCDA at closure is presented in Table A-18. The basis for this inventory is the projected inventory provided by Yu et al. 2001 (Table 2.4-2).

8.2.3.1.1.6 Slit Trenches (East, Central and West)

Slit Trenches are below-grade earthen disposal units with essentially vertical side slopes. The excavated soil is stockpiled for later placement over disposed waste. Individual slit trenches are generally 20 feet deep, 20 feet wide, and 656 feet long. Ten feet to 14 feet of undisturbed soil separates each trench. A set of five, 20-foot-wide slit trenches are generally grouped together within a 157-foot-wide by 656-foot-long standard Slit Trench footprint or unit. Eight such footprints, designated Slit 1 through 8, have been currently sited and waste has been placed within all eight units. The remaining Slit footprints have not yet been developed, but the planned trenches are accounted for in this inventory. (WSRC 2008a) For the purpose of establishing disposal limits, the Slit footprints have been divided into three groupings termed Slit East, Slit Central and Slit West. Slit East will include eight Slit units, which on an area basis constitute 7.66 standard Slit footprints. Slit Central will include seven Slit units, which on an area basis constitute seven standard Slit footprints. Slit West will include six Slit units, which on an area basis constitute 4.93 standard Slit footprints.

The basis for estimating the total closure inventory of each Slit grouping (East, Central, and West) is to utilize the anticipated "typical" Slit footprint (157-foot by 656-foot) inventory for that grouping (see Sink 2007) and multiply it by the number of equivalent standard Slit footprints within that grouping. The 2025 Slit Trench radionuclide inventories are presented in Table A-19, Table A-20, and Table A-21, for Slit Trench groupings East, Central, and West, respectively.

8.2.3.2 TRU Pad 1

8.2.3.2.1 Background

Currently the TRU waste stored on TRU Pad 1 is slated for shipment to the Waste Isolation Pilot Plant (WIPP) in Carlsbad, NM for disposal in accordance with the SRS End State Vision (DOE 2005). To aid in the determination as to whether this approach is appropriate, a PA, which assumes that the TRU Pad 1 waste is disposed in place at SRS, is currently in the early stages of development. For conservatism in this CA, the disposal of TRU Pad 1 in place at SRS will be taken as the base case. However since it is likely that the TRU waste stored on TRU Pad 1 will be removed from its current location the following two inventory sensitivities have been considered: 1) 100% of the inventory is shipped off site to WIPP; 2) 100% of the inventory is transferred to H-Canyon and HB Line[1].

TRU Pad 1 is an at-grade 12-inch thick reinforced concrete pad measuring approximately 150 feet long by 60 feet wide. Mound and Los Alamos Scientific Laboratory (LASL - now Los Alamos National Laboratory [LANL]) shipped containers of heat source scrap (Pu238) and contaminated equipment items to SRS between 1970 and 1972. These containers of TRU waste were overpacked in concrete containers on Pad 1, and the concrete overpack containers were buried under an interim soil cover. TRU Pad 1 is considered an operational facility, since the soil cover may be removed at a future date. Maintenance activities are conducted to maintain the soil cover (WSRC 2007c).

The plutonium is present on a variety of equipment (lathes, dissolvers, balances, presses, vessels, etc.) and material (steel, glass, paper, rags, plastic, resin, etc.), and also present in combinations of materials labeled as trash, miscellaneous residues and sweepings.

8.2.3.2.2 Inventory and Basis for Estimate

A summary of the radionuclide inventory of the plutonium containing Mound and LASL waste shipped to SRS during 1970 to 1972, the bulk of which is contained on TRU Pad 1, is provided in Table A-22. The inventory is taken from the inventory provided by Towler (1979) for the waste as of 1972. This inventory is considered conservative, because one box and two culverts of plutonium bearing waste included in the inventory have been shipped offsite.

[1] Currently, a portion of the soil covering the culverts on Pad 1 has been removed. About one-third of the culverts have been removed and some of the culverts have been opened and the drums have been prepared for shipment to WIPP. Plans are to ship as much of this material to WIPP as can be done safely. Any residual material will have to be disposed onsite.

8.2.3.3 Mixed Waste Management Facility (MWMF) and Low-Level Radioactive Waste Disposal Facility (LLRWDF)

8.2.3.3.1 Background

Disposal of low-level waste within the 119-acre Burial Ground Expansion (BGE) or 643-7E began in 1972 and ceased in 1995. A 58-acre portion of the BGE, designated the Mixed Waste Management Facility (MWMF) or 643-28E, was carved out of 643-7E and closed in 1990 after receiving mixed waste from 1972 to 1986. An additional 18 acres of the BGE (643-7E), designated as the Low Level Radioactive Waste Disposal Facility (LLRWDF), received waste from 1986 to 1995. Thirteen of the 18 acres were also determined to have received mixed waste from 1986 to 1990. These 13 acres were carved out of the LLRWDF, added to the MWMF (643-28E), and closed in 1998. The remaining five acres of the LLRWDF continued to receive low-level waste for disposal until 1995. The 1998 closure also covered these five acres, which received low-level waste for disposal between 1990 and 1995. The remaining 43 acres of the BGE (643-7E) are used to store TRU waste and mixed waste prior to shipment to off-site disposal facilities. Waste disposed within the BGE (643-7E) included job control waste and bulk waste (Bullington and Frye-O'Bryant 1993, ERDA 1977, WSRC 1991, WSRC 1999, Phifer and Serrato 2000, WSRC 2003a).

Closure of the original 58-acre MWMF (643-28E) consisted of waste trench dynamic compaction for physical stability and installation of a closure cap with a compacted clay (kaolin) hydraulic barrier (Phifer 1991, WSRC 1991, WSRC 2007a). Closure of the subsequent 18 acres of the LLRWDF consisted of waste trench dynamic compaction for physical stability and installation of a closure cap with a composite hydraulic barrier (i.e., geosynthetic clay liner [GCL] overlain by a high density polyethylene [HDPE] geomembrane) (Bullington and Frye-O'Bryant 1993, WSRC 1999, WSRC 2007a).

8.2.3.3.2 Inventory and Basis for Estimate

The 2002 radionuclide inventory for the LLRWDF (643-7E) and MWMF (643-28E) is presented in Table A-23. Two main sources of information were utilized to estimate the radionuclide inventory, the Environmental Restoration Division estimate documented in ERD (2001) and the SRNL estimate documented in WSRC (1997).

The remedial actions identified in Section 8.2.3.3.1 included the placement of a low-permeability closure cap over the facilities. In such situations the CA analysis does not assume that the closure cap lasts for the full period of analysis (1,000 yrs) hence it is possible that more radionuclides than are listed in ERD (2001) as COCs could impact the groundwater at concentrations greater than the MCL. Hence, the longer list of radionuclides for the LLRWDF/MWMF in the previous CA (WSRC 1997) were retained and utilized to project an SRS End State radionuclide inventory.

The strategy employed in this calculation was to compare the decay corrected radionuclide inventory reported in both ERD (2001) and WSRC (1997) for year 2002 and take the higher inventory as the current estimate for 2002. Because quite a few additional radionuclides are reported to be present in the LLRWDF/MWMF in the WSRC (1997) analysis, the inventories for those radionuclides were decay corrected to 2002 and included in the current estimate. In the ERD (2001) estimate the inventory for several isotopes of Pu and U were combined into a composite inventory, so the combined inventory was apportioned back to the individual isotopes according to the ratio of inventory of those isotopes identified in the previous CA (WSRC 1997).

8.2.3.4 Old Radioactive Waste Burial Ground (ORWBG) / General Separations Area Consolidation Unit (GSACU)

8.2.3.4.1 Background

The Old Radioactive Waste Burial Ground (ORWBG) is part of the GSACU, which consists of four primary waste units: H-Area Retention Basin (HRB) (281-3H), Warner's Pond (685-23G), HP-52 Ponds, and the ORWBG (643-E) and which also includes 22 underground storage tanks (Old Solvent Tanks [OSTs] 650-01E through 650-22E). Collectively, these waste units are identified as a single operable unit (OU) because of their proximity to each other and similar health and environmental threats.

The ORWBG consists of the following four distinct subunits:

- The ORWBG - a former disposal area for solid radioactive waste produced at the SRS and other DOE sites. Most waste was placed in the ORWBG from 1952 until 1972. Radioactive, hazardous, and mixed waste remain buried at depth in the ORWBG.
- Twenty-two OSTs - emptied in 1977 (by transferring the liquid solvent from the ORWBG to another facility), the OSTs were originally used to store spent plutonium-uranium extraction solvent from 1953 to 1977. Very little residual liquid and sludge remain in the OSTs.
- Mercury Hot Spot – a distinct area containing approximately 20% of the total mercury in the ORWBG. Each burial consists of two or three 1-liter polyethylene bottles filled with elemental mercury, double-bagged, and containerized in 5-gallon cans.
- Radioactive Hot Spots – multiple and distinct areas containing relatively high concentrations of radionuclides (>60 Ci per 20 foot x 20 foot grid cell). Generally these consist of tritium, transuranic isotopes, C14, and fission products such as Cs137 and Sr90.

The unit is listed as a RCRA 3004(u) Solid Waste Management Unit/CERCLA unit in Appendix C of the Federal Facility Agreement (FFA 1993) for the SRS. Final remedial requirements have been consolidated under one final ROD (WSRC 2000, WSRC 2007a).

The final remedial action for the ORWBG was installation of a closure cap with a composite hydraulic barrier (i.e., geosynthetic clay liner (GCL) overlain by a HDPE geomembrane) on top of the existing grade. A vegetative layer was added and the area seeded and compacted to prevent erosion. The mercury hot spot and the radioactive hot spots are addressed in the final ROD for the GSACU (WSRC 2002). The approved remedial action for the mercury hot spot and for the ORWBG as a whole is a low permeability cap and institutional controls. The approved remedial action for the long-lived persistent radioactive hot spots is the ORWBG low permeability cap (which has already been constructed) and institutional controls with individual intruder barriers that are to be installed at a future time before the institutional controls at the ORWBG are terminated. The likely intruder barrier will be a heavy rip-rap placed over the low permeability cap but beneath a soil cover.

8.2.3.4.2 Inventory and Basis for Estimate

Three main sources of information were utilized to estimate the radionuclide inventory for the ORWBG: an SRNL estimate documented in WSRC 1997, an estimate documented in WSRC (1997a), and an Environmental Restoration Division estimate documented in ERD (2001).

The remedial actions identified in Section 8.2.3.4.1 include the placement of a low-permeability closure cap over the ORWBG. In such situations the CA analysis does not assume that the closure cap lasts for the full period of analysis (1,000 yrs) hence it is possible that more radionuclides than are listed in ERD (2001) and WSRC (1997a) as COCs could impact the groundwater at concentrations greater than the MCL. Hence, the longer list of radionuclides listed for the ORWBG in the previous CA (WSRC 1997) were retained and utilized to project an SRS End State radionuclide inventory.

The strategy employed in this calculation was to compare the decay corrected radionuclide inventory reported in WSRC (1997), WSRC (1997a), and ERD (2001) for year 2002 and take the higher inventory as the current estimate for 2002. Because quite a few additional radionuclides are reported to be present in the ORWBG in the WSRC (1997) analysis, the inventories for those radionuclides were decay corrected to 2002 and included in the current estimate. In the ERD (2001) estimate the inventory for several isotopes of Pu and U were combined into a composite inventory, so the combined inventories were apportioned back to the individual isotopes according to the ratio of inventory of those isotopes identified in the previous CA (WSRC 1997).

The contaminated soil from the H-Retention Basin, Warner's Pond and the HP-52 Basin was excavated as part of the Remedial Actions for those facilities and the excavated soil was placed within the ORWBG and the closure cap was installed over the combined contamination. Within the regulatory documentation this combined contamination has been designated the General Separations Area Consolidated Unit (GSACU). A detailed calculation of the inventory associated with these soils is described in WSRC (2003c) and was re-verified as a part of this investigation. Table A-24 provides the combined 2002 radionuclide inventory for the ORWBG / GSACU.

8.2.4 F-Area Facilities

8.2.4.1 F-Area Materials Storage Facility (FAMS)

8.2.4.1.1 Background

The FAMS facility's primary mission is the receipt, storage, and disbursement of plutonium bearing materials to support both the SRS and the DOE complex. The Facility also supports specific missions for the repackaging of these materials. In addition to the original vault areas, additional areas within the facility have been modified to provide storage space. The only facilities that are currently operational are four vaults that provide storage of nuclear materials and repackaging of approved Rocky Flats Environmental Technology Site materials in the Designated Repackaging Areas in support of various SRS missions. FAMS also houses the Plutonium Fuel Form (PuFF) Facility, an (old) metallography glovebox line that was part of the PuFF Facility, and the Plutonium Experimental Facility (PEF). These facilities were once used to fabricate heat sources from Pu238 oxide powder for space program applications. A new Metallography Laboratory (New Met Lab) was constructed to support operations in the PuFF Facility and PEF but was never started up. The New Met Lab has been modified to support miscellaneous operations and was renamed the Material Transfer Room.

8.2.4.1.2 Inventory and Basis for Estimate

The inventory estimate developed for FAMS was based on the inventories analyzed in the 2003 Safety Analysis Report (SAR) (WSRC 2003b). An End State Reduction Factor of 1E-2 has been applied to approximate the fraction of the facility inventory that might remain with an End State configuration of "In-Situ Disposal" (Table 4.3b, Appendix J, DOE 2005).

The SAR uses residual inventories for each cell within the FAMS building. The grams of residual PuO_2 for each cell given on pages 9.1-7 and 9.1-8 of the SAR were multiplied by the Ci/g values given in Table 9.1-2 to give the curie quantities of each radionuclide in each cell. Residual NpO_2 contamination was taken from p. 9-1-8 of the SAR and multiplied by the Ci/g factors in Table 9.1-3 to give residual curies of Np237 and Pu238. The materials within FAMS are the products of refining processes carried out elsewhere. These materials should only contain the radionuclides given in WSRC (2003b). The estimates for all of the cells were summed to give building totals. The inventory was assumed to be for the year of publication of the SAR, 2003. The resulting 2003 radionuclide inventory for FAMS is provided in Table A-25.

8.2.4.2 F-Area Inactive Process Sewer Lines

8.2.4.2.1 Background

The F-Area Inactive Process Sewer Line (FIPSL) extends from Building 221-F to the F-Area Seepage Basins and consists of various materials and diameters. Vitrified clay lines were utilized from 1955 to 1982 to transport hazardous and low-level radioactive wastes. Due to deterioration and leakage, the vitrified clay lines were abandoned in place and replaced with a set of HDPE lines (WSRC 1993).

The FIPSL closure requirements were added to the modification of the RCRA permit application for the F-Area HWMF (904-41G, -42G, -43G) in 2005. The selected remedy is to grout manholes and treblers in place and to install a geosynthetic cover over the length of the vitrified clay pipe and around the perimeters of the manholes and treblers associated with the HDPE lines (WSRC 2007a). This action will be completed as a part of the closure of the F-Area Tank Farm and Separations facility to avoid impacting underground and overhead interferences necessary to ensure safe operation of the F-Area facilities.

8.2.4.2.2 Inventory and Basis for Estimate

Two estimates of the radionuclide inventory in the FIPSL were utilized to determine the inventory to utilize in this CA. The most recent estimate of radionuclide inventory originates in ERD (2001), which is an SGCP inventory of the COCs for their closure units. These COCs represent only those expected to impact groundwater at concentrations greater than the MCL; hence the additional radionuclides identified and quantified in the previous CA (WSRC 1997) were included in this estimate. These estimates are based on the results of the maximum inventories for each radionuclide encountered in the soil characterization program for FIPSL and estimating the mass of soil material potentially contaminated by leakage. The base radionuclide inventory is identified for 2002, which is the date of the ERD (2001) inventory (December 2001) and the radionuclide inventory retained from the previous CA was conservatively assumed to reflect this date.

This analysis is for the entire length of the FIPSL, including segments located from Bldg. 221-F to the perimeter fence as well as the length from the perimeter fence to the F-Area Seepage Basins. The 2002 radionuclide inventory for the FIPSL provided in Table A-26 reflects the full length of the FIPSL.

8.2.4.3 F-Canyon and FB Line

8.2.4.3.1 Background

Currently, F-Canyon and FB Line have been deactivated to a state specified by DOE and are considered long-term Surveillance and Maintenance (S&M). Deactivation involved placing the facility in a safe and stable condition, including the removal of readily removable hazardous and radioactive materials. S&M consists of activities necessary to maintain systems and equipment essential for decommissioning while ensuring overall safety of the worker, public and environment until further disposition actions are taken to reduce residual risks from the facility.

Since 1995, F-Canyon and FB Line had been operating to stabilize "at risk" nuclear materials and spent fuel from throughout the DOE complex. The F-Canyon production mission ended in March 2002 with the last Plutonium-Uranium Extraction (PUREX) campaign. FB-Line continued with an operations mission, which was to characterize plutonium materials, stabilize them by packaging into 3013 containers, and ship the material to K-Area Material Storage. The FB Line production mission concluded in February 2005 with deinventory of all plutonium packages.

The following facility descriptions and estimated inventory were obtained from WSRC 2007.

8.2.4.3.1.1 F-Canyon

F-Canyon was a radiochemical processing plant that used the PUREX process. The F-Canyon structure is approximately 816 feet long and 122 feet wide. The primary operations conducted in F-Canyon included the separation and recovery of Pu239 and U238 from irradiated materials and the stabilization of plutonium residues. The process equipment is located in two parallel canyons, a "hot" and a "warm" canyon. The more highly radioactive processing operations, such as dissolution of irradiated materials, precipitation and centrifugation, fission product separation, and the associated waste evaporation operations, were performed in the "hot" canyon. The final purification of the product streams was performed in the "warm" canyon. Equipment and fuel were transported and manipulated by means of remote cranes.

One of the by-product streams from F-Canyon was a uranyl nitrate solution. The function of the FA-Line facility was to convert the uranyl nitrate solution to uranium trioxide (UO_3) powder. Conversion to UO_3 powder was halted in the early 1990s and uranyl nitrate solutions stored in FA-Line have been removed from F-Area. Outside Facilities F-Area (OF-F) provided general support for F-Canyon and FB-Line operations. This included chemical storage and transfer operations, cold chemical preparation operations, acid recovery operations, and waste concentration operations.

8.2.4.3.1.2 FB-Line

FB-Line is a reinforced concrete structure and was used to convert plutonium nitrate, recovered from irradiated natural and depleted uranium in F-Canyon, to plutonium metal. FB-Line is located in the F-Canyon Building in Sections 1 through 5, on the third through sixth levels. FB-Line operations utilized a portion of Section 1 on the second level.

The FB-Line process included purification and concentration of plutonium by cation exchange, precipitation of plutonium as a tri-fluoride, recovery of the tri-fluoride by filtration, conversion of the tri-fluoride to PuO_2/PuF_4, and reduction with calcium metal to form plutonium metal buttons. A material characterization process was located on the south end of sixth level and material packaging activities were located on the third and fifth levels. Other non-PUREX FB-Line processes included Mechanical Line, Material Characterization, Bagless Transfer, vault activities including stored material surveillance programs, solid waste handling, and stabilization and storage projects.

8.2.4.3.2 <u>Inventory and Basis for Estimate</u>

The inventory estimates developed for the F-Canyon and FB Line were based on the inventories analyzed in the 2007 SAR (WSRC 2007). This inventory was based on the estimated holdup in the entire F-Canyon Complex, which included F-Canyon, FB Line, and the 294-F/294-1F Sand Filters. An End State Reduction Factor of one has been applied because the complex has undergone deactivation, and it is possible that no further inventory reduction will occur before facility closure. An End State configuration of "In-Situ Disposal" (Table 4.3b, Appendix J, DOE 2005) is assumed.

Table 8.3-1 and the accompanying note from WSRC (2007) were used to calculate the residual inventory of F Canyon due to holdup of 60 kg of Pu239. Table 8.1-2 from WSRC (2007) was used to give the residual inventory of the F Canyon process vessels due to a holdup of 450 g of Pu239. The information in Table 8.1-3 from WSRC (2007) was used to calculate the residual inventory in the FB-Line assuming that the holdup of 21.4 kg of Pu239 was both Mk42 Target Oxide and Specification Grade Pu. Material in the Uranium Oxide Storage Building was assumed to be completely removed prior to facility closure. Radionuclide activity ratios from Apperson (1983) were used to estimate inventories for those radionuclides important for CA purpose that are not significant to SAR analyses, i.e., long-lived mobile radionuclides such as C14, Tc99 and I129 for the F Canyon holdup. This information was combined to give the total Ci of each radionuclide. The resulting 2007 radionuclide inventories are presented for F-Canyon, F-Canyon Process vessels, and FB Line in Table A-27, Table A-28, and Table A-29, respectively.

8.2.4.4 Canyon Auxiliary Facilities (Building 211-F)

8.2.4.4.1 Background

Canyon Auxiliary Facilities (Building 211-F) is divided into two sections, the Waste Handling Vault (WHV) and the Outside Section (OS). The former houses Tanks 800, 801, 804, 808, and 809, and contains a sump. The remaining portion of Building 211-F constitutes the OS. The text below describes equipment present during facility operations, the majority of which was removed during facility deactivation. The following facility descriptions and estimated inventory were obtained from WSRC (2008).

8.2.4.4.1.1 Waste Handling Vault

The WHV section, often referred to as the 800-series underground tanks, consists of five underground tanks housed in a concrete structure 48 feet wide by 60.5 feet long by 34 feet deep. The vault consists of six cells with removable covers. The concrete floor of the vault is sloped toward an unlined sump, which is located in the sixth cell of the vault. The sump contents were pumped overhead; there were no below grade external lines from the sump. The vault and sump are considered radiologically contaminated.

The primary function of the five tanks of the WHV section was to receive Low-Activity Waste (LAW) and High-Activity Waste (HAW). The designations LAW and HAW were assigned by the SRS laboratories to separate them, based on radionuclide content, to facilitate from which side of the canyon (warm or hot) to recover potential materials.

Flushing of Tanks 800 and 801 achieved the deactivation end-points of less than five grams of plutonium, less than 0.2 parts per million (ppm) of mercury, and less than five ppm of lead, and pH greater than two but less than 12.5. For Tanks 804, 808, and 809 flushing did not achieve these deactivation endpoints. SRNL analytical results of tank contents indicate that the total plutonium in Tank 804 is estimated to be 1,530 grams (591 liters); the total plutonium in Tank 808 is estimated to be 317 grams (510 liters); and the total plutonium in Tank 809 is estimated to be 92.4 grams (283 liters). Deactivation will continue for Tank 804 to reduce the plutonium content to below 390 grams to address potential nuclear criticality concerns and may involve removal and disposition of the entire tank. A synthetic coating was applied to the floor of the 804 cell for worker protection during deactivation activities. The sludge density is 1.17 g/mL in Tank 804, 1.52 g/mL in Tank 808, and 1.42 g/mL in Tank 809 (WSRC 2008).

8.2.4.4.1.2 Outside Section

The facilities making up the Outside Section (OS) include the Chemical Storage Facility, Water Handling Facility, Acid Recovery Unit (ARU), General Purpose (GP) Evaporators, GP Waste Tanks, Segregated Solvent Facility, Tank 805 Cell, Tank 820 Cell, and recycle sump. Vessels, sumps, and equipment in the OS were stationed in dike areas or on concrete aprons. The aprons served as dikes because the sides are elevated and act as a containment feature. The contents of the sumps in the OS were pumped overhead to various locations. There were no underground discharges from the OS sumps.

The concrete slabs within the OS, except for the Chemical Storage Facility and a portion of Segregated Solvent are covered with stainless steel plating. The Chemical Storage Facility is divided into three sections. The OS of the Chemical Storage Facility are not lined, but the middle section is lined with a coating material. The aprons and liners of the OS of the 211–F Auxiliary Facilities are not part of the original construction, but were part of modifications made during the mid-1980s (WSRC 2006c).

The Chemical Storage Facility is a nonradiological facility that contained 11 above-ground tanks that were removed during deactivation. The tanks received, stored, and dispensed nitric acid, tributyl phosphate, n-paraffin, aluminum nitrate, and sodium hydroxide for use in the F-Canyon and 211-F processes. No radiological inventory is attributed to the Chemical Storage Facility (WSRC 2006c).

The Water Handling Facility, also known as the 500 apron, is a radiologically contaminated area that consisted of two skimmers, a skimmed solvent hold tank, two acidified water storage tanks, and two wastewater storage tanks that were removed during deactivation. The facilities received demineralized water from the F-Area Deionized Water System (284-2F), prepared acidified water for the F-Canyon process, and received rainwater and acidic-wash solution from other sources. The solvent and aqueous solutions processed in the Water Handling Facility were radiologically contaminated. The remaining slab is 104 feet by 43 feet (4472 square feet).

The ARU, also known as the 600 apron, included the reduced-pressure fractional distillation column, with a reboiler and associated feed and storage tanks that are being removed during deactivation. The ARU is radiologically contaminated. It received nitric acid from about five to nominally 50 percent concentration for reuse. The dilute acid came from the overheads of acidic evaporation of various F-Canyon process solutions. ARU overheads were transferred to the Effluent Treatment Plant (ETP). The equipment has been removed and the remaining apron covers approximately 2580 square feet.

The GP Evaporators concentrated low-activity radioactive alkaline aqueous materials. The principal components of the GP Evaporators (700 apron) included two evaporators, preheaters, and associated feed tanks, hold tanks, and storage tanks that are being removed during deactivation. The GP Evaporators are radiologically contaminated. The evaporator feed tanks 701-1 and 701-2 were used to neutralize LAW transferred from Tanks 812, 813, and 802, and Evaporators 703-1 and 703-2 evaporated LAW after neutralization. Bottoms Holding Tank 710 received evaporator bottoms from the two GP Evaporators and further neutralized (pH-adjusted) the liquid before feeding it to 241-F. The GP Evaporators operated under reduced pressure as a flash evaporator with forced-bottoms circulation. The remaining concrete pad covers 1870 square feet.

The GP Waste Tanks, (800 apron), consist of eight above ground tanks that were removed during deactivation. The GP Waste Tanks, having received LAW from sumps and evaporator overheads, washed spent solvent, and other miscellaneous sources, are radiologically contaminated. The remaining concrete pad covers 4472 square feet.

The Segregated Solvent Facility, a radiological facility, is an open 46 foot by 60 foot (2,760 square feet) concrete pad that contained six above ground tanks that were removed during deactivation. The 901 Tank (sodium carbonate wash tank) and 902 Tank (wash hold tank) are shielded by a concrete wall. Tanks 903 and 905 were nitric-acid wash tanks; tanks 904 and 906 were solvent tanks. In addition to the six tanks, the Segregated Solvent Facility contained two sumps: the decant sump and a floor drain sump. They removed degradation products and radioactive contaminants from spent solvent, neutralized alkalinity from entrained carbonate washes, and returned the treated solvent to the extraction process. The Tank 805 Cell consists of a stainless steel below ground tank and sump housed within a 20 foot by 11 foot by 26 foot deep concrete vault with a removable sealed cover. The concrete vault, located in a radiologically contaminated area, is lined with stainless steel as part of its original construction. The tank cell, also referred to as the sump collection tank and sump, provided secondary containment and collected LAW liquid from the 294 northeast sand-filter sump, canyon, and building air tunnel sumps and the 291–F stack catch tanks (602). No radiological inventory is attributed to the Tank 805 Cell (WSRC 2006c).

The Tank 820 Cell, or hot canyon sump collection tank, is physically located underground within a 16 foot by 16 foot by 19 foot-deep stainless steel lined (part of the original construction) concrete vault, and has a removable sealed cover. The tank cell provided secondary containment for radiologically and chemically contaminated material collected from the hot canyon sumps. The tank cell has been used primarily to collect rainwater in-leakage since 1980. No radiological inventory is attributed to the Tank 820 Cell (WSRC 2006c).

The recycle sump is a shielded, stainless steel lined, underground, 20,000 gallon concrete sump constructed in the early 1950s. The stainless steel liner in the sump is part of its original construction. The sump, sealed with a 16 inch thick reinforced slab, is in a radiologically contaminated area; it received the chemical (acids, bases, and organics), solvent, and radioactive liquid drains and overflows from various sources in the OS.

8.2.4.4.2 Inventory and Basis for Estimate

For each of the facilities evaluated in this report, several samples of the remaining materials (i.e., concrete slab) were analyzed to verify that the decommissioning and decontamination activities were complete. Slab remnants were divided into EAs and each EA was evaluated to determine the total curies of each detectable radionuclide based on the area of the EA and assuming the results apply to the top six inches of the slab for tritium and the top two inches for other radionuclides. The 2006 radionuclide inventories for individual facilities that constitute the F-Canyon Auxiliary Facilities (Building 211-F) are presented individually in Table A-30 and are also totaled in the right hand column. Inventories were originally reported for several combined isotopes, Cm243/244, Pu239/244, U233/234 and U235/236. These aggregated isotopes are listed separately in Table A-30. Because no other information was available to apportion the inventory between the two isotopes, the listed aggregate inventory was conservatively assigned to each individual isotope.

8.2.4.5 F-Area Retention Basin 281-3F

8.2.4.5.1 Background

The F-Area Retention Basin (FRB) (281-3F) was designed and constructed as an unlined, temporary basin for potentially contaminated cooling water from the chemical separations process and storm sewer drainage from the FTF (Scott et al. 1987). The basin measures 61 m (200 feet) long and 36.6 m (120 feet) wide and 2.1 m (6.9 feet) deep, with a total volume capacity of approximately 4688 m^3 (6132 yd^3) (Scott et al. 1987). Operation began in 1955 and the basin remained active until 1972. It was replaced by a lined retention basin (281 -8F) located west of the unit. In December 1978, the FRB was excavated, backfilled with dirt, and covered with grass. Excavated soils were transported to the Burial Grounds (643 -7 G). A process sewer line that extended from the basin to a diversion box approximately 152 m (500 feet) to the north was abandoned when the basin was backfilled. The FRB is located outside and south of the F-Area perimeter fence and east of Bldg. No. 281-8F. It is located approximately 3937 feet from Fourmile Branch.

Between 1955 and 1973, the FRB received aqueous streams of process cooling water and storm water from the chemical separations area via a process sewer line. The process water was used in the cooling coils that surround process vessels that contained radioactive materials. Periodically, the cooling coils would leak and entrain radioactivity from these process vessels. The process streams were monitored and, when elevated radioactivity was detected, process flows (normally discharged to the F-Area Seepage Basins) were diverted to the retention basins. This diverted discharge was intermittent and variable in its content of radioactivity (WSRC 2007a).

The remedial actions selected in the ROD for the basin soil at this OU are identified in WSRC 2007 and include in situ stabilization, a low permeability soil cover, institutional controls, and groundwater monitoring. The remedial actions selected in the ROD for the process sewer line soil at this OU include pipeline grouting and soil excavation and disposal in the basin.

8.2.4.5.2 Inventory and Basis for Estimate

Two estimates of the radionuclide inventory in the FRB (281-3F) were utilized to determine the inventory to utilize in this CA. The most recent estimate of radionuclide inventory originates in ERD (2001), which is an SGCP inventory of the COCs for their closure units. These COCs represent only those expected to impact groundwater at concentrations greater than the MCL when the remedial actions (low-permeability closure cap and in-situ stabilization of soil) are assumed to retain their properties in perpetuity. Hence, the additional radionuclides identified and quantified in the previous CA (WSRC 1997) were included in this estimate because that assumption does not apply for this CA. The inventory estimate for each radionuclide is based on the maximum inventory, reported in pico Curies per gram (pCi/g), determined amongst all samples collected in the soil characterization program, estimating the mass of soil material potentially contaminated by leakage, and performing the associated multiplications. In the case where inventories for certain radionuclides are reported in both the previous CA and in ERD (2001), the highest inventory from either source was assumed to be the appropriate inventory in this CA.

The base radionuclide inventory is identified for 2002, which is the date of the ERD (2001) inventory (December 2001) and the radionuclide inventory retained from the previous CA were conservatively assumed to reflect this date. The 2002 radionuclide inventory for FRB 281-3F is presented in Table A-31.

8.2.4.6 F-Area Seepage Basins

8.2.4.6.1 Background

The F-Area Seepage Basins (904-41G, -42G, -43G) are part of the F-Area HWMF Operable Unit. The F-Area HWMF OU is located in the central portion of SRS, approximately five miles from the nearest site boundary and operated from 1955 until November 7, 1988. During that time, the facility received waste effluents from F-Area chemical separation facilities processes such as the nitric acid recovery unit, waste storage system evaporator overheads, and general-purpose evaporator overheads. The facility consists of three basins with a combined maximum operating capacity of 20.5 million gallons of wastewater.

Significant amounts of nitrate and caustic were received in the basins. Radioactive releases were greater than 99 percent tritium. A 1984 soil coring study showed that approximately 90 percent of the radionuclides (other than tritium), cations, and anions were concentrated within the top one foot of basin soil.

These basins were closed by dewatering, physically and chemically stabilizing the remaining sludges, and placing a protective multi-layer compacted clay cover system over them to reduce rainwater contact with basin bottoms. A RCRA Part B permit application for post-closure care was submitted in December 1990, and a hazardous waste permit was issued effective November 1992.

8.2.4.6.2 Inventory and Basis for Estimate

Two main sources of information were utilized to estimate the radionuclide inventories for the F-Area HWMF OU, the Environmental Restoration Division estimate documented in ERD (2001) and the SRNL estimate documented in WSRC (1997).

The remedial actions for the F-Area HWMF OU identified in WSRC (2007) identify the placement of a low-permeability closure cap over the facility and the in-situ chemical stabilization through the use of limestone and blast furnace slag layers and in-situ physical stabilization through the use of granite rip-rap. The CA analysis does not assume the closure cap and the soil-grout integrity last for the full period of analysis (1,000 yrs) hence it is possible that more radionuclides than are listed in ERD (2001) as COCs could impact the groundwater at concentrations greater than the MCL. Hence, the longer list of radionuclides for the F-Area HWMF OU in the previous CA (WSRC 1997) were retained and utilized to project an SRS End State radionuclide inventory.

The strategy employed in this calculation was to compare the inventories of radionuclides reported in both ERD (2001) and WSRC (1997) and take the higher inventory as the base estimate for 2002. Because quite a few additional radionuclides are reported to be present in the F-Area HWMF OU in the WSRC (1997) analysis, the inventory for those radionuclides was included in the current estimate and conservatively assumed to be valid for 2002. In the ERD (2001) estimate the inventory for several isotopes of Pu and U were combined into a composite inventory. In the situation where the combined inventory exceeded the combined inventory for the same isotopes reported in WSRC (1997), the ERD (2001) estimate was adopted and the composite inventories were apportioned back to the individual isotopes according to the ratio defined in the previous CA (WSRC 1997).

The base radionuclide inventory is identified for 2002, which is the date of the ERD (2001) inventory (December 2001) and the radionuclide inventory retained from the previous CA was conservatively assumed to reflect this date. The 2002 radionuclide inventory for the F-Area Seepage Basins is presented in Table A-32.

8.2.4.7 F-Area Tank Farm and Ancillary Equipment

8.2.4.7.1 Background

The F-Area is in the north-central portion of the SRS and occupies 364 acres. The FTF is an active waste storage facility consisting of 22 carbon steel underground waste tanks (Tank Numbers 1-8, 17-20, 25-28, 33, 34, and 44-47) that store, or once stored liquid radioactive waste generated primarily from the F-Canyon PUREX process. Tank 17 and Tank 20 have already been filled with grout and closed via a SCDHEC and EPA reviewed and approved Closure Plan and Closure Modules (WSRC 2008b).

The FTF waste tanks were built during four separate construction periods, with a different tank design for each period, leading to the designation of the following four different tank groups:

- Tanks 1 through 8 are Type I tanks and were constructed between 1951 and 1953
- Tanks 17 through 20 are Type IV tanks and were constructed between 1956 and 1958
- Tanks 33 and 34 are Type III tanks and were constructed between 1969 and 1972
- Tanks 25 through 28 and 44 through 47 are Type IIIA tanks and were constructed in two phases between 1975 and 1978, and 1977 and 1980, respectively

The End State configuration for each of the 22 underground waste tanks in the FTF includes cleaning, filling with grout to stabilize residual material, and closing in place.

The FTF contains ancillary equipment with a residual radiological inventory that must be accounted for as a part of facility closure. This ancillary equipment includes buried pipe (transfer lines), pump tanks, and evaporators, all of which have been in contact with liquid waste over the operating life of the facility. The amount of contamination on these components depends on such factors as the service life of the component, its materials of construction, and the contaminating medium in contact with the component.

Ancillary equipment including the evaporator buildings, the catch tank, pump pits (PPs), and diversion boxes (DBs) to the extent practical will remain in place and be grouted, as practical, to eliminate subsidence potential. An exception to leaving ancillary equipment in place will be made for equipment that is significantly higher in elevation than the adjacent waste tanks and would therefore result in a significant increase in the closure cap elevation. Above grade structures, utilities, equipment, etc., (other than substantial above grade concrete associated with the tanks and ancillary equipment) that could interfere with closure cap construction will be removed from the FTF area prior to installation of the closure cap.

Also included in this section is the ETP lift station location next to FTF. Routine low-level radioactive wastewater generated in the F-Tank Farm is collected at the lift station and pumped to the treatment plant in H-Area. The wastewater is then processed through the treatment plant and pumped to Upper Three Runs Creek for discharge at a permitted outfall.

8.2.4.7.2 Inventory and Basis for Estimate

8.2.4.7.2.1 FTF Tanks

The basis for the radionuclide inventory for the F-Area radioactive liquid waste storage tanks is summarized as follows. For all tanks except Tanks 17 through 20, the residual curie inventory is decayed to 9/30/2020, and is based on an average height of 0.0625 inch of settled solids on the tank bottom based on experience with cleaning Tank 16 using Oxalic Acid (OA), a cleaning process planned for use on the remaining tanks. The average height of 0.0625 inches of settled solids is used to develop an estimated residual solids volume which is used to calculate the inventories. However, 0.0625 inch is not a required or target value, because it is the curie or kilogram value that is important to the CA. The total radionuclide and non-radiological constituent inventories in the 18 remaining FTF waste tanks are estimated by multiplying the estimated residual solids volumes by the concentrations currently in the Waste Characterization System (WCS) without credit for preferential removal of radiological or non-radiological constituents during the OA cleaning process. This technique may over estimate concentrations/inventory for highly soluble radionuclides (e.g., $Tc99^{+7}$), and could underestimate radionuclide concentrations/inventory for other radionuclides (e.g., actinides) that are not removed as effectively by OA. Tank 5 sludge sample analysis results in WSRC (2007d) indicate that Sr and Pu were not concentrated by OA at 50°C, while other contaminant concentrations were indeterminate because values were less than detectable.

The Tank 17 and 20 inventories are based on measurements of residual solids developed during tank closure activities as described in the respective characterization reports. Sample analysis results were used for Tank 17 and 20 inventories when available. In addition, this inventory has been decayed to 1/1/2020. (WSRC 1997b, WSRC 1996a, WSRC 2004c)

The Tank 18 and 19 inventories are based on a similar process, as described in the respective characterization reports. (SRS 2003, WSRC 2002a) Volume inventories in Tanks 18 and 19 have been reduced by 75% to reflect removal plans. Additionally, the radiological inventory assigned to the corrosion products on the walls of Tanks 18 and 19 has been adjusted based on revised estimates of the weight of accumulated corrosion products and a distribution (or sorption or partition) coefficient (K_d) for Uranium isotopes. (Wiersma 2007, Wiersma 2007a, WSRC 2007e) The total FTF radionuclide inventory is estimated by adding the estimates for the 18 tanks not yet cleaned to the measured radionuclide inventories for Tanks 17 through 20.

The anticipated 2020 radionuclide inventory for all F-Area High Level Tanks in their closure configuration is presented in Table A-33. Closure of the FTF is regarded to be attained in the year 2020.

8.2.4.7.2.2 FTF Ancillary Equipment

For the FTF ancillary equipment, it is assumed that over the operating life of the facility radioactive waste comes in physical contact with some of the ancillary equipment, contaminating them and hence, leaving contamination on the components. The degree of contamination depends on many factors, which include, but are not limited to, the service life of the component, the material of construction, and the type of waste in contact with the component. Some of the listed equipment serves only as secondary containment and may not have contacted the waste.

The rationale for estimating the radionuclide inventory for each of the ancillary equipment components is summarized as follows. The basis was to estimate what the residual quantity of process material at closure would be and to then utilize an average radionuclide distribution and activity for that residue in order to calculate the inventory.

The inventory of each tank was used to establish the characterization of the contamination medium estimated by inventory tracking in the SRS WCS and decaying the inventory to the expected date of closure, 9/30/2020 (Tran 2007). The results of a review of waste transfers within FTF and between FTF and HTF have been sorted to determine the percentage of the volume of all waste transfers that can be attributed to each FTF waste tank. The representative concentration was then determined by applying a weighted average to each isotopic distribution in the FTF tanks. Because the characterization of dry sludge was used for each tank for conservatism, it is assumed that the sludge would have to be converted to a slurry in order to remove it from the tanks through the ancillary equipment.

It is important to note that, while the sludge concentrations are used, dry sludge is only a small portion of the total waste that passes through the transfer lines that are routinely flushed with a high volume of supernate. Using the dry sludge concentrations provides a conservative representation of the actinides and long-lived isotopes. The Technical Safety Requirements (TSR) limit waste transfers to less than 16.7 wt% solids. (WSRC 2008c) Therefore, the slurry concentration is reduced to 20% of the dry sludge concentration.

Additional information utilized to estimate the radionuclide inventory associated with individual items of FTF ancillary equipment are provided below.

Transfer Lines

The amount of residual material in the piping systems was determined analytically (WSRC 2005). This includes half of the inter-area transfer line between F and H areas. The methodology in the referenced document is used for transfer lines, but with different concentrations. Waste in contact with piping systems adheres to the pipe in three ways: diffusion into metal, into an oxide film and as residue left behind after a transfer and flush. Diffusion calculations assume a 100-year contact time and a 100°C exposure temperature.

Pump Tanks and the FTF Catch Tank
Pump tanks and the FTF catch tank differ from piping systems with respect to such features as geometry and usage. Only residue left behind after rinsing and flushing is considered for these components. After tanks are cleaned and inspected, it is expected that residual inventory will be very low. It is assumed that 1/16 inch of residual material will remain in these vessels, consistent with the waste tank residual depth.

Evaporator Pot and Concentrate Transfer System (CTS) Tank and Other Ancillary Equipment
Field characterization data for the FTF evaporators and the 242-3F Concentrate Transfer System (CTS) pump tank were used to estimate the residual material in each evaporator and in the CTS. While future cleaning is anticipated for the CTS pump tank, the present inventory is assumed for PA modeling.

ETP Lift Station
The underground lift station is assumed to contain contamination. Recent measurements of solids contained within the lift stations produced alpha, beta/gamma and I129 levels. The beta/gamma levels were set equal to the Sr90 concentration and the other radionuclides (except I129 which was measured) were set equal to the ratio found in the FTF and HTF inventories.

No significant inventory is assumed for the remaining ancillary equipment (e.g., pump pits, DBs, valve boxes, transfer line jackets). The remaining ancillary equipment is not intended for waste storage and is providing a secondary containment function.

The 2020 radionuclide inventory for the combined components that comprise the FTF ancillary equipment is presented in Table A-34. The inventory is presented for the anticipated closure date of 2020.

8.2.4.8 Central Laboratory Facility (F-Area)

8.2.4.8.1 Background
The Central Laboratory Facility (CLAB) consists of Building 772-F, which was placed in service in the mid-1950s, Building 772-1F, which began operations in 1987, and Building 772-4F which was placed in service in July 1993.

The primary mission of the CLAB over the last 45 years has been to support the chemical separations processing activities at Building 221-F and H-Canyon. Samples received from the canyons and other site areas are subjected to the required radiological and chemical analysis tests. The projected future use of the facility is to continue its mission to support the separations processes and to provide support for the increasing waste management, waste characterization, waste stabilization, and environmental remediation activities at SRS.

The Airborne Radiation Removal Facility (Building 772-4F) represents a major addition to the main exhaust system of Building 772-F, housing two stages of High Efficiency Particulate Air (HEPA) filters, the associated main exhaust fans, and a 190-foot stack for the main exhaust system in Building 772-F. Overall floor space is approximately 7,600 square feet.

Building 772-1F is a relatively new structure designed to house a portion of the analytical operations previously performed in Building 772-F. It was built in the mid-1980s as a standard commercial structure and houses approximately 15 laboratory modules, four shielded cells, a Control Room (CR), and other ancillary facilities. Overall floor space is approximately 31,400 square feet. A 135-foot stack is used to discharge the building exhaust after it is filtered through HEPA filters.

8.2.4.8.2 Inventory and Basis for Estimate

The inventory estimates developed for the CLAB were based on the inventories analyzed in the 2006 Safety Analysis Report (WSRC 2006d). An End State Reduction Factor of 1E-3 has been applied to approximate the fraction of the facility inventory that might remain with an End State configuration of ”Demolish” (Table 4.3b, Appendix J, DOE 2005). Tables 3.3-3 and 3.3-4 in the SAR give inventory limits for 772-F and 772-1F, respectively, in a number of categories. For the “total alpha” category, the isotopic distribution for alpha emitters with respect to Pu239 given in Apperson (1983) was set up in a spreadsheet and the Pu239 quantity was varied until the total alpha activity equaled the amount in the table (310 Ci for 772-F and 420 Ci for 772-1F). The same methodology was used for beta-gamma emitters using Apperson’s isotopic distribution of beta-gamma emitters with respect to Cs137 and adjusting the Cs137 quantity until the total beta-gamma amount, excluding tritium and Pu241, was found (2,600 Ci for 772-F and 210 Ci for 772-1F). The tritium and Pu241 quantities are given in the tables. The tables also give the total mass of fissile material allowed in each building (2000 g Pu239 for both 772-F and 772-1F). A table of fissile gram equivalent conversion factors (Table B-1 of Hanford Waste Acceptance Criteria, HNF-EP-0063, Rev 114; Hanford undated) and specific activities from Cook (2007) were used to calculate criticality-based inventory limits for each facility. In cases where two inventories were calculated for a radionuclide, the larger was used. Table 3.3-6 (of the SAR) gives an estimated inventory for 772-4F. Additional radionuclides of interest to the CA were estimated using distributions taken from Apperson (1983). CLAB is an operating facility, which is not slated for closure until the assumed SRS end state date of 2025. Therefore the end state inventory estimated from the 2006 SAR is assumed to occur in 2025 at the buildings’ end state. The 2025 radionuclide inventory for the Central Laboratory Facility (Buildings 772-F, 772-1F, and 772-4F) is provided in Table A-35.

8.2.4.9 Old F-Area Seepage Basin (OFASB)

8.2.4.9.1 Background

The Old F-Area Seepage Basin (OFASB) is located approximately 600 feet north of F Area and approximately six miles from the nearest SRS boundary. The OFASB was an unlined seepage basin (approximately 300 feet long by 200 feet wide and 13 feet deep) that received 9- to 14-million gallons of low-radioactivity wastewater between November 1954 and mid-May 1955. The basin took waste water from the first hot runs of the F-Area Canyon Complex. Wastewater included overhead condensates from evaporates, laundry wastewater, non-reactor cooling water from F and H Areas, and possibly other chemicals. After 1955, the OFASB received occasional discharges of cooling water and rainfall runoff. During a three-month period in 1969, spent nitric acid solutions used to etch depleted uranium were discharged (via tanker truck) to the basin. It is estimated that over one metric ton of dissolved depleted uranium was discharged into the OFASB.

Wastewater disposal was discontinued after the 1969 discharge. An estimated 1.8 curies of radioactivity was released to the basin during its use. Due to natural radioactive decay, an estimated inventory of less than 0.8 curies remained in 1998 at the time of the Explanation of Significant Differences to the ROD (WSRC 2004d). Major contaminants include Cs137 and mercury. The top two feet of the former basin bottom soils contain 53% of the Cs137 and 97% of the mercury. Groundwater monitoring data also revealed that I129, nitrate, Sr90, uranium isotopes, and tritium are present in the groundwater above maximum contaminant levels.

The remedial actions selected in the OFASB ROD (WSRC 1997e) were (1) removal of vegetation; (2) implementation of institutional controls for the pipeline and pipeline soils; (3) removal of the top two feet of ditch line soils with placement in the OFASB; (4) in situ grouting of the top two feet of basin bottom soils and ditch line soils placed in the basin; (5) backfill of the basin with clean soil with compaction to grade; (6) placement of a low-permeability soil cover over the OFASB area to minimize surface infiltration; and (7) installation of a monitoring well network evaluating groundwater and institutional controls. The remedial action was started on September 10, 1998, and completed on June 9, 2000 (WSRC 2007a).

8.2.4.9.2 Inventory and Basis for Estimate

The remedial actions for the OFASB identified in WSRC (2007a) identify the placement of a low-permeability closure cap over the facility and the in-situ stabilization by the mixing of grout with the soils. The CA analysis does not assume the closure cap and the soil-grout integrity last for the full period of analysis (1,000 yrs); hence it is possible that more radionuclides than are listed in ERD (2001) as COCs could impact the groundwater at concentrations greater than the MCL.

A calculation of radionuclide inventory was made based on soil analysis results provided in WSRC (1995) and an estimate of the mass of contaminated soil associated with this waste unit. The strategy employed was to compare the computed inventory of radionuclides to those reported in ERD (2001) and take the higher activity level as the base estimate for 2002. Because quite a few additional radionuclides are reported to be present in WSRC (1995), the inventory for those radionuclides was included in the current estimate and conservatively assumed to be valid for 2002. In the ERD (2001) estimate, the inventories for several isotopes of U were combined into a composite inventory; however this composite inventory was less than the combined inventories for U238 and U235 that were computed from WSRC (1995); hence the computed inventories were adopted. For Sr90, the ERD (2001) estimate was higher and that was the inventory that was adopted for this CA.

The base radionuclide inventory is identified for 2002, which is the date of the ERD (2001) inventory (December 2001) and the radionuclide inventory retained from the previous CA were conservatively assumed to reflect this date. The 2002 radionuclide inventory for the OFASB is presented in Table A-36.

8.2.5 H-Area Facilities

8.2.5.1 H-Area Tanks and Ancillary Equipment

8.2.5.1.1 <u>Background</u>

The HTF is in the north-central portion of the SRS and occupies approximately 22 acres within H-Area. The HTF is an active radioactive waste storage facility consisting of 29 carbon steel waste tanks and ancillary equipment such as transfer lines, evaporators and pump tanks. The HTF stores and processes liquid radioactive waste generated primarily from the PUREX process. HTF began radioactive operations in 1954. The HTF waste tanks and systems are operated under an Industrial Wastewater permit granted by SCDHEC.

The HTF waste tanks were built during four separate construction periods, with a different tank design for each period, leading to the designation of the following five different tank groups:

- Tanks 9 through 12 are Type I tanks and were constructed in the early 1950s.
- Tanks 13 through 16 are Type II tanks and were constructed between 1955 and 1956.
- Tanks 21 through 24 are Type IV tanks and were constructed between 1958 and 1962.
- Tanks 29 through 32 are Type III tanks and were constructed between 1967 and 1970.
- Tanks 35 through 43 and 48 through 51 are Type IIIA tanks and were constructed between 1974 and 1981.

The HTF is currently in the operational period, during which residual waste removal from the waste tanks is in progress, and the tanks will be grouted at closure. It is currently anticipated that the operational period will last until between 2020 and 2030. The End State for each of the 29 underground waste tanks in HTF will include cleaning, filling with grout to stabilize residual material, and closing in place. Once the HTF waste tanks and ancillary equipment have been grouted, a closure cap will be installed and a 100-year period of institutional control will begin. The closure cap will be monitored, maintained and repaired as necessary during the institutional control period.

Ancillary equipment includes transfer lines, transfer line secondary containment, pump tanks, pump pits, the HTF catch tank, DBs, valve boxes, and the evaporator systems. Over the operating life of the facility, radioactive waste comes in physical contact with some of these components, contaminating them and hence, leaving contamination on the components. The degree of contamination depends on many factors, which include, but are not limited to, the service life of the component, the material of construction, and the type of waste in contact with the component. Some of the listed equipment serves only as secondary containment and may not have contacted the waste.

For the purpose of this effort ancillary equipment inventories are estimated for the following three categories: 1. Transfer lines; 2. Pump tanks; and 3. Evaporator Vessels. Ancillary equipment including the evaporator buildings, the catch tank, PPs, and DBs to the extent practical will remain in place and be grouted, as practical, to eliminate subsidence potential. An exception to leaving ancillary equipment in place will be made for equipment that is significantly higher in elevation than the adjacent waste tanks and would therefore result in a significant increase in the closure cap elevation. Decontamination and dismantling of such ancillary equipment would be conducted to reduce it to an appropriate elevation for closure cap construction. Above grade structures, utilities, equipment, etc., (other than substantial above-grade concrete associated with the tanks and ancillary equipment) that could interfere with closure cap construction will be removed from the HTF area prior to installation of the closure cap.

8.2.5.1.2 Inventory and Basis for Estimate

8.2.5.1.2.1 HTF Tanks

The estimates provided apply to the 29 underground waste tanks in the HTF. They include estimated inventories for 70 radionuclides based on projections and sampled values of HTF waste tank residual concentrations at closure. (WSRC 2008d, WSRC 2005a, WSRC 2006e, SRNS 2008)

The radionuclides included were not evaluated for their influence on groundwater modeling or dose. Instead, the list of radionuclides was based on the inventory present within HTF.

The SRS has used the following general approach for estimating radiological and non-radiological inventories:

- The concentrations of 40 radionuclides in the residual material are estimated using data from the SRS WCS; the concentrations of 30 other radionuclides are calculated separately. For all waste tanks, the residual curie inventory is based on the amount of activity present within a tank's heel with an average height of 0.0625 inch of settled solids on the waste tank bottom. This heel volume is based on experience with cleaning Tank 16 using oxalic acid, a cleaning process planned for use on the remaining tanks. The total radionuclide and non-radiological constituent inventories in these 29 waste tanks are estimated by multiplying the estimated residual solids volumes by the concentrations.
- The total HTF radionuclide inventory is estimated by adding the estimates for the 29 waste tanks.
- Tank 16 has been cleaned with oxalic acid and samples of residual material have been analyzed. The sample characterization was not extensive. Therefore, Tank 15 was used as a surrogate to estimate the remaining inventory in Tank 16.
- Only the Type II tanks' annuli (13-16) are assumed to contain residual material, including the sand bed beneath the primary liner, due to the leak history of these tanks.
- Only the sand bed beneath the Tank 16 secondary liner is assumed to contain residual material, due to a tank leak where material overflowed the containment pan.
- The HTF non-radiological inventory is estimated using the same process, making use of non-radiological concentration data in the WCS database.

The system plan calls for the last waste tank to be grouted at the end of the fiscal year 2032. Therefore all the inventories have been decay corrected to October 1, 2032. The anticipated 2032 radionuclide inventory for all H-Area High Level Tanks in their closure configuration is presented in Table A-37.

8.2.5.1.2.2 HTF Ancillary Equipment

For the HTF ancillary equipment, it is assumed that over the operating life of the facility, radioactive waste comes in physical contact with some of the ancillary equipment, contaminating them and hence, leaving contamination on the components. The degree of contamination depends on many factors, which include, but are not limited to, the service life of the component, the material of construction, and the type of waste in contact with the component. Some of the listed equipment serves only as secondary containment and may not have contacted the waste.

The rationale for estimating the radionuclide inventory for each of the ancillary equipment components is summarized as follows. The basis was to estimate what the residual quantity of process material at closure would be and to then utilize an average radionuclide distribution and activity for that residue in order to calculate the inventory.

The inventory of each tank was used to establish the characterization of the contamination medium estimated by inventory tracking in the SRS WCS and decaying activity to the expected date of closure, 10/1/2030. The results of a review of waste transfers within HTF and between HTF and FTF have been sorted to determine the percentage of the volume of all waste transfers that can be attributed to each HTF waste tank. The representative concentration was then determined by applying a weighted average to each isotopic distribution in the HTF tanks.

Because the characterization of dry sludge was used for each tank for conservatism, it is assumed that the sludge would have to be converted to a slurry in order to remove it from the tanks through the ancillary equipment. It is important to note that, while the sludge concentrations are used, dry sludge is only a small portion of the total waste that passes through the transfer lines that are routinely flushed with a high volume of supernate.

Using the dry sludge concentrations provides a conservative representation of the actinides and long-lived isotopes. The TSR limits waste transfers to less than 16.7 wt% solids. (WSRC 2008c) Therefore, the slurry concentration is reduced to 20% of the dry sludge concentration.

Additional information utilized to estimate the radionuclide inventory associated with individual items of HTF ancillary equipment are provided below.

Transfer Lines

The amount of residual material in the piping systems was determined analytically. The methodology in WSRC 2005 is used for the transfer lines, but with HTF concentrations. The HTF ancillary equipment transfer line inventory includes half the inter-area transfer line between FTF and HTF, the entire transfer line between HTF and the Defense Waste Processing Facility (DWPF), and the entire transfer line between HTF and the Saltstone Processing Facility (SPF). Waste in contact with piping systems adheres to the pipe in three ways: diffusion into the metal, into an oxide film, and as residue left behind after a transfer and flush. Diffusion calculations assume a 100-year contact time and a 100°C exposure temperature.

Pump Tanks

Pump tanks differ from piping systems with respect to such features as geometry and usage. Only residue, left behind after rinsing and flushing, is considered for these components. After tanks are cleaned and inspected, it is expected that residual inventory will be very low. It is assumed that 1/16 inch of residual material will remain in these vessels, on the tank bottom, consistent with the waste tank residual depth.

Evaporator Vessels

The evaporator vessels are cylindrical shaped with a cone bottom, and are constructed of stainless steel. The stainless steel will allow for thorough cleaning, prior to closure. A nominal volume of 2 gallons of residue is used to estimate an inventory at closure, consistent with results experienced during the 242-F evaporator cleaning.

No significant inventory is assumed for the remaining ancillary equipment (e.g., pump pits, DBs, valve boxes, transfer line jackets). The remaining ancillary equipment is not intended for waste storage and is providing a secondary containment function.

The radionuclide inventory for the combined components that comprise the HTF ancillary equipment is presented in Table A-38 and has been decay corrected to October 1, 2032. This estimate also includes the ETP Lift Station as part of the ancillary equipment because it is located within the HTF.

8.2.5.2 H-Area Inactive Process Sewer Lines

8.2.5.2.1 Background

The H-Area Inactive Process Sewer Line (HIPSL) extends from H-Canyon to the H-Area Seepage Basins and consists of manholes, treblers, a diversion box, and 18-inch diameter sewer lines. The vitrified clay lines were utilized from 1955 to 1982 to transport hazardous and low-level radioactive wastes. Due to deterioration and leakage, the vitrified clay lines were abandoned in place and replaced with a set of HDPE lines (WSRC 1993). The portion of the HIPSL outside the H-Area fence has been closed under RCRA, however this CA addresses the entire length from H-Canyon to the H-Area Seepage Basins.

The HIPSL portion located outside of the fenced area was remediated in accordance with the RCRA Closure Plan for the General Separations Area Consolidated Unit. The selected remedy is to grout manholes and treblers in place and to install a geosynthetic cover over the length of the vitrified clay pipe and around the perimeters of the manholes and treblers associated with the HDPE lines (WSRC 2007). The segment located within the fenced area will be remediated at a later date.

8.2.5.2.2 Inventory and Basis for Estimate

Two estimates of the radionuclide inventory in the HIPSL were utilized to determine the inventory to utilize in this CA. The most recent estimate of radionuclide inventory originates in ERD (2001), which is an SGCP inventory of the COCs for their closure units. These COCs represent only those expected to impact groundwater at concentrations greater than the MCL; hence the additional radionuclides identified and quantified in the previous CA (WSRC 1997) were included in this estimate. These estimates are based on the results of the maximum inventories for each radionuclide encountered in the soil characterization program for HIPSL and estimating the mass of soil material potentially contaminated by leakage. Where both sources identified an inventory for the same radionuclide, the higher inventory was adopted. Where the ERD (2001) estimate combined different isotopes of the same element (Pu and U), the combined inventory for the same isotopes in the previous CA were used to compare and determine the higher inventory. In both cases, the levels from the previous CA were higher and hence, they were conservatively adopted for use in this analysis.

The base radionuclide inventory is identified for 2002, which is the date of the ERD (2001) inventory (December 2001) and the radionuclide inventory retained from the previous CA was conservatively assumed to reflect this date.

This analysis is for the entire length of the HIPSL, including segments located from the H-Canyon to the perimeter fence as well as the length from the perimeter fence to the H-Area Seepage Basins. The 2002 radionuclide inventory provided in Table A-39 reflects the full length of the HIPSL.

8.2.5.3 H-Area Retention Basins (281-1H, -2H, -8H)

8.2.5.3.1 Background

Basin 281-1H is the Return Water Delaying Basin, which was brought into operation in 1955 and is still active. It is located within the H-Area fence and sits adjacent to the H-Canyon building. Basin 281-2H is the Return Water Pumping Basin, which was brought into operation in 1955 and is still active. This basin is also located within the H-Area fence and sits adjacent to the H-Canyon building. The planned End State of these facilities is expected to be ISD. Basin 281-8H is the lined Storage Basin. It was brought into operation in 1974 and is still active. This basin is located near H-Area, just south of E-Road and immediately to the west of basin 281-3H. Proposed End State is ISD.

8.2.5.3.2 Inventory and Basis for Estimate

An initial estimate of the radionuclide inventory for all three combined basins is presented in ERD (2001). A separate estimate was computed for each basin, individually, by utilizing the soil characterization data from basin 281-3H (maximum concentrations, in pCi/g), which is documented in WSRC (2003c), and multiplying that concentration by the mass of soil estimated for 281-1H, -2H and -8H. Where inventories were reported for combined isotopes, the inventories were apportioned to each isotope based on the ratios of those isotopes reported for the ORWBG in the previous CA.

The 2002 radionuclide inventory for these three retention basins is presented in Table A-40. An inventory has been provided for each basin, individually.

8.2.5.4 H-Area Seepage Basins

8.2.5.4.1 Background

The H-Area Seepage Basins are a part of the H-Area HWMF OU, which is located in the central portion of SRS, approximately six miles from the nearest site boundary. The H-Area HWMF operated from 1955 until November 7, 1988. The original H-Area HWMF consisted of Basins 904-44G, 904-45G and 904-46G and operated from 1955 to 1962. In 1962, 904-46G was replaced by 904-56G. At the time of closure, the H-Area HWMF (904-44G, 904-45G, and 904-56G) had a combined maximum operating capacity of 26.5 million gallons of wastewater.

The H-Area HWMF received waste effluents from H-Area chemical separation facilities such as the nitric acid recovery unit, waste storage system evaporator overheads, and general purpose evaporator overheads. Significant amounts of nitrate and caustic were received in the basins. Radioactive releases were greater than 99 percent tritium. A 1984 soil coring study showed that approximately 90 percent of the radionuclides, cations, and anions were concentrated within the top one foot of basin soil.

The four basins were closed by dewatering; physically and chemically stabilizing the remaining sludges with a layer of granite, limestone, and blast furnace slag; and placing a protective multi-layer compacted clay cover system over them to reduce rainwater contact with basin bottoms. The remedy prevents physical exposure to contaminants and mitigates further migration of contaminants from the H-Area HWMF to groundwater by minimizing a liquid medium pathway (rainwater percolation) for transport. Closure of the unit was begun in 1989 and completed in May 1991. The H-Area HWMF was certified closed in July 1991.

8.2.5.4.2 Inventory and Basis for Estimate

Two main sources of information were utilized to estimate the radionuclide inventories for the H-Area Seepage Basins, the Environmental Restoration Division estimate documented in ERD (2001) and the previous CA estimate documented in WSRC (1997).

The remedial actions for the H-Area HWMF OU identified in WSRC (2007a) identify the placement of a low-permeability closure cap over the facility and the in-situ stabilization of the soils within the Seepage Basins. The CA analysis does not assume the closure cap and the stabilized soil will maintain integrity for the full period of analysis (1,000 yrs) hence it is possible that more radionuclides than are listed in ERD (2001) as COCs could impact the groundwater at concentrations greater than the MCL. Hence, the longer list of radionuclides listed for the H-Area HWMF OU in the previous CA (WSRC 1997) were retained and utilized to project an SRS End State radionuclide inventory.

The strategy employed in this calculation was to compare the inventory of radionuclides reported in both ERD (2001) and WSRC (1997) and take the higher inventory as the base estimate for 2002. Because several additional radionuclides are reported to be present in the Seepage Basins in the WSRC (1997) analysis, the inventory for those radionuclides was included in the current estimate and conservatively assumed to be valid for 2002.

In the ERD (2001) estimate the inventory for several isotopes of Pu and U were combined into a composite inventory. In the situation where the combined inventory exceeded the combined inventory for the same isotopes reported in WSRC (1997), the ERD (2001) estimate was adopted and the composite inventories were apportioned back to the individual isotopes according to the ratio defined in the previous CA (WSRC 1997).

The base radionuclide inventory is identified for 2002, which is the date of the ERD (2001) inventory (December 2001) and the radionuclide inventory retained from the previous CA was conservatively assumed to reflect this date. The 2002 radionuclide inventory for the H-Area Seepage Basins is presented in Table A-41.

8.2.5.5 HB Line

8.2.5.5.1 Background

The HB-Line is comprised of a hardened structure located on the Fifth and Sixth Levels of the H-Canyon, a one-story office building appendage located on the Fifth Level of the H-Canyon, and a segregated area (outside the hot and warm canyons) in the southwest corner of the H-Canyon Third and Fourth Levels. The hardened structure and the office building, which is south of the hardened structure, are commonly referred to as the new HB-Line. The segregated area on the Third and Fourth Levels of H-Canyon is commonly referred to as the Old HB-Line (WSRC 2006f).

8.2.5.5.2 Inventory and Basis for Estimate

The inventory estimates developed for the HB Line were based on the inventories analyzed in the 2006 Safety Analysis (WSRC 2006f). An End State Reduction Factor of 1E-2 has been applied to approximate the fraction of the facility inventory that might remain with an End State configuration of ISD (Table 4.3b, Appendix J, DOE 2005).

The SAR uses inventories for the third, fourth, fifth, and sixth levels of the H Canyon, which make up the HB Line. The estimates for all of the cells were summed to give HB-Line totals. The materials processed in the HB Line are either product or scrap product and thus the assumption is made that no other radionuclides occur in the residual contamination at facility closure. For the CA analysis the residual radioactive material should be considered to be distributed within a concrete monolith. The HB Line is an operating facility, which is not slated for closure until the assumed SRS end state date of 2025. Therefore the end state inventory estimated from the 2006 SA is assumed to occur in 2025 at the building's end state. The 2025 radionuclide inventory for the HB Line is presented in Table A-42.

8.2.5.6 H-Canyon

8.2.5.6.1 Background

The H-Canyon Building is located near the center of the SRS in H-Area. The nearest site boundary to H-Area is approximately 7.5 miles to the west.

- The H-Canyon Building is a radiochemical processing plant. The process operations conducted in H-Canyon are called the H-Modified process. Operations conducted in H-Canyon include the separation and recovery of uranium, plutonium, and neptunium from irradiated fuel and targets, and purification of Pu238 in support of HB Line. The process equipment is located in two parallel canyons, a "Hot" and a "Warm" Canyon, separated from each other by a central operating and service section that is divided into four levels

The more highly radioactive processing operations are performed in the Hot Canyon; these operations include dissolution of irradiated materials, precipitation and centrifugation, bulk fission product separation, ion exchange, and HAW evaporation. The final purification of the product streams and LAW evaporation are performed in the Warm Canyon. Equipment and fuel are transported and manipulated by remote cranes.

8.2.5.6.2 Inventory and Basis for Estimate

The inventory estimates developed for the H-Canyon were based on the inventories analyzed in the 2007 SAR (WSRC 2007f). An End State Reduction Factor of 1E-2 has been applied to approximate the fraction of the facility inventory that might remain with an End State configuration of ISD (Table 4.3b, Appendix J, DOE 2005).

Table A.1.3-1 in WSRC (2007f) provides concentrations, in Curies per pound (Ci/lb), for four process groups in H Canyon. The total Ci/lb for each group is also given in the same table. The first group includes the processes that separate fission products from actinides, so that group is the only one assumed to contain fission products in significant quantities. Table 8.1-3 in the same report gives the capacity of each vessel involved with processing in H Canyon, in lb of water, as well as the specific gravity and Ci/lb of the stream processed in the vessel. Radionuclide activity ratios from Apperson (1983) were used to estimate inventories for those radionuclides important for CA purposes that are not significant to SAR analyses, i.e., long-lived mobile radionuclides such as C14, Tc99 and I129. This information was combined to give the total Ci of each radionuclide. The H-Canyon is an operating facility, which is not slated for closure until the assumed SRS end state date of 2025. Therefore the end state inventory estimated from the 2007 SAR is assumed to occur in 2025 at the building's end state. The estimated 2025 radionuclide inventory for the H-Canyon is presented in Table A-43.

8.2.5.7 Receiving Basin for Offsite Fuels (RBOF)

8.2.5.7.1 Background

The Receiving Basin for Offsite Fuels (RBOF) has been transitioned from operations in preparation for ultimate disposition (WSRC 2007g). The main missions of RBOF were to:

- Store legacy material from RBOF (e.g., irradiated and unirradiated material, Porostone Filters, sludge, entrained radionuclides, basin water, and residual radioactive and chemical materials) throughout the facility in shutdown systems
- Stage/Store casks and other miscellaneous equipment from other Spent Fuel Project (SFP) facilities
- Stage and handle facility waste
- Maintain radionuclide inventory of RBOF below Hazard Category 3 guidelines and Criticality mass limits

The RBOF (Building 244-H) encloses six reinforced concrete water-filled basins that extend to a maximum below grade depth of 45 feet. RBOF also houses a CR, contains a below-grade structure, and has two concrete-shielded cells. The basins, connected by two transfer canals, have a maximum volume of approximately 500,000 gallons, with the largest basin containing 200,000 gallons. There is also a pit for washing and decontaminating casks.

The RBOF Basins have been de-inventoried of all spent nuclear fuel. Targets and neutron sources have also been removed. Future operations involving these materials will not be performed. Radiological hazardous material (contaminated water, sludge, and activated scrap) remains in the basin and cask wash pit. Contaminated material (e.g., scrap, tools, storage racks and associated hardware, slug buckets, yokes, beams, test reactor components and containers, and underwater saw) also remains.

The Resin Regeneration Facility (RRF) (Building 245-H) is coupled with Building 244-H and is considered an integral part of the facility. It contains the process and support equipment that was used for regeneration of anion and cation exchange resins from the spent fuel basin facilities. The RRF (Building 245-H) consists of three major units: a concrete-shielded resin regeneration cell, a CR, and an exterior platform with two chemical pads, two parking areas, and two hose cabinets.

Resin was transferred between the RBOF and RRF by means of permanently installed piping. The tanks (EP-57, EP-59, and EP-63) in the resin regeneration cell were utilized to separate and regenerate the anion and cation resins used to deionize the basin water in the 100 Areas and RBOF. These tanks were also utilized to deplete the RBOF resin prior to disposal. Resin regeneration or resin depletion is no longer performed. All equipment has been deactivated and flushed. Only residual material remains in the piping and vessels.

8.2.5.7.2 Inventory and Basis for Estimate

The inventory estimates developed for the nuclear facilities within RBOF were based on the inventories analyzed in the 2007 Auditable Safety Analysis (ASA) (WSRC 2007g). An End State Reduction Factor of 1E-2 has been applied to approximate the fraction of the facility inventory that might remain with an End State configuration of ISD (Table 4.3b, Appendix J, DOE 2005).

Table 1 of the ASA gives the residual radionuclide inventory for both RBOF and RRF on a radionuclide basis for a number of locations within the facility. The column giving the total building inventory with sludge was used. Radionuclides of importance to the PA and CA, such as C14, Tc99 and I129, are included in the ASA analysis, so additional quantities were not calculated. Since the ASA was published in 2007, the inventories were assumed to be for that year. For the CA analysis the residual radioactive material should be considered to be distributed within a concrete monolith. The 2007 radionuclide inventory for RBOF is provided in Table A-44.

8.2.5.8 H-Area Sand Filter

8.2.5.8.1 Background

The H-Area Sand Filter, also referred to as the H-Canyon Exhaust Filter, is an underground, concrete structure associated with the H-Canyon building that was utilized to filter exhaust air from the process areas to remove virtually all airborne radioactive material prior to venting to the atmosphere. This filter is rectangular in shape and contains beds made of layers of coarse stone and succeeding layers of finer and finer gravel and sand for a total filter depth of about 8.5 feet. A 61-meter (200-foot)-tall stack behind the canyon discharges this filtered air to the atmosphere and serves as the pathway for airborne emissions associated with the normal operation of the canyons. The facility became operational in 1955 and is still active. The proposed End State is ISD.

8.2.5.8.2 Inventory and Basis for Estimate

The radionuclides associated with the H-Area Sand Filter and the inventories of each were obtained from the previous CA, which is documented in WSRC 1997. The inventories for each radionuclide were representative of 1997, but are conservatively assumed to be the same in 2002. The 2025 radionuclide inventory for the H-Area Sand Filter is provided in Table A-45.

8.2.5.9 Original Tritium Facilities

8.2.5.9.1 Background

The original Tritium Facilities, some constructed as early as 1954, were designed and built to provide the U.S. Government with tritium manufacturing and processing capabilities necessary for thermonuclear weapons production and other non-weapon uses (WSRC 2007h). The term "original" here refers to those facilities associated with the original SRS mission of tritium production in association with SRS on-site reactors. The process buildings for the original Tritium Facilities are as follows:

- Building 232-H (tritium purification and separation)
- H-Area New Manufacturing (HANM) Building (reservoir/special container processing and isotope separation/purification)
- H-Area Old Manufacturing (HAOM) Building (tritium reservoir finishing, packaging and shipping; inert reservoir loading, finishing, packaging and shipping)
- Building 236-H (He3 purification)
- Building 238-H (reservoir reclamation and burst testing)
- Building 234-7H (reservoir surveillance operations [RSO] and gas transfer system [GTS] missions)

Nonprocess buildings include the following:

- Vault 217-H (included as part of the HAOM facility since the HAOM footprint includes the vault)
- Building 233-1H (enclosed passageway that connects Buildings HAOM and 249-H)
- Building 237-H (storage building - a portion of its floor space dedicated to the storage of reclaimed reservoirs, but access to the storage area is by way of Building 238-H. It is included as part of Building 238-H in this document)
- Building 249-H (service building which houses auxiliary equipment used in support of operations in the HANM facility)

8.2.5.9.1.1 Historical Mission

Construction of Building 232-H began in 1954. The respective startup times for Lines I/II, and Line III of Building 232-H are 1955 and 1962. The original mission of the facility was to extract tritium and to provide separation and enrichment of the hydrogen isotopes recovered from the lithium/aluminum targets and control rods received from the SRS reactors, and from the gas returned from reservoirs at the end-of-life cycle.

Historically, the RSO and GTS missions were performed in 232-H. Planning for the deactivation of 232-H necessitated the transfer of these missions to a new building. Building 234-7H was designed and built for this purpose.

Construction of the HAOM Building began in 1956. Startup of the HAOM Building was in the time frame of 1957-1958. This building provided a large-scale production facility for the tritium loading/unloading operations. The HAOM Building received tritium-filled reservoirs from DOE/Department of Defense (DOD) for unloading and virgin reservoirs from DOE for initial loading. Until the construction and startup of Building 238-H in 1969, the unloaded reservoirs were packaged and sent to the Solid Waste Disposal Facility. The HAOM Facility also provided finishing, packaging and shipping operations for all filled reservoirs.

Construction and startup of Building 236-H occurred in the time frame of 1964-1965. This facility purified the tritium decay byproduct He3 and burst-tested unloaded reservoirs for the Tritium Facilities. A pneumatic burst-test facility was constructed on the north side of the building in 1981; however, the burst test facility has been deactivated.

Construction and startup of Building 238-H occurred in 1969. The mission of this facility was to reclaim unloaded tritium reservoirs and return them to the HAOM Building for reloading.

Construction of the HANM Building began in 1989. Startup of the facility was approved in 1993. The mission for this facility is to replace the aging loading facilities in the HAOM Building. The facility loads new reservoirs supplied by the DOE or recycles reclaimed reservoirs. The reservoirs are loaded with a deuterium/tritium mixture as specified by the Design Agencies - Lawrence Livermore National Laboratory (LLNL), Sandia National Laboratory - Livermore, or LANL. Returned gas can be separated and purified by this facility for reloading into reservoirs.

A significant process addition began in the HANM Building in 1999 with project S-7726, Tritium Facility Modernization and Consolidation. The purpose of this modification is to add isotope purification and separation capability for protium-tritium mixtures utilizing hydride technology enclosed within glovebox confinement systems as well as an isotope recovery system to collect residual protium-tritium gas mixtures from processing operations.

8.2.5.9.1.2 Current Missions

The isotope purification and separation mission for 232-H has been transferred to the HANM Building. Building 232-H has been de-inventoried and is no longer in service. The current mission for the HANM Building is centered on the unloading and loading of tritium reservoirs and also has separation and enrichment capabilities. Recently, additional purification and separation capability has been added utilizing hydride technology enclosed within glovebox confinement systems. The HAOM Building is now used to load inert-filled reservoirs and supplies the finishing, packaging and shipping operations for all reservoirs and Non-War Reserve shipments. The facility is used to store reservoirs returned from the field until the deuterium/tritium gas is needed in the HANM Building process.

Building 234-7H is equipped to environmentally condition weapons assemblies/components for the RSO and GTS missions, and to support material characterization studies of tritium-exposed specimens. The current missions for Building 236-H are byproduct (He3) purification/loading, and inter-building transfer of tritium contaminated gases. The mission for Building 238-H is now to house the Hydraulic Burst Test System, which is part of the SRS RSO mission. This new mission is in addition to the reservoir reclamation mission that continues. The current mission also includes sealing empty returned reservoirs, then sending them to the Solid Waste Disposal Facility.

8.2.5.9.2 Inventory and Basis for Estimate

The inventory estimate developed for the original Tritium Facilities was based on the inventories analyzed in the 2007 SAR (WSRC 2007h). An End State Reduction Factor of 1E-2, applicable to an End State configuration of "In Situ Disposition", has been applied to the original Tritium Facilities, which are massive concrete structures that are not likely to be demolished (Building 232-H, HANM Building, and Building 236-H). An End State Reduction Factor of 1E-3, applicable to an End State configuration of "Demolish to Concrete Slab" has been applied to the original Tritium Facilities which are steel frame structures that are likely to be demolished (HAOM Building, Vault 217-H in the HAOM Building, Building 234-7H, and Building 237/238-H). [These assumptions result in a revision to the CA Inventory report (Hiergesell et al. 2008) estimated inventory for the HAOM Building, Vault 2-17-H in the HAOM Building, Building 234-7H, and Building 237/238-H, where the "In Situ Disposition" configuration was incorrectly assumed for these structures]. Chapter 4 of the End State Vision Report (DOE 2005) says that the tritium mission at SRS will extend past the time frame encompassed in that report, so no End State is projected.

The SAR uses residual inventories for portions of the original Tritium Facilities. The mass of residual tritium for each part is given in Table 3.3-7 of WSRC (2007h). The specific activity of tritium from Cook (2007) was used to convert mass to inventory in curies. The mass of Pu238 in sources in the HAOM Building was converted to inventory in the same manner. The materials within the original Tritium Facilities are essentially pure tritium and pure source material, so the assumption is made that no other radionuclides occur in the residual contamination at facility closure. Building 232-H in no longer in use, therefore its inventory was assumed to be for the year of publication (i.e., 2007) of the SAR (WSRC 2007h). The other original Tritium Facilities (HANM Building, Building 236-H, HAOM Building, Vault 217-H in the HAOM Building, Building 234-7H, and Building 237/238-H) are still operating facilities, which are not slated for closure within the SRS End State Vision Report (DOE 2005). However for the purposes of the CA an end state date of 2025 has been assumed. Therefore the end state inventory estimated from the 2007 SAR is assumed to occur in 2025. For the CA analysis the residual radioactive material should be considered to be distributed within a concrete monolith for those facilities with an assumed End State of "In Situ Disposition" and within a concrete slab for those facilities with an assumed End State of "Demolish to Concrete Slab." The inventories for the original Tritium Facility buildings are presented in Table A-46.

8.2.5.10 Tritium Extraction Facility and Remote Handling Building

8.2.5.10.1 ***Background***

Present-day SRS Tritium Facilities, as part of the Tritium Extraction Facility (TEF) process, provide the means of extracting tritium from tritium-bearing targets irradiated in commercial light-water reactors, rather than in on-site reactors (WSRC 2007h). The TEF and Remote Handling Building (Building 264-2H) together are used in carrying out the TEF process. Buildings 233-32H, 264-1H, 264-6H and 264-7H are other industrial facilities associated with the TEF process.

8.2.5.10.1.1 Historical Mission

Historically, tritium was provided to the nuclear weapons stockpile from heavy water moderated reactors at SRS and extracted in Building 232-H. The TEF and Building 264-2H currently provide the capability to extract tritium from Tritium Producing Burnable Absorber Rods (TPBARs) irradiated in commercial Tennessee Valley Authority Light Water Reactors.

8.2.5.10.1.2 Current Missions

The two primary missions housed in these two buildings include the extraction mission (Building 264-2H) and the tritium processing mission. The primary mission of the extraction process is to receive and extract tritium from TPBARs that have been irradiated in a Commercial Light Water Reactor (CLWR). The primary mission of tritium processing is to clean up extraction gases for delivery to the HANM Building.

The TEF and Building 264-2H are also associated with several other support buildings and structures. The functions of these buildings are as follows:

- Provide personnel housing (Building 264-1H)
- Electrical supply (Buildings 252-68H and 254-21H)
- Ventilation supply Heating, Ventilation, and Air Conditioning (HVAC) (Bldg 264-4H)
- Service gases (Building 264-5H)
- Process and HVAC chillers (Building 264-6H)
- Cooling towers (Building 264-3H)
- Low level waste storage (Building 233-32H)

8.2.5.10.1.3 Future Missions

The future missions of the TEF and Building 264-2H should not significantly change. Provisions have been made for a future expansion to add a second Target Rod Preparation Module, third Extraction Furnace Module, and second Mass Spectrometer, if necessary, to support increased production at a later date.

8.2.5.10.2 Inventory and Basis for Estimate

The inventory estimate developed for the TEF and Building 264-2H was based on the inventories analyzed in the 2007 SAR (WSRC 2007h). An End State Reduction Factor of 1E-2 has been applied to approximate the fraction of the facility inventory that might remain with an End State configuration of "In Situ Disposition". Chapter 4 of the End State Vision Report (DOE 2005) says that the tritium mission at SRS will extend past the time frame encompassed in that report, so no End State is projected. However for the purposes of the CA an end state date of 2025 has been assumed. Therefore the end state inventory estimated from the 2007 SAR is assumed to occur in 2025.

The maximum total mass of tritium for the TEF and Building 264-2H was taken from Table 3.3-8 and 3.3-7, respectively of WSRC (2007i). The specific activity of tritium from Cook (2007) was used to convert mass to inventory in curies. The materials within the TEF are essentially pure tritium, so the assumption is made that no other radionuclides occur in the residual contamination at TEF facility closure. During operations Building 264-2H contains irradiated TPBARs; therefore activation products associated with the TPBARs in addition to tritium are assumed to occur as residual contamination at Building 264-2H closure. For the CA analysis the residual radioactive material should be considered to be distributed within a concrete monolith for the "In Situ Disposition" End State. The 2025 radionuclide inventories for the TEF and Building 264-2H is presented in Table A-47.

8.2.6 N-Area Facilities

8.2.6.1 Ford Building Seepage Basin

8.2.6.1.1 Background

The Ford Building Seepage Basin (FBSB) is part of the FBSB OU and is listed as a RCRA 304(u) Solid Waste Management Unit/CERCLA unit in Appendix C of the FFA for SRS. The media associated with the FBSB OU include soil and groundwater beneath the OU. However, the results of the groundwater investigation contained in the RCRA Facility Investigation/Remedial Investigation (RFI/RI) with Baseline Risk Assessment (BRA) for the FBSB, which included a collection of groundwater samples and analyses, have revealed that the groundwater associated with FBSB OU is not contaminated. The FBSB OU is located near the middle of the SRS between Fourmile Branch and Pen Branch (WSRC 2007a). The selected remedy for the FBSB OU was excavation, dispositioning, backfilling, vegetative covers, and institutional controls.

8.2.6.1.2 Inventory and Basis for Estimate

The radionuclide inventory presented in ERD (2001) was taken as the appropriate 2002 inventory to use in this CA. The only COC identified was Cs137. The 2002 radionuclide inventory for the Ford Building Seepage Basin is provided in Table A-48.

8.2.7 Reactor Area Facilities

8.2.7.1 Reactor Buildings (P-, R-, C-, K-, and L-Reactor Areas)

8.2.7.1.1 Background

SRS brought five production, heavy-water-moderated, reactors into operation between December 1953 and March 1955 for the primary purpose of producing tritium and Pu239 for the national nuclear weapons complex (DOE 1997). In addition to producing tritium and Pu239, over 100 different radioisotopes were made in the production reactors. Most of these were made in minute amounts for various non-weapons and research purposes, and others were made incidentally in the course of other programs. These radioisotopes included, among others, Am243, Cf252, Cs137, Co60, Cm244, Eu152, I131, Ir192, Pu238, Pu240, Pu242, Pu244, Pm147, Pa231, Po210, Sr90, Tc99, Tm170, and U233. The five SRS production reactor buildings and their years of operation are provided in Table 8-5. Table 8-6 provides location information for each production reactor and the SRS custodial organization currently responsible for the building.

Table 8-5. SRS Reactor Operations (Reed et al. 2002)

Reactor Building	Period of Reactor Operations	Years of Operation
105-P	February 20, 1954 – 1988	~35
105-R	December 28, 1953 – 1964	~11
105-C	March 28, 1955 – 1985	~31
KAC	August 14, 1954 – 1988; June 8, 1992 (test run)	~35
LAC	August 11, 1954 – 1968; 1985 - 1988	~19

Table 8-6. SRS Reactor Building Location and Custodial Organization

Reactor Building	SRS Area	Coordinates	SRS Custodial Organization
105-P	P	445,852 UTM E 3,676,551 UTM N	SDD and SGCP
105-R	R	446,014 UTM E 3,681,655 UTM N	SDD and SGCP
105-C	C	436,963 UTM E 3,678,987 UTM N	Nuclear Materials Storage (NMS) Project
KAC	K	438,064 UTM E 3,674,714 UTM N	NMS Project
LAC	L	441,917 UTM E 3,674,622 UTM N	NMS Project

Each production reactor building, in addition to the reactor itself, housed an assembly area for storage and assembly of fresh fuel; a disassembly area, consisting of a large pool of water, for storage of irradiated fuel and disassembly and preparation for transport to the separations plants; and a purification area, for heavy water treatment and purification. These buildings were equipped with filtered ventilation systems to mitigate potential airborne radioactivity releases (DOE 1997).

The five production reactor buildings are essentially the same with just a few differences, especially between the first, 105-R, and the last, 105-C. In essence, the 105-R building is larger than the others, and had more elaborate facilities. As some of these were found to be unneeded, they were either reduced in size or eliminated all together. While the reactor tanks in 105-R, 105-P, KAC, and LAC are essentially identical, the reactor tank in 105-C, the last reactor built, was larger to accommodate a heavy water reflector which increased the size of the high-power flat zone and, hence, total reactor power (Reed et al. 2002).

D&D, by DOE, of facilities listed in Appendix K-1 of the FFA is coordinated with EPA and SCDHEC. The 105-R, -P, and -C buildings have been declared surplus facilities by DOE and are listed in Appendix K-1 of the FFA as CERCLA facilities to be decommissioned. These building are currently in various stages of the decommissioning process. The SRS organizations, Site Deactivation and Decommissioning (SDD) and SGCP, work jointly to decommission these buildings according to the FFA requirements (Musall 2008; SRS 2008; FFA 1993; WSRC 2008e).

The KAC and LAC buildings have not been declared surplus and in fact are involved in current missions. When they are declared surplus they will become CERCLA facilities, and their decommissioning will be performed consistent with the FFA. The KAC currently provides the only SRS interim safe storage for much of the U.S. Department of Energy's (DOE) excess plutonium (SRS 2008a). Since 1998, the refurbished LAC Disassembly Basin has been used to safely handle and store all spent nuclear fuel at SRS (SRS 2008b).

The 105-P building is the most advanced ISD project within the FFA D&D process, with the issuance of an Early Action Proposed Plan for the P Area OU, which includes the 105-P building (WSRC 2008e). Based upon this plan, the 105-P building End State will be one of ISD rather than demolition and disposal. The projected End State of the 105-P building is considered to be representative of all the SRS production reactors. The 105-R building is following close behind 105-P in the FFA D&D process. It is anticipated that ISD for the 105-P building would involve the following (Musall 2008; WSRC 2008e):

- Removal of the stack
- Relocation of certain above-grade contaminated process equipment and associated piping and wiring to below-grade to be encapsulated with grout in place
- Encapsulation with grout of all below-grade equipment in place, including the reactor tank, then capping with a concrete dome
- Demolition of the above-grade portions of the Disassembly Basin, placement of that material in the below-grade portions of the Disassembly Basin, and subsequent grouting in place and covering with an environmental cover system
- Sealing all above- and below-grade penetrations to prevent intrusion

8.2.7.1.2 105-P Reactor Building Inventory and Basis for Estimate

In support of the planned 105-P building ISD (WSRC 2008e), extensive work has been performed by SDD, SGCP, and SRNL to estimate the building radiological inventory (Rose 2008; Rose 2008a; Musall 2008; Vinson and Webb 2008). These 105-P building inventory estimates are based upon sampling and analysis, radiological surveys, characterization, process history, and modeling. The 105-P building inventory and type of building materials containing that inventory, excluding the reactor core, has been extracted from Musall (2008). The inventories extracted from Musall (2008) are based upon a combination of sampling and analysis, radiological surveys, characterization, and process history. The reactor core inventory has been extracted from Vinson and Webb (2008), and the reactor core materials containing that inventory have been extracted from Rose (2008). The Vinson and Webb (2008) reactor core inventory is based upon considering the power flux operational history of the P-Reactor and modeling the resulting production of activation products within the reactor core materials and production of fission products and actinides within the reactor tank stainless steel. The Musall (2008) and Vinson and Webb (2008) inventories were decay corrected to December 31, 2007. The building and reactor core materials containing the inventory have been divided into the following types of materials:

- Volumetrically contaminated 304 stainless steel
- Volumetrically contaminated aluminum
- Volumetrically contaminated concrete
- Surface contamination (the bulk of the surface contamination is located below-grade and will be encased in grout)

Tables A-49 through A-52 provide the 2008 105-P radionuclide inventory in curies for 304 stainless steel, aluminum, concrete, and surface contamination, respectively, as extracted from Rose (2008), Rose (2008a), Musall (2008), and Vinson and Webb (2008).

8.2.7.1.3 105-R Reactor Building Inventory and Basis for Estimate

The 105-R building is the second most advanced ISD project within the FFA D&D process, with the issuance of a Remedial Investigation Work Plan for the R Area OU, which includes the 105-R building (WSRC 2008f). In support of the 105-R decommissioning effort some work has been performed to estimate the R-Reactor radiological inventory by SDD and SRNL (DOE 2002; Rose 2008; Vinson and Webb 2008). The inventory of the reactor core has been estimated by Vinson and Webb (2008), based upon considering the operational history of the P-Reactor, considering the materials of construction of the reactor core (Rose 2008) and modeling the resulting production of activation products within the reactor core materials and production of fission products and actinides within the reactor tank stainless steel.

The inventory of R-Reactor Disassembly Basin liquid and sludge has been estimated based upon sampling and analysis, characterization, and process history as documented within DOE (2002) and WSRC (2008g). The Emergency Basin radiological content has been estimated. No other 105-R building specific inventories have been developed, therefore for building areas and components for which building specific inventories are not available, the inventory of those areas and components have been taken as that of the 105-P building from Musall (2008). This is considered reasonable, because the two reactor buildings are essentially the same and the R-Reactor was only operated for approximately 11 years, whereas the P-Reactor was operated for approximately 35 years. Additionally the P-Reactor core inventory is approximately 3½ times greater than the R-Reactor core inventory. The inventories extracted from Musall (2008) are based upon a combination of sampling and analysis, radiological surveys, characterization, and process history. The inventories are decay corrected to December 31, 2007. The building and reactor core materials containing the inventory have been divided into the following types of materials:

- Volumetrically contaminated 304 stainless steel
- Volumetrically contaminated aluminum
- Volumetrically contaminated concrete
- Surface contamination (the bulk of the surface contamination is located below-grade and will be encased in grout)

Tables A-53 through A-56 provide the 2008 105-R radionuclide inventory in curies for 304 stainless steel, aluminum, concrete, and surface contamination, respectively, as extracted from DOE (2002), Rose (2008), Rose (2008a), Musall (2008), Vinson and Webb (2008), and WSRC (2008g).

8.2.7.1.4 105-C, KAC, and LAC Reactor Building Inventory and Basis for Estimate

While the 105-C building has been declared a surplus facility and listed as a CERCLA facility to be decommissioned consistent with the FFA, essentially no building specific characterization has been performed in support of estimating an inventory. The KAC and LAC buildings are currently operating facilities and have not been declared surplus and are therefore not yet classified as CERCLA facilities requiring decommissioning consistent with the FFA. Therefore the inventories of the 105-C, KAC, and LAC Reactor Buildings will be taken as that of the 105-P building. This is considered reasonable, because the P-Reactor was operated for approximately 35 years and the C-, K-, and L-Reactors operated for 35 years or less (approximately 31, 35, and 19 years, respectively, for C-, K-, and L-Reactors). Additionally as stated previously, the five production reactor buildings are essentially the same. Therefore the 2008 radionuclide inventories for each of the 105-C, KAC, and LAC Reactor Buildings can be taken as follows:

- Table A-49 for volumetrically contaminated 304 stainless steel
- Table A-50 for volumetrically contaminated aluminum
- Table A-51 for volumetrically contaminated concrete
- Table A-52 for surface contamination (the bulk of the surface contamination is assumed to be located below-grade and will be encased in grout)

8.2.7.2 Reactor Seepage Basins (P-, R-, C-, K-, and L-Reactor Seepage Basins)

8.2.7.2.1 Background

Each of the five SRS production reactor buildings contained a Disassembly Basin, which consisted of a large pool of water, for storage, disassembly, and preparation for transport to the separations plants of irradiated fuel and target rods (DOE 1997). Radionuclide contaminants, predominately tritium, entered the disassembly basins with the film of water on the irradiated rods as they were discharged from the reactor tank to the Disassembly Basin, from oxide corrosion films on the irradiated rods, and infrequently from leaks from the rods.

Periodic purging of the Disassembly Basin water was required in order to keep tritium at safe levels (less than 400,000 pCi/ml) for workers in the area (Pekkala 1987). Disassembly Basin purge water was either discharged to the seepage basins or mixed with large volumes of reactor heat exchanger cooling water and discharged to Reactor Discharge Canals, which subsequently entered site streams. Mixed-bed deionizers and sand filters were later used to remove the bulk of the radionuclides, except tritium, from the Disassembly Basin purge water prior to discharge to the canals or seepage basins (WSRC 2002b; WSRC 2003d; WSRC 1998; WSRC 2001; WSRC 2000a; Stone and Christensen 1983). The reactor seepage basins were unlined earthen basins intended to allow as much radioactive decay as possible, particularly for tritium, prior to outcropping into the local streams (Reed et al. 2002). A total of 14 reactor seepage basins were used to receive Disassembly Basin purge water from the five reactor buildings. The seepage basins associated with each reactor area and the period of operation of each seepage basin is provided in Table 8-7. Table 8-8 provides information on the dimensions of each seepage basin, and Table 8-9 provides information on the location for each seepage basin grouping and the SRS custodial organization currently responsible for the seepage basin.

Table 8-7. SRS Reactor Area Seepage Basins Operations

Reactor Area	Seepage Basin Number	Seepage Basin Building Number	Period of Seepage Basin Operations	Years of Operation
R [1]	1	904-57G	November 1957 – January 1958	<1
	2	904-58G	November 1957 – 1960	~3
	3	904-59G	November 1957 – 1960	~3
	4	904-60G	December 1957 – 1960	~3
	5	904-103G	March 1958 – 1960	~3
	6	904-104G	March 1958 – June 1964	~7
P [2]	1	904-61G	1957 – 1970; 1978 – 1988	~25
	2	904-62G	1957 – 1970; 1978 – 1988	~25
	3	904-63G	1957 – 1970; 1978 – 1988	~25
K [3]	1	904-65G	1957 – 1960	~4
L [4]	1	904-64G	1958 – 1968; 1985 – 1988	~15
C [5]	1	904-66G	1957 – 1970; 1978 – 1986	~23
	2	904-67G	1957 – 1970; 1978 – 1986	~23
	3	904-68G	1957 – 1970; 1978 – 1986	~23

[1] WSRC 2002c; [2] WSRC 2003d; [3] WSRC 1998; [4] WSRC 2001; [5] WSRC 2000a

Table 8-8. SRS Reactor Area Seepage Basins Dimensions

Reactor Area	Seepage Basin Number	Seepage Basin Building Number	Seepage Basin Surface Dimensions
R [1]	1	904-57G	L-shaped: 30 ft wide by 394 ft long by 10 ft deep
	2	904-58G	46 ft wide by 131 long ft by 10 ft deep
	3	904-59G	30 ft wide by 295 ft long by 10 ft deep
	4	904-60G	36 ft wide by 305 ft long by 6.6 ft deep
	5	904-103G	39 ft wide by 295 ft long by 10 ft deep
	6	904-104G	46 ft wide by 495 ft long by 16.4 ft deep
P [2]	1	904-61G	L-shaped: 50 ft wide by 465 ft long by 13 to 17 ft deep
	2	904-62G	70 ft wide by 211 ft long by 8 ft deep
	3	904-63G	70 ft wide by 340 ft long by 9 ft deep
K [3]	1	904-65G	70 ft wide by 135 ft long by 7 ft deep
L [4]	1	904-64G	L-shaped: 36 ft wide by 200 ft long by 7 ft deep
C [5]	1	904-66G	L-shaped: 35 ft wide by 430 ft long by 7 ft deep
	2	904-67G	60 ft wide by 300 ft long by 11 ft deep
	3	904-68G	90 ft wide by 180 ft long by 12 ft deep

[1] WSRC 2002c; Pekkala 1987; [2] WSRC 2003d; [3] WSRC 1998; [4] WSRC 2001;
[5] WSRC 2000a

Table 8-9. SRS Reactor Area Seepage Basins Location and Custodial Organization

Reactor Area	Approximate Seepage Basin Coordinates	SRS Custodial Organization
R	445,789 UTM E 3,681,954 UTM N	SGCP
P	445,663 UTM E 3,676,330 UTM N	SGCP
K	437,827 UTM E 3,674,581 UTM N	SGCP
L	442,078 UTM E 3,674,363 UTM N	SGCP
C	436,659 UTM E 3,678,832 UTM N	SGCP

The FFA (FFA 1993) lists all reactor seepage basins as CERCLA facilities, for which final RODs regarding their remediation and closure have been issued. Additionally SGCP has completed all remediation and closure for all reactor seepage basins in conformance with the RODs as documented within Post-Construction Reports / Final Remediation Reports for each set of reactor seepage basins (WSRC 2002d; WSRC 2003e; WSRC 2004e; WSRC 2006g; WSRC 2008h). As such all of the reactor seepage basins are currently in their end-state configuration.

8.2.7.2.2 R-Reactor Seepage Basins (RRSB) Remedial Actions

By 1960 R-Reactor Seepage Basins (RRSBs) 1 through 5 were deactivated and backfilled (WSRC 2008h). In 1960 basins 2, 4, and 5 were filled with "tight" clay and basin 3 was backfilled. Basin 1 had been previously backfilled some time after January 1958. Also in 1960, basins 1 and 3 were surrounded by clay dikes keyed into clayey strata beneath the basins and capped with clay (WSRC 2002b). Basin 6 was deactivated in 1964 and backfilled in 1977 (WSRC 2008h). Also in 1977, the entire seepage basin area was sprayed with an asphalt emulsion. In 1996 approximately 1.5 feet of clean, compacted soil was placed over the existing asphalt emulsion, contoured, and covered with 4 inches of asphalt to enhance runoff (WSRC 2002b). During 2006 and 2007, as part of the final ROD remedial action for the RRSBs (WSRC 2003f), a concrete intruder barrier was placed over all the seepage basins and an asphalt bioturbation barrier was placed surrounding the concrete intruder barrier.

8.2.7.2.3 R-Reactor Seepage Basin Inventory and Basis for Estimate

Table 4-11 in WSRC (2002b) provides an upper limit inventory, decay corrected to 1998, within the combined RRSBs, for the following radionuclides: C14, Sr90, Tc99, I129, Cs137, Pu238, Pu239/240, and Am241. The Table 4-11 upper limit inventory for the basins was derived from an evaluation of past inventory estimates, including direct analysis of Disassembly Basin water discharged to the basins, gamma probe data, and soil boring analyses. During sampling and analysis of soils beneath the seepage basins, the maximum amount detected of the following additional radionuclides was greater than twice the average background concentration for that radionuclide: Na22, K40, Co60, Eu154, Th228, Th230, U235, Np237, Am241/Cm246, and Cm 243/244 (WSRC 2002b, Table 4-5). These radionuclides are also considered contaminants due to seepage basin operation. The inventory for these radionuclides decay corrected to 1998 was determined based upon the inventory of Cs137 and the ratio between the radionuclide's average soil concentration to that of Cs137 within the soils beneath the seepage basins as documented within WSRC 2002b (Table 4-5).

Finally, Ru103/106 and Pm147 were also documented as contaminants due to seepage basin operations within Pekkala et al. (1987) and Holmes et al. (1983), respectively. The inventory of these radionuclides within the RRSBs was taken from these documents, decay corrected to 1998, and shown in Table A-57. Where inventory estimates combined two radionuclides, such as Cm243/Cm244, the inventory for each individual radionuclide in the combination was assumed to be the same as the given combined inventory. The tritium that was discharged to the basins is not included in the Table A-57 inventory, because the tritium has already migrated from the basins to form a groundwater plume, whose inventory has been evaluated separately.

8.2.7.2.4 P-Reactor Seepage Basins (PRSB) Remedial Actions

The P-Reactor Seepage Basins (PRSBs) were closed under a final Plug-In ROD remedial action (WSRC 2003d; WSRC 2006g). During 2005, as part of the final Plug-In ROD remedial action for the PRSBs (WSRC 2006g), the contaminated soils of basins 1 and 2 were stabilized/solidified with grout; the pipelines to the basins were grouted, excavated, and placed in basin 1; and a geosynthetic closure cap was installed over all three basins. Stabilization/solidification of the Basin #1 and #2 contaminated soil consisted of transferring the soil to an on-site portable pugmill, mixing the prescribed amounts of cement, bentonite and water to the soil and transporting the mix back into Basin #1 and #2. The L-Area Hot Shop foundation was also excavated and used to fill Basin #3. Basin #3 was then filled with controlled low strength material (CLSM) before the final engineered cover was constructed.

8.2.7.2.5 P-Reactor Seepage Basins Inventory and Basis for Estimate

During sampling and analysis of soils beneath the PRSBs the maximum detection of the following radionuclides was greater than twice the average background concentration for that radionuclide within Basin #1: C14, Na22, K40, Co60, Ni63, Sr90, Tc99, Sb125, Cs134, Cs137, Pm147, Eu154, Pb212, Bi214, Ac228, Ra228, Th228, Th230, Th232, Np237, Pu238, Pu239/240, Am241, Cm242, Cm243/244, and Cm245/246 (WSRC 2003d Table 1). These radionuclides are considered contaminants due to disassembly basin operation, when contaminated water was purged to the seepage basins. Of four inventory estimates provided for the PRSBs (Holmes et al. 1983; DOE 1990; ERD 2001; WSRC 2003d), ERD (2001) estimated the greatest inventory of Cs137 at 91 curies, decay corrected to 2001. Cs137 is the most predominant radionuclide within all these inventory estimates. Therefore a Cs137 inventory of 91 curies within the PRSBs has been taken as the basis for the PRSBs inventory.

The inventory for these radionuclides in the PRSBs, decay corrected to 2001 and shown in Table A-58, was determined based upon the inventory of Cs137 (91 curies) and the ratio between the radionuclide's total soil concentration and that of Cs137 within the soils beneath Basin #1 as documented within WSRC (2003d Table 2). Additionally, as outlined above, the pipelines to the basins were grouted, excavated, and placed in basin 1. Based upon sampling and analysis of soils adjacent to the pipelines (WSRC 2003d) and the volume of excavated material placed in Basin #1 (WSRC 2006g), the inventory of this material was conservatively estimated to be 2 percent of the PRSBs inventory. Therefore the total inventory of each of the radionuclides within the PRSBs was increased by two percent to account for the pipelines placed in Basin #1.

Where inventory estimates combined two radionuclides, such as Pu239/240, the inventory for each individual radionuclide in the combination was assumed to be the same as the given combined inventory. The tritium that was discharged to the basins is not included in the Table A-58 inventory, because the tritium has already migrated from the basins to form a groundwater plume, whose inventory has been evaluated separately

8.2.7.2.6 K-Reactor Seepage Basin (KRSB) Remedial Actions

The K-Reactor Seepage Basin (KRSB) was closed under a final Plug-In ROD remedial action (WSRC 1998; WSRC 2002d). During 2001, as part of the final ROD remedial action for the KRSB (WSRC 2002d), the contaminated soils in the basin were stabilized/solidified; the pipeline to the basin was grouted in place and a broken portion was excavated and disposed in the basin. A soil cap was then installed over the basin. In situ stabilization/solidification of the basin's contaminated soils consisted of soil mixing to a minimum depth of four feet using a dual auger soil mixing rig to blend the prescribed amounts of cement, bentonite, and water with the basin soils. The above describes the KRSBs' end-state, which appears similar to that of the PRSBs.

8.2.7.2.7 K-Reactor Seepage Basins (KRSB) Inventory and Basis of Estimate

A detailed evaluation and determination of which radionuclides are considered contaminants due to KRSB operation was conducted within WSRC (1998, Section 4.3.1), and the following radionuclides were designated contaminants: C14, Na22, Co60, Sr90, Cs137, Ra226, U233/234, U238, Pu238, Pu239/240, Am241, and Am243. Of four inventory estimates provided for the KRSB (Holmes et al. 1983; Pekkala et al. 1987; WSRC 1998; ERD 2001), ERD (2001) estimated the greatest inventory. ERD (2001) estimated inventories of Co60, Sr90, Cs137, Pu239/240, and Am241 for the KRSB, decay corrected to 2001, is shown in Table A-59. The inventory for the other designated radionuclide contaminants, decay corrected to 2001, was determined based upon the inventory of Cs137 (4.7 curies) and the ratio between the radionuclide's average soil concentration to that of Cs137 within the soils beneath the basin as documented within WSRC (1998, Table 4.6).

Finally, Pm147 was also documented as a contaminant due to seepage basin operations within Pekkala et al. (1987). The Pm147 inventory from Pekkala et al. (1987) was decay corrected to 2001 and proportioned to the Cs137 inventory of 4.7 curies. Where inventory estimates combined two radionuclides, such as U233/234, the inventory for each individual radionuclide in the combination was assumed to be the same as the given combined inventory. The tritium that was discharged to the basins is not included in the Table A-59 inventory, because the tritium has already migrated from the basins to form a groundwater plume, whose inventory has been evaluated separately.

8.2.7.2.8 L-Reactor Seepage Basin (LRSB) Remedial Actions

The L-Reactor Seepage Basin (LRSB) was closed under a final Plug-In ROD remedial action (WSRC 2001; WSRC 2004e). In situ stabilization/solidification of the basin's contaminated soil was determined not to be necessary for the LRSB due to the low levels of radionuclide contamination. During 2002 and 2003, as part of the final ROD remedial action for the LRSB (WSRC 2004e), pipelines associated with the basin were grouted in place or grouted, excavated, and placed in the basin; and a soils cap was installed over the basin. The 450-foot long, 3-inch diameter, HDPE pipe from the Disassembly Basin to the basin was determined to be free of leaks and contamination (WSRC 2001), and therefore was grouted in situ and left in place (WSRC 2004e). The 250-foot long, six-inch diameter, carbon steel pipe located between the LRSB and L-Area Oil & Chemical Basin (LAOCB) was never used and was determined to be free of contamination (WSRC 2001). However, this pipeline was grouted, excavated, and placed in the basin, due to the concern that left in place it could provide access to the basin (WSRC 2001; WSRC 2004e). The above describes the LRSB end-state, which appears similar to that of the PRSBs.

8.2.7.2.9 L-Reactor Seepage Basins Inventory and Basis for Estimate

During sampling and analysis of soils beneath the LRSB the maximum detection of the following radionuclides was greater than twice the average background concentration for that radionuclide within the basin: Na22, K40, Co57, Co60, Zn65, Sr90, Tc99, Cs137, Pm147, Pb212, Ra228, Th230, Th234, U234, U238, Pu238, and Pu239/240 (WSRC 2001, Appendix A.5). These radionuclides are considered contaminants due to seepage basin operation. Of three inventory estimates provided for the LRSB (Holmes et al. 1983; ERD 2001; WSRC 2000b), ERD (2001) estimated the greatest inventory of Co60 at 0.33 curies, decay corrected to 2001; and WSRC (2002d) estimated the greatest inventory of Sr90 and Cs137 at 1.31 and 0.837 curies, respectively, decay corrected to 2001. These inventories form the basis for the LRSB inventory, shown in Table A-60.

The inventory for the other radionuclides, decay corrected to 2001, was determined based upon the inventory of Cs137 (0.837 curies) and the ratio between the radionuclide's total soil concentration and that of Cs137 within the soils beneath the basin as documented within WSRC 2001 (Appendix A.5). Where inventory estimates combined two radionuclides, such as Pu239/240, the inventory for each individual radionuclide in the combination was assumed to be the same as the given combined inventory. The tritium that was discharged to the basins is not included in the Table A-60 inventory, because the tritium has already migrated from the basins to form a groundwater plume, whose inventory has been evaluated separately.

8.2.7.2.10 C-Reactor Seepage Basin (CRSB) Remedial Actions

The C-Reactor Seepage Basins (CRSBs) were closed under a final Plug-In ROD remedial action (WSRC 2003e). During 2001 and 2002, as part of the final ROD remedial action for the CRSBs (WSRC 2003e), the contaminated soils of Basin #1 were stabilized/solidified; the pipeline from the Disassembly Basin to the CRSBs was grouted in situ and left in place; and a soils cap was installed over all the basins.

In situ stabilization/solidification of the contaminated soils in Basin #1 consisted of soil mixing to a depth between four and six feet using a dual auger soil mixing rig to blend the prescribed amounts of cement, bentonite, and water with the basin soils. The soil stabilization/ solidification mixture utilized for CRSB Basin #1 was the same as that for the KRSB. Grouting of the three-inch diameter polyvinyl chloride (PVC) pipeline consisted of pumping controlled low strength material (CLSM) into the 739 foot pipeline and leaving it in place. The above describes the CRSB end-state, which appears similar to that of the PRSBs.

8.2.7.2.11 <u>C-Reactor Seepage Basins Inventory and Basis for Estimate</u>

During sampling and analysis of soils beneath the CRSBs, the maximum detection of the following radionuclides was greater than twice the average background concentration for that radionuclide within Basin #1 or where background concentrations were not available, the radionuclide was detected in more than a third of the samples: C14, K40, Co60, Ni63, Sr90, Zr95,Cs137, Pm147, Eu152, Eu154, Eu155, Ra226, Ra228, Ac228, U235, U238, Pu238, Pu239/240, Am241, Am243, and Cm243/244 (WSRC 1998a, Appendix E). These radionuclides are considered contaminants due to seepage basin operation. WSRC 1997c (Table 1) provides an inventory within the combined CRSBs, based upon the estimated total release to the basins, for the following radionuclides: S35, Cr51, Co60, Sr89, Sr90, Zr95/Nb95, Ru103/106, Sb124/125, Cs134, Cs137, Ce141/144, and Pm147. This inventory is decay corrected to 1997.

The WSRC 1997c inventory is greater than the inventory estimates from three other sources (Holmes et al. 1983; WSRC 1998a; ERD 2001). Therefore, the inventories documented in WSRC 1997c for the radionuclides provided (i.e., S35, Cr51, Co60, Sr89, Sr90, Zr95/Nb95, Ru103/106, Sb124/125, Cs134, Cs137, Ce141/144, and Pm147) were used in the CRSBs inventory documented herein. The inventory for the other radionuclides not provided in WSRC 1997c (i.e., C14, K40, Ni63, Eu152, Eu154, Eu155, Ra226, Ra228, Ac228, U235, U238, Pu238, Pu239/240, Am241, Am243, and Cm243/244) was determined based upon the inventory of Cs137 provided in WSRC 1997c (i.e., 1.2 curies) and the ratio between the radionuclide's average soil concentration and that of Cs137 within the soils beneath Basin #1 as documented within WSRC 1998a (Appendix E). Inventories for these other radionuclides had to be calculated in this manner, since no inventories for them were available from any of the above referenced reports. Where inventory estimates combined two radionuclides, such as Pu239/240, the inventory for each individual radionuclide in the combination was assumed to be the same as the given or calculated combined inventory. The tritium that was discharged to the basins is not included in the inventory, because the tritium has already migrated from the basins to form a groundwater plume, whose inventory has been evaluated separately. Table A-61 provides the 1997 radionuclide inventory for the CRSBs.

8.2.7.3 Reactor Cask Car Railroad Tracks

8.2.7.3.1 Background

Irradiated fuel and target assemblies were loaded into lined casks within the disassembly basin of each reactor building (P, R, C, K, and L) during reactor operations. The casks were transported by rail to either the F- or H-Area Separations Areas for chemical processing. Site procedures required radiological screening of cask cars prior to entering or exiting the reactor building, as well as screening the tracks after the movement of cars. These procedures were designed to minimize the potential for release of radiological materials onto the railroad lines. However even with these precautions some radiological material associated with the casks has contaminated the reactor cask car railroad beds and tracks (WSRC 2008e, WSRC 2008i).

The Early Action Proposed Plan (WSRC 2008e) for the P-Reactor cask car railroad tracks proposes eliminating a radiologically contaminated hot spot (high contamination area, or HCA) on the rail bed material along approximately 60 feet of the cask car railroad tracks (mostly Cs137, 200 pCi/g), covering a 600 square foot area. Cs137 is present in SRS soils above 1 pCi/g due to fallout from nuclear testing, making it difficult to determine areas impacted due to SRS operations. At this location, Co60 is detected with Cs137. Because Co60 is easily detected and not present in SRS soils as background, proposed cleanup includes removal of contaminated media that exceed a risk-based concentration of 0.0596 pCi/g for Co60. This would ensure removal of Cs137 contaminated media also. The waste disposition location could be onsite, offsite, or within the 105-P reactor building. The final decision will be documented in the early action record of decision (WSRC 2008e). It will be assumed that similar remedial actions will be taken for each of the other reactor cask car railroad tracks (R, C, K, and L).

8.2.7.3.2 P-Reactor Cask Car Railroad Tracks Inventory and Basis for Estimate

During sampling and analysis of the P-Reactor cask car railroad bed material (WSRC 2007j) the maximum detection of ten radionuclides was determined to be greater than twice their average background concentration. These radionuclides are assumed to be contaminants resulting from reactor operations. Additionally the sampling and analysis indicated that approximately 60 linear feet of the railroad bed was a radiologically contaminated hot spot (HCA), resulting in an estimate of 21 cubic yards of contaminated materials to be excavated from the P-Reactor cask car railroad tracks (WSRC 2008e). The radionuclide inventory for each of the ten radionuclides has been determined based upon their average concentration within the radiologically contaminated hot spot and the estimated 21 cubic yard volume of contaminated material. Table A-62 provides the 2007 radionuclide inventory for the P-Reactor cask car railroad tracks.

8.2.7.3.3 R-Reactor Cask Car Railroad Tracks Inventory and Basis for Estimate

A total gamma and Cs137 survey was performed along the entire length of railroad gravel inside the R-Reactor fence to identify all "hotspot" areas along the railroad tracks. Two "hotspot" areas of Cs137 contamination were detected along the railroad tracks. Area 1 is a larger contamination area along the tracks to the south of the disassembly basin, and Area 2 is a smaller contamination area along the tracks to the west of the disassembly basin. Forty-two samples were collected from the ground surface to typically two feet below ground surface. During sampling and analysis of the R-Reactor cask car railroad bed material, the maximum detection of eleven radionuclides was determined to be greater than twice their average background concentration. These radionuclides are assumed to be contaminants resulting from reactor operations. Based upon the data, the total volume of Cs137 contamination greater than 10 pCi/g (the maximum background soil concentration of Cs137 is 3.3 pCi/g) was estimated to be approximately 15,892 cubic feet (WSRC 2008i). The radionuclide inventory for each of the eleven radionuclides has been determined based upon their average concentration within the radiologically contaminated hot spots and the estimated 15,892 cubic feet volume of contaminated material. Table A-63 provides the 2007 radionuclide inventory for the R-Reactor cask car railroad tracks.

8.2.7.3.4 C-, K-, and L-Reactor Cask Car Railroad Tracks Inventory and Basis for Estimate

While the 105-C building has been declared a surplus facility and listed as a CERCLA facility to be decommissioned consistent with the FFA, essentially no area specific characterization has been performed in support of estimating an inventory for its cask car railroad tracks. Additionally, the KAC and LAC buildings are currently operating facilities and have not been declared surplus and are therefore not yet classified as CERCLA facilities requiring decommissioning consistent with the FFA. Due to being operating facilities essentially no area specific characterization has been performed in support of estimating an inventory for their cask car railroad tracks. Therefore the inventories of the C-, K-, and L-Reactor cask car railroad tracks will be taken as that of the P-Reactor cask car railroad tracks.

This is considered reasonable for two reasons:

1) The P-Reactor was operated for approximately 35 years and the C-, K-, and L-Reactors operated for 35 years or less (approximately 31, 35, and 19 years, respectively, for C-, K-, and L-Reactors).
2) Using the inventory of the reactor area seepage basins as an indication of the relative level of radionuclides released to the environment would indicate that the P-Reactor cask car railroad track inventory would bound that of C-, K-, and L-Reactors. The P-Reactor seepage basins Cs137 inventory has been estimated at 93 curies (see Table A-58), whereas that of the C-, K-, and L-Reactor seepage basins is all less than 5 curies each (see Table A-61, Table A-59, and Table A-60, respectively).

Therefore the 2007 radionuclide inventories for each of the C-, K-, and L-Reactor cask car railroad tracks can be taken as that provided in Table A-62.

8.2.7.4 P-Reactor Outfalls P02 and P007

8.2.7.4.1 Background

The P02 Outfall is located northeast of P Area, outside the perimeter fence. Early aerial radiological surveys indicated a localized area of elevated gamma activity covering about 0.16 acres in the upper outfall reaches (WSRC 2007j, WSRC 2008e).

The P007 Outfall is just north of the P-Area Ash Basin Investigative Unit. Early aerial radiological surveys indicated a localized area of elevated gamma activity covering about 0.9 acres. Cs137 and Co60 were deposited in the ash at the outfall from a possible process sewer line cross-connected to a storm sewer that originates from the disassembly basin floor drains (WSRC 2007j, WSRC 2008e).

The Early Action Proposed Plan (WSRC 2008e) proposes eliminating radiologically contaminated soil at P02 and P007 outfalls. Cs137 is present in SRS soils above 1 pCi/g due to fallout from nuclear testing, making it difficult to determine areas impacted due to SRS operations. At these two outfalls, Co60 is detected with Cs137. Because Co60 is easily detected and not present in SRS soils as background, proposed cleanup includes removal of contaminated media that exceed a risk-based concentration of 0.0596 pCi/g for Co60. This would ensure removal of Cs137 contaminated media also. The waste disposition location could be on site, off site, or within the 105-P reactor building. The final decision will be documented in the early action record of decision (WSRC 2008e).

8.2.7.4.2 Inventory and Basis for Estimate

During sampling and analysis of the P02 outfall, the maximum detection of one radionuclide was determined to be greater than twice its average background concentration. This radionuclide, Tl208, is assumed to be a contaminant resulting from reactor operations (WSRC 2007j). During sampling and analysis of the P007 outfall, the maximum detection of ten radionuclides was determined to be greater than twice their average background concentration. These radionuclides are assumed to be contaminants resulting from reactor operations (WSRC 2007j).

The Early Action Plan for the P-Area Operable Unit (WSRC 2008e) provides volume estimates for the P02 and P007 outfalls of 411 cu yd and 4356 cu yd, respectively, for the volumes of contaminated ash to be removed. The 2007 radionuclide inventories for the P02 and P007 outfalls are provided in Table A-64 and A-65, respectively. These inventories have been determined based upon the average concentrations of the radionuclides within the radiologically contaminated areas to be removed, the volume of ash to be removed, and the assumed unit weight of the ash (i.e., 108.4 pounds per cubic foot, pcf) (Alleman et al. 1996).

8.2.8 S-Area Facilities

8.2.8.1 Defense Waste Processing Facility (DWPF)

8.2.8.1.1 Background

The DWPF converts the liquid nuclear waste currently stored at the SRS into a solid glass form suitable for long-term storage and disposal. It is the largest such plant in the world. By immobilizing the radioactivity in glass, the DWPF reduces the risks associated with the continued storage of liquid nuclear wastes at SRS and prepares the waste for ultimate disposal in a federal repository. Construction of DWPF began in late 1983, and it began radioactive operations in March 1996. The end of operations is expected to occur in 2030 and therefore the date at which the End State is reached is assumed to be 10/1/2031 (WSRC 2007k).

Waste feed to the DWPF originates in the High Level Waste Tanks at the SRS. The liquid nuclear waste in tank storage exists in two forms, a sludge form and a salt form. The sludge form, while comprising only about 10% of the volume in the tanks, contains about half of the radioactive material. DWPF is designed to treat the highly radioactive radionuclides from both forms of waste; however, DWPF treatment of those from the salt waste awaits the startup of new pre-treatment facilities, including the Modular Caustic Side Solvent Extraction Unit and Actinide Removal Process and the Salt Waste Processing Facility.

The DWPF includes multiple buildings and components that are considered individually when determining the eventual end-state of the facility and estimating the residual radionuclide inventory. Individual components include: Building 221-S; Low Point Pump Pit (LPPP) Building; Glass Waste Storage Buildings (GWSBs); S Area Sandfilter; Organic Waste Storage Tank (OWST); Failed Equipment Storage Vault (FESV); Used Melters; and the Canister Shipping Facility. One facility, the DWPF Recycle Evaporator, is yet to be built and will be added to the CA inventory accounting at a later date, if needed.

8.2.8.1.2 Inventory and Basis for Estimate

No single End State is expected to apply to every component of the DWPF and, hence, different assumptions were made for each in estimating an End State radionuclide inventory. The general approach for estimating the closure inventory for DWPF was to estimate the amount (volume) of material to remain at the End State for each component and multiplying that by an estimated concentration. The anticipated End State of each of the DWPF component facilities and the assumed quantity of residual material is presented in Table A-66. The inventory of the transfer line between HTF and DWPF has been included in the HTF ancillary equipment transfer line inventory provided in Table A-38.

It should be pointed out that for the S-Area Sandfilter that it is possible that some hydraulic washing of the filter material could possibly occur as part of the closure process. Also, it is not known at this time how effective any grouting of the facility would be because much of the material is fine sand, and grout may not penetrate it. Therefore, the assumption of a residual of ten gallons of sludge material is a conservative estimate of the eventual residual radionuclide inventory expected to remain at the End State for this facility. The 2031 radionuclide inventory estimate using the assumed quantities of residual waste material and the appropriate radionuclide concentrations and inventories is presented in Table A-67.

8.2.9 T-Area (Former TNX Area) Facilities

T Area is located at SRS 0.4 km (0.25 mi) east of the Savannah River on a terrace between Upper Three Runs Creek to the north and Fourmile Branch to the south (WSRC 2003g). The former TNX Area was a pilot-scale testing and evaluation facility that supported fuel and target manufacturing, chemical processes, and the Defense Waste Processing Facility (WSRC 2003g).

During operational activities at the former TNX Area, radionuclides were present in a variety of facility buildings, piping, sumps, and wastewater discharge areas. During D&D activities all of the buildings and laboratories located in the former TNX Area were demolished to the slab and the demolition debris was removed from the area with the following exceptions: the 678-5T pump test facility and ancillary structures, the 702-T telecommunications building, the 906-T air stripper and a Soil Vapor Extraction system (WSRC 2007L). Soil and debris that did not meet specific closure criteria were removed and disposed offsite. Some radionuclide contaminated soils and ancillary equipment (sumps, trenches, slabs) remain onsite and are covered by a closure cap. Based on review of D&D and closure documentation, the following areas have been identified as containing residual radionuclide inventories at the former TNX Area:

- TNX Burying Ground (TBG)
- Old TNX Seepage Basin (OTSB)/Inactive Process Sewer Line (IPSL)
- TNX Discharge Gully (DG)
- TNX Outfall Delta (OD) & Swamp
- Buildings 677-T and 678-T Sumps & Slabs & OD Removal Action Soil

Certain areas remain uncovered due to engineering constraints (e.g., swamp). The TBG, OTSB/IPSL, DG (upper and lower), and the TNX Industrial Area, which contains the Building 677-T and 678-T sumps and slabs and the relocated OD Removal Action soil, are covered by a low-permeability closure cap (WSRC 2007L, Figure 9).

8.2.9.1 TNX Burying Grounds

8.2.9.1.1 Background

The TBG was created in 1953 after the accidental explosion of an experimental evaporator that contained 590 kg (1,300 lb) of uranyl nitrate. Debris from the explosion was buried in three known trenches and one suspected trench. The exact depths of these trenches are not known. This debris included materials such as conduit, drums, and structural steel. Between 1982 and 1984, during an expansion of the TNX Area facilities, a portion of the buried material was excavated and sent to the SRS Radioactive Waste Burial Ground (WSRC 1997d). Buildings were constructed over the TBG during the remaining operational phase of the area. These buildings were removed during D&D activities, some sumps and ancillary equipment (trenches, piping) were filled with concrete, and a low-permeability closure cap was installed over the industrial portion of the former TNX Area.

8.2.9.1.2 Inventory and Basis for Estimate

The methodology for the TBG consisted of first calculating a total soil volume using a given inventory of U235/238 and then using that volume and the reported average concentrations for each nuclide to calculate the proportion of each nuclide present in the total volume of soil remaining in the waste unit. The RFI/RI for the TNX OU (WSRC 1997d) contains sampling data for the radionuclides in the TBG. The RFI/RI sampling data are presented as an average concentration for each radionuclide detected above two times the average background concentration for that radionuclide. This data was used in conjunction with a residual inventory for U235/238 that was calculated in 2001 (ERD 2001) to determine inventories for each nuclide reported in the RFI/RI sampling data. For this analysis it was assumed that the 0.1 Ci inventory of U238/U235 estimated in 2001 came from the same volume and mass of soil that resulted in the given inventory for U235/238. Table A-68 provides the 1997 remaining TBG radionuclide inventory.

8.2.9.2 Old TNX Seepage Basin/Inactive Process Sewer Line (OTSB/IPSL)

8.2.9.2.1 Background

The OTSB was an unlined excavation designed to remove wastewater solids (in the inlet section) and contain wastewater (in the seepage section) until it could infiltrate into the underlying sediments that were believed to act as natural ion exchange media. The basin was operated from 1958 to 1980. During the 22-year operating period, overflow from the basin resulted in the creation of a gully (the DG) on its eastern end. The incision of the gully resulted in the formation of the OD inside the eastern side of the TNX Swamp. The OTSB received a number of chemicals ranging from inorganic salts and low-level radionuclides to organic solvents through a series of process sewer lines originating from Buildings 677-T and 678-T. These lines are now inactive and are designated the IPSL (WSRC 1997d).

In 1981, the west wall of the basin was breached to drain the impounded liquids to the nearby swamplands; the basin was backfilled with sand and clay, and then covered with clay. A portion of the cover was revegetated and the remainder was covered with asphalt (WSRC 1997d).

During final closure in 2005, the OTSB was excavated to approximately 2.4 m below the original basin surface and the soil removed was disposed offsite. The basin was backfilled with clean material and a closure cover installed. The IPSL was excavated, ends capped, excavation areas backfilled and covered with the closure cap as well (WSRC 2006h).

8.2.9.2.2 Inventory and Basis for Estimate

The RFI/RI for the TNX OU (WSRC 1997d) contains sampling data for the radionuclides in the OTSB/IPSL. The sampling data is presented as an average concentration for each radionuclide detected above two times the average background concentration for that radionuclide. A 1997 total OTSB/IPSL inventory estimate for each radionuclide, detected greater than two times its average background concentration, was determined based upon its average concentration and the estimated volume of contaminated soil in the basin (approximately 24 m by 53 m by 7 m). Total inventories calculated for isotopes reported in combination in the sampling data (e.g., U233/234) were conservatively assigned to each individual isotope from a combination.

Because contaminated soil was removed from the OTSB/IPSL during final closure and disposed at an approved facility outside the SRS, the inventory contained in that soil was estimated as outlined above and subtracted from the 1997 total OTSB/IPSL inventory previously calculated in order to produce the 1997 remaining OTSB/IPSL inventory. This volume was estimated using the basin dimensions (approximately 24 m by 53 m) and soil removal depths described in WSRC (2006h).

Table A-69 provides the 1997 remaining OTSB/IPSL radionuclide inventory.

8.2.9.3 TNX Discharge Gully

8.2.9.3.1 Background

During the 22-year operating period of the OTSB, overflow from the basin resulted in the creation of a gully (the DG) on its eastern end (WSRC 1997d). During final closure in 2005, the DG was cleared and backfilled with clean soil then covered by the closure cap. Any soil or debris removed in preparation for final cover installation was considered to be uncontaminated, was relocated to the upper TNX industrial area, and remains beneath the closure cap in that location (WSRC 2006h).

8.2.9.3.2 Inventory and Basis for Estimate

The RFI/RI for the TNX OU (WSRC 1997d) contains sampling data for the radionuclides in the DG. The sampling data is presented as an average concentration for each radionuclide detected above two times the average background concentration for that radionuclide. Sample data is divided into two separate locations, the Upper DG and the Lower DG. The number of curies present in the DG for each radionuclide was estimated using an estimated soil volume, an assumed soil bulk density, and the average soil data from the RFI/RI. The Upper DG and Lower DG areas were estimated separately and then added together. Total inventories calculated for isotopes reported in combination in the sampling data (e.g., U233/234) were conservatively assigned to each individual isotope from a combination. Table A-70 provides the radionuclide inventory for the TNX DG.

8.2.9.4 TNX Outfall Delta & Swamp

8.2.9.4.1 Background

The TNX Swamp was used to routinely manage surface runoff and storm water. The swamp was divided into four areas, the OD, the Inner Swamp (IS), the High Ground Swamp, and the Outer Swamp. Portions of the swamp are flooded periodically, and the water table is very near the land surface most of the year. The High Ground Swamp is a small alluvial terrace that separates the floodplain into the IS and the Outer Swamp. It also serves as the bank of the Savannah River during high stages. This area is flooded less frequently than the adjacent lowlands. Periodically, overflows of process wastewater from the upgradient OTSB would impact the area (WSRC 2007L). During a removal action prior to final closure, contaminated soil was removed from the OD/IS and relocated to the upgradient industrial section of the former TNX Area for placement under the final closure cap. Following the removal action, apatite was applied to areas of the OD and IS where radionuclide (uranium) concentrations exceeded the action limit (WSRC 2007L).

8.2.9.4.2 Inventory and Basis for Estimate

The RFI/RI for the TNX OU (WSRC 1997d) contains sampling data for radionuclides in the OD and the Swamp High Ground. The available documentation presented conflicting information regarding the sampling locations that yielded the data labeled Swamp High Ground. For this evaluation, based on historical knowledge of swamp area flooding and topography, it is assumed that the samples are representative of the entire Swamp. It is unlikely that the data represents the High Ground Swamp in that this area is not prone to the periodic flooding that transported the contaminants from the OD. Considering the data as representative of the entire swamp is a conservative assumption. The sampling data is presented as an average concentration for each nuclide detected above two times the average background concentration for that nuclide.

The estimated radionuclide inventory remaining in the OD and Swamp were calculated separately for each area and then added together. For the OD, the estimated soil volume was reduced by the amount documented to have been excavated and relocated to the upper industrial area for disposition under the low-permeability soil cover (the OD and Swamp remain uncovered following final closure). Documentation refers to the removal action as the OD Removal Action, however, figures in the report indicate that the soil was actually excavated from within the IS not the OD (WSRC 2007L). Assuming that the soil was removed from the OD as labeled is conservative because it leaves a larger volume of contaminated soil remaining in the IS which is closer to the Savannah River.

The number of curies present for each radionuclide was estimated using an estimated soil volume, an assumed soil bulk density, and the average soil data from the RFI/RI. Total inventories calculated for isotopes reported in combination in the sampling data (e.g., U233/234) were conservatively assigned to each individual isotope from a combination. Table A-71 provides the radionuclide inventory for the TNX OD and Swamp.

8.2.9.5 Buildings 677-T and 678-T

8.2.9.5.1 Background

Building 677-T was constructed in 1950 to house numerous pilot scale experimental facilities of various sizes. The building last contained a maintenance shop, a laboratory for bench-top small-scale demonstration projects, a bio-remediation laboratory, and a process area bay for large-scale pilot plant experiments/demonstrations (WSRC 2005b). Review of D&D, removal action, and closure documentation indicates that some fixed radiological contamination remains on some slab areas which have been painted prior to closure. Additional residual contamination remains in area sumps which may have been exposed to depleted uranium solutions during operations (WSRC 2004f).

The building was decommissioned in 2003 and 2004. The above-ground portion of the structure was demolished to grade elevation and removed, all interfacing utilities were isolated, disconnected, and plugged; and sumps and other floor openings greater than two inches were filled with concrete to achieve a uniform surface. Anchor bolts have been cut off flush with the floor. The slab on which the facility rested and the outside tank foundations remain under the T-Area low-permeability closure cap. All sumps have been filled with concrete (WSRC 2005b).

Building 678-T was constructed in 1950. It was one of the original buildings in TNX. It was a steel-braced, frame building with corrugated steel wall and roof panels. It was erected on a concrete pad and included two large process areas, four first floor offices and a second floor laboratory. The north section was erected in 1952; a second section (South Annex) was erected in 1978. The 1st floor – approximately 15,700 square feet, included a high bay area and ancillary equipment around the building. The process area in the north section held a tank gallery. In 1953 an evaporator exploded, spreading uranyl nitrate throughout the area and damaging the east side of the building. The building was repaired and evaporator work continued for several more years. In the 1980s an ashcrete test facility and the Off-Gas Test Facility (a 1/10th scale model) for the Consolidated Incineration Facility were built in the South Annex.

The decommissioning of Building 678-T was completed in 2004. The building was demolished to grade, utilities were isolated, and sumps and other floor openings greater than two inches in diameter were filled with concrete (except for the west pit sump, which was filled with gravel). The trench sump was a stainless steel sump that was in good physical condition and radiological surveys indicated high contamination levels on the sump walls prior to removal. There was no concrete under the sump, only soil. Radioactive contamination was identified in the soil. Following SDD spill cleanup guidelines, radiological readings were taken of the soil and "hot spots" removed. Cleanup was unsuccessful as contamination continued into the soil column.

Cleanup activities did not reduce the radiological level in the sump. The radiologically contaminated soil identified beneath the trench sump was outside the scope of the facility decommissioning project, therefore, SGCP was notified. Concrete was placed into the resulting hole. The floor and tank foundations in the process area were painted to fix contamination (WSRC 2005b).

No additional soil or equipment removal is noted in the closure reports reviewed for this inventory estimate. The pad on which the facility was constructed was left in place and remains under the low-permeability closure cap that was installed over the upper industrial portion of the former TNX Area (WSRC 2007L).

8.2.9.5.2 Inventory and Basis for Estimate

Sumps and slabs inventory estimates were calculated using material volumes and radionuclide concentrations reported in the investigation and D&D documentation (Noller and Haynes (2005) and WSRC (2004f)) and assumed bulk densities. Data for sumps and slabs were reported separately; therefore, estimates were calculated separately for these areas and added together. Additionally the soil removed from the OD as discussed in Section 8.2.9.4.2 was also added to the Building 677-T and 678-T radionuclide inventory. Total inventories calculated for isotopes reported in combination in the sampling data (e.g., U233/234) were conservatively assigned to each individual isotope from a combination.

Table A-72 provides the 2004 radionuclide inventory for Building 677-T Sumps and Slabs, Building 678-T Sumps and Slabs and in the OD Removal Action Soil that was disposed in the vicinity of the two buildings.

8.2.10 Z-Area Facilities – Saltstone Facility

The Saltstone Facility is made up of the Saltstone Processing Facility and the Saltstone Disposal Facility. The Saltstone Processing Facility contains processing equipment to incorporate the salt waste solution into saltstone grout, which is pumped into the Saltstone Disposal Facility vaults. The Saltstone Processing Facility will be decommissioned to leave no residual radionuclide inventory. Therefore, this section deals only with the Saltstone Disposal facility.

8.2.10.1.1 Background

To estimate the material to be disposed within the SDF, the source of the material to be disposed needs to be identified. The only source is the material currently stored in the Tank Farm waste tanks and additional material received from the H-Canyon. This material in the Tank Farms will be disposed in one of three locations, Saltstone, the DWPF, or remain in the tanks.

The DOE plans call for stabilizing and disposing of the high-level waste in a deep geologic repository for spent nuclear fuel and high-level radioactive waste. This will be done by stabilizing the high-level waste in a borosilicate glass matrix through vitrification in a facility known as the DWPF. This process has been on-going since 1996. As for the salt waste: DOE contemplates removing fission products and actinides from these materials using a variety of technologies, combining the removed fission products and actinides with the metals being vitrified in DWPF, and solidifying the remaining low-activity salt stream into a grout matrix known as saltstone grout suitable for disposal at the SDF at the SRS (DOE 2006).

8.2.10.1.2 Inventory and Basis for Estimate

To determine the material for disposal at Saltstone, a starting point of the Tank Farm inventory was used. Then the radionuclides were removed based on treatment processes' estimates. The remaining inventory was assumed to be sent to Saltstone and was distributed among the actual and projected disposal units.

Saltstone presently has material disposed in two vaults (1 and 4). Plans do not include disposing of additional material in Vault 1. Vault 4 is being used to dispose material and plans include continuing to add material. The current plan was used to estimate the additional material to be added to Vault 4 (WSRC 2007k).

Once the Vault 4 inventory was estimated, it was removed from the Tank Farm inventory. The remaining inventory was adjusted to account for the removal of treated radionuclides. This resulting inventory was distributed amongst the additional disposal units (SRNS 2008a). The basis for the estimated closure inventory in the Inventory report was updated in 2009 (SRNS 2009), and is reflected in the CA inventory. The inventory of the transfer line between HTF and SPF has been included in the HTF ancillary equipment transfer line inventory provided in Table A-38.

The system plan calls for the last disposal unit to be filled at the end of the fiscal year 2030. Therefore all the inventories have been decay corrected to October 1, 2030. The 2030 radionuclide inventories for Saltstone Vault 1, Saltstone Vault 4 and for the remaining 64 anticipated Saltstone Vaults are presented in Table A-73, A-74 and A-75, respectively.

8.2.11 General Site Area Facilities

8.2.11.1 Integrator Operable Units

8.2.11.1.1 <u>Background</u>

Integrator Operable Units (IOUs) are defined as surface water bodies (e.g., streams and lakes) and associated wetlands, including sediment/soil, sediment, and related biota. They represent the integration of potential contamination discharged to surface water or migrating through groundwater from source OUs, site evaluation areas, National Pollutant Discharge Elimination System (NPDES) outfalls, and operational facilities to points of potential receptor exposure. Similarly, all sources of residual radioactive material at the SRS End State, including certain closure units, decommissioned facilities and LLW disposal facilities which contain significant quantities of radioactive constituents may potentially discharge to these integrating units over the CA period of compliance (1,000 years). There are seven IOUs at the SRS which correspond to the surface water drainages. These are listed below and are shown in Figure 8-1:

- Fourmile Branch IOU
- Lower Three Runs IOU
- Pen Branch IOU
- Savannah River and Floodplain Swamp IOU
- Steel Creek IOU
- Upper Three Runs IOU

Background information for each of these IOUs has been compiled for the Fourmile Branch IOU in WSRC (2000c); for the Lower Three Runs IOU in WSRC (2001a); for the Pen Branch IOU in WSRC (2001b); for the Savannah River and Floodplain IOU in WSRC (2000d); for the Steel Creek IOU in WSRC (1999a); and for the Upper Three Runs IOU in WSRC (2003h).

Radionuclides associated with IOUs are adsorbed to the streambed sediment and soils of the shallow floodplains that lie immediately adjacent to stream channels. Soil sampling has been conducted for each of the IOU's by SGCP and is reported in the Periodic Update reports associated with each IOU. In the case of several IOUs, discharge canals are included as a part of the IOU. Examples of this would include the C-Reactor Discharge Canal being considered as part of the Fourmile Branch IOU, and the R-Reactor Discharge Canal being considered as part of the Lower Three Runs IOU. PAR Pond and the other Pond located between PAR Pond and R-Reactor are all considered to be a part of the Lower Three Runs IOU. On the other hand, Tims Branch (a tributary to Upper Three Runs) and Steed Pond (located on Tims Branch) are not considered to be a part of the Upper Three Runs IOU, hence stand alone inventories are reported for each, and have been presented earlier. No remedial actions are planned for IOUs except for restricting access to SRS streams during the period of institutional control.

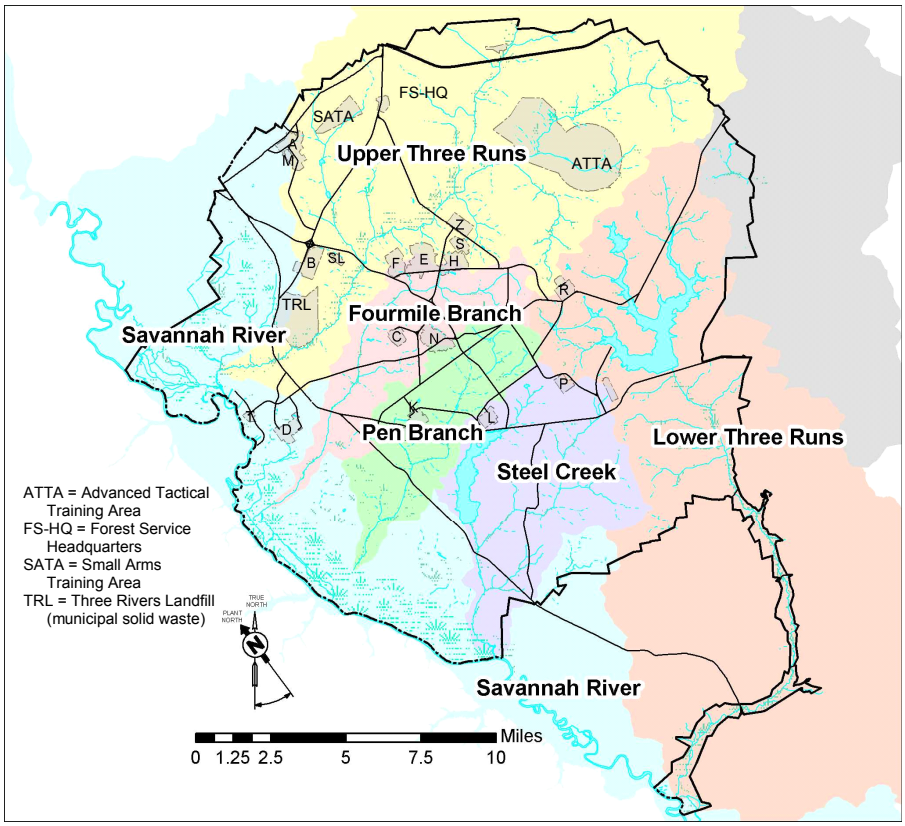

Figure 8-1. Map of Integrator Operable Units at SRS

8.2.11.1.2 Inventory and Basis for Estimate

The radionuclide inventory for Cs137 was obtained from ERD (2001). The basis of that estimate was utilizing analyses conducted on soil and sediment samples collected along and within each IOU and utilizing those results to estimate the total inventory along the entire length of the individual streams. The inventories are reported for 2002, based on the date of the ERD report (December 2001).

The 2002 radionuclide inventory associated with the IOUs is presented in Table A-76. The only radionuclide reported is Cs137, although minor levels of other radionuclides could be present.

8.2.11.2 Groundwater Closure Units

8.2.11.2.1 Background

Groundwater contamination has resulted from the release of certain contaminants from shallow basins and other operating facilities at SRS. Plumes of contaminants existing within the saturated zone extend from a position vertically below the contaminant source areas and extending laterally toward shallow discharge zones, which occur adjacent to nearby streams. These plumes have been monitored with shallow groundwater wells since the late 1970s.

The major plumes of groundwater contamination have been organized into Groundwater OUs. The following Groundwater OUs have been identified in ERD (2001) as the ones which contain a significant quantity of some radionuclide.

- C-Area Reactor Groundwater OU
- D-Area Groundwater OU
- F-Area Groundwater OU
- H-Area Groundwater OU
- K-Area Reactor Groundwater OU
- L-Area Northern and Southern Groundwater OU
- MWMF Groundwater OU
- P-Area Reactor Groundwater OU
- R-Area Reactor Groundwater OU
- T-Area (TNX) Groundwater OU

8.2.11.2.2 Inventory and Basis for Estimate

The only radionuclide mentioned in the SGCP inventory document (ERD 2001) for Groundwater OUs is tritium, which is highly mobile in the subsurface once it becomes part of the water molecule. No explanation is provided as to how the estimate of tritium inventory was achieved, however it is assumed that the investigators utilized groundwater monitoring data to determine average plume groundwater concentration and estimated plume dimensions to determine a quantity of groundwater contaminated at that concentration. In the case of the L-Area Southern Groundwater OU, a more recent estimate of total tritium inventory than is presented in ERD (2001) was available (WSRC 2006i) and therefore the more up-to-date estimate is presented herein. The 2002 radionuclide inventory for each of the Groundwater OUs is presented in Table A-77.

8.2.11.3 Miscellaneous Spills

8.2.11.3.1 Background

A number of spills have occurred at different locations within SRS during the period of operations. Most of these spills are small and contain little, if any radionuclides but others contain some low level of radioactivity. The spills identified in this CA are those which were listed in the previous CA (WSRC 1997).

8.2.11.3.2 Inventory and Basis for Estimate

The 1997 spills and inventories reported in Table A-78 are derived from the previous CA. In Tables 2-1 and 3-83 of the Inventory report (Hiergesell et al. 2008), Tanks 3 and 8 were incorrectly associated with H-Area and Tanks 9, 13, 16, and 37 were incorrectly associated with F-Area; this has been corrected in Table A-78. It is assumed that the basis of these inventories is derived from incident reports. No further characterization information has been obtained to date. Furthermore, it is assumed that the contaminated material is soil.

8.2.12 Future SRS Operational Facilities

8.2.12.1 Salt Waste Processing Facility (SWPF)

8.2.12.1.1 Background

The SWPF will be responsible for decontamination of salt solution from both FTF and HTF. The SWPF will separate the highly radioactive cesium and actinides from the salt solution using caustic side solvent extraction (CSSX) and monosodium titanate absorption/filtration, respectively. After completing the initial separation process, the concentrated cesium and actinide waste will be sent to the nearby Defense Waste Processing Facility where it will be immobilized in a glass matrix and stored in the Glass Waste Storage Building until it can be placed in a geological repository. The decontaminated salt solution will be mixed with grout at the nearby Saltstone facility for disposal onsite.

The assumed End State for this facility under construction is that all the vessels that come into contact with radioactive material will be cleaned, filled with grout to stabilize residual material, and closed in place. The SWPF is expected to be closed 10/1/2031.

8.2.12.1.2 Inventory and Basis for Estimate

The basis of the inventory was to assume a volume of residual material within the vessels at closure and estimate the concentration of radionuclides within this material. The residual volume estimated was based on the suction level of associated transfer piping. These levels are present in construction drawings and process review documentation. The material assumed to remain was estimated using the design material balance for the stream exiting each tank (Parsons 2007). No adjustment was made to these concentrations (i.e., no credit for cleaning that would reduce the material's concentration), which represents a considerable conservatism. No transfer line inventory was assumed associated with this facility. The 2031 inventory for the SWPF is presented in Table A-79.

8.2.12.2 MOX Facility

8.2.12.2.1 Background

Plutonium and nuclear material management missions now being conducted at SRS will be expanded to include materials from dismantled weapons and surpluses from other DOE sites. This mission will support conversion of excess weapons-usable plutonium to a form that can be used in commercial power reactors. Construction of this facility, called the MOX Facility, commenced on August 1, 2007 in F Area at SRS. As designed, the project site would occupy an area of about 16.6 ha (41 acres). Approximately 6.9 ha (17 acres) of the site would be developed with buildings, other facilities, and paving. The remaining 9.7 ha (24 acres) would be landscaped with either grass or gravel.

The MOX facility will consist of the following buildings:

- MOX Fuel Fabrication Building
- Emergency Diesel Generator Building
- Standby Diesel Generator Building
- Secured Warehouse Building
- Administration Building
- Technical Support Building
- Reagents Processing Building
- Receiving Warehouse Building

The MOX facility is being designed and constructed to convert plutonium dioxide and depleted uranium dioxide to MOX fuel. Operations at the facility would begin with the receipt of the plutonium dioxide and depleted uranium dioxide feed materials. The plutonium dioxide would then be purified in the aqueous polishing process before being blended with the depleted uranium dioxide. The blended material would then be formed into pellets, the pellets incorporated into fuel rods, the fuel rods placed in fuel assemblies, and the assemblies loaded into transport casks for shipment to the nuclear power plants authorized to use MOX fuel.

The technology used in the fuel fabrication process includes recycling of waste and scrap streams. Plutonium dioxide powder would be processed at the proposed MOX facility to remove impurities, such as americium and gallium, and would be mixed with the depleted uranium dioxide to form the MOX fuel. The final blend for MOX fuel would have a required plutonium content of 2.3% to 4.8% (percent by weight). The facility would be capable of producing MOX fuel with a plutonium content of up to six% (NRC 2005).

8.2.12.2.2 Inventory and Basis for Estimate

The inventory estimates developed for the MOX Facility were based on the inventories analyzed in the 2005 Environmental Impact Statement (NRC 2005). An End State Reduction Factor of 1E-2 has been applied to approximate the fraction of the facility inventory that might remain with an End State configuration of "In-Situ Disposal." Because the exact fuel blend that is required is specified to be between 2.3 and 4.8 percent, by weight, and that the facility be capable of producing up to 6.0 percent, by weight, the inventory expected to occur for each blend is presented in the inventory table, including the mean of 2.3 and 4.8 percent (NRC 2005). No decay was applied as it was assumed the facility would operate until 2025 and then be closed; hence this inventory is assumed to occur at the SRS End State. The 2025 radionuclide inventory for the MOX facility is provided in Table A-80.

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9.0 MODELING

9.1 CONCEPTUAL MODEL AND MODELING PHILOSOPHY

This section of the Composite Analysis (CA), based on the more detailed report *Conceptual Model for the Savannah River Site's Composite Analysis* (Phifer et al. 2008c), describes the conceptual models addressing release of radionuclides from near-surface and groundwater plume sources (identified in Section 8.0), transport of radionuclides from these sources through the vadose zone and aquifers to surface water, and transport from streambed sources to surface water. This section fulfills CA Tasks 4.2.14, Transport Conceptual Model (Initial), and 4.2.16, Modeling Philosophy, as outlined in the *Comprehensive SRS Composite Analysis Program Plan* (Phifer and Cook 2007) document. The CA conceptual model described herein is consistent with the following:

- Background SRS hydrogeologic information presented in CA Section 3.0 and in Phifer et al. 2008b
- Conceptual models developed for the E-Area Low-Level Waste Facility (ELLWF), F-Tank Farm (FTF), and Saltstone Disposal Facility (SDF) Performance Assessments (PAs) as described in WSRC 2008a, 2008b, and LWO 2009
- Existing Savannah River Site (SRS) groundwater models
- SRS End State Vision (i.e., land use plan) (DOE 2005)

This section outlines the modeling approach that was taken; the model output required; the base case conceptual model, including conceptualization of source releases, that formed the basis of the CA model; the modeling codes that were utilized; input data requirements; and other considerations.

9.1.1 Modeling Approach

A hybrid modeling approach was taken for the CA. A deterministic base case was developed as a reasonably-conservative[1] best estimate dose at each Point of Assessment (POA). See Figure 4-2 for the locations of the POAs. Additionally, alternative sensitivity cases were developed and probabilistic (stochastic) modeling was performed to aid in the interpretation of the deterministic base case.

[1] The term "conservative" is used to describe a situation where insufficient data exists to properly quantify a parameter or to define a sub-set of a model. Assigning a "conservative" selection is a value judgment that is often made on the basis of expert opinion. In such situations it is not practical to lay out all the factors taken into account in the judgment.

The sensitivity cases fell into the following eight categories (see Section 9.4.1) evaluating sensitivity of results to:

- Source release modeling (i.e., accounting for waste forms, barriers, caps, etc.)
- Stream and river flow rates (includes alternative POA)
- Source inventory (inventory required to reach 30 mrem/yr dose)
- Location of the groundwater divide in the General Separations Area (GSA)
- Alternative end state date (i.e., 2025 versus 2050)
- C14 bioaccumulation factor
- Presence of clay in the aquifer path
- Time of assessment (i.e., considering a 100,000-yr period of assessment)

The probabilistic modeling addressed, for each POA and all sources contributing greater than 0.05 mrem/yr to the total maximum dose, uncertainty in results attributed to distributions associated with:

- Distribution (or sorption or partition) coefficients (K_ds)
- Material property values (e.g., soil porosity)
- Infiltration rates for soil and engineered barriers
- Concrete aging
- Stream and river flow rates
- Consumption and human exposure parameters

9.1.2 Model Output Requirements

The 49 parent radionuclides identified as requiring assessment in the CA were modeled (see Table 7-12 in Section 7.2.5). The output of the CA deterministic base case, alternative sensitivity cases, and probabilistic (stochastic) modeling was structured such that an annualized time history Total Effective Dose Equivalent (TEDE) at each applicable POA from each of these parent radionuclides from each source for each pathway was captured. In addition, an aggregated annualized time history TEDE at each POA from all combined radionuclides and sources contributing to dose at each POA was produced. This allowed identification of the radionuclides, sources, and pathways that are projected to contribute most to the annual TEDE over time. The TEDEs were calculated according to the Assessment Periods (APs) identified in the CA Section 4.0 as follows:

- Annual TEDE as a function of time (annualized time history dose) with the maximum dose and time of maximum dose within a 1,000-year AP (beginning in year 2025)
- Maximum TEDE and time of maximum dose over 10,000 years (beginning in year 2025)
- Maximum TEDE and time of maximum dose over 100,000 years (beginning in year 2025)

Annualized time-averaged concentration data for the Savannah River, suitable for use in addressing the dose impact from the use of Savannah River water by downstream users (i.e., Hilton Head, Port Wentworth, and Beaufort-Jasper), was also captured as model output (i.e., concentration data at the U.S. Highway 301 Bridge).

9.1.3 Base Case Conceptual Model

The inventory associated with each source modeled in the CA is provided in Appendix A. The 49 parent radionuclides which were modeled are provided in Table 7-12. Transport modeling of the 49 parent radionuclides and their daughters with a half-life of 3 years or greater was performed explicitly, whereas daughters with half-lives less than 3 years were assumed to be in secular equilibrium with their parent. Three daughters with half-lives greater than three years were identified, such that transport of 52 radionuclides was modeled. The conceptual model for radionuclide release from sources varies based upon the characteristics of the sources and is discussed in more detail in Section 9.1.3.1 below. The conceptual model for radionuclide transport from near-surface sources is one of radionuclide transport by water flow through the vadose and aquifer zones to surface water (i.e., streams and Savannah River) to the POAs, consistent with the transport and pathways screening (Wilhite and Phifer 2008). The conceptual model for radionuclide transport from groundwater plume sources is one of radionuclide transport by water flow through the aquifer zone to surface water (i.e., streams and Savannah River) to the POAs. The conceptual model for radionuclide transport from streambed sources is one of groundwater-recharge-rate-limited release from the streambed sediment to surface water (i.e., streams and Savannah River) to the POAs.

A dose module for the CA production modeling was developed based upon the dose modules utilized within the ELLWF PA (WSRC 2008a), FTF PA (WSRC 2008b), and the CA radionuclide screening (Section 7.2). Doses were calculated for transport and exposure pathways and radionuclides determined to be of interest as a result of the screening process described in Section 7.0 at the POAs identified in Section 4.0, Table 4-2 and Figure 4-2. The recreational dose scenario was evaluated within the mouth of the streams prior to mixing with the Savannah River, whereas the residential dose scenario was evaluated after mixing with the Savannah River. Under non-flood stage Savannah River flow rates, as outlined in Section 0, SRS tributaries to the river flow directly into the river channel and completely mix with the river flow within a few thousand feet (Heffner et al. 1998). Therefore complete mixing in the river was assumed for the residential dose scenario. It is assumed that recreational boaters have access to the mouths of on-site streams. However, residents within the SRS boundary are prohibited by SRS land use plans (Section 4.2.3); therefore residents will not have access to on-site streams or groundwater. The dose module explicitly modeled the 49 parent radionuclides and all their daughters regardless of half-life, therefore negating the need for roll up dose conversion factors (DCFs). Integration of the dose module with the CA transport model was accomplished by importing the activity concentration of the 52 CA radionuclides in the streams and river from the transport module into the dose module. Within the dose module, secular equilibrium was assumed for the shorter-lived daughters (i.e., half-lives less than or equal to three years), by considering relative positions in the decay chains and branching fractions. The dose module then explicitly handles these shorter-lived daughters in calculating human exposure, such that 102 (or 50 additional) radionuclides are considered (Phifer et al. 2009). Additional information concerning the dose module can be found in Section 9.3.2.

Because concentrations for dose modeling were needed only within surface water sources at the SRS boundary (i.e., stream mouths or Savannah River) for the base case, the CA base case model was required to consider only the flux of radionuclides from the source through the vadose and aquifer zones to the streams/river, where concentrations were determined from the radionuclide flux and the flow rate of the stream/river. Conservation of radionuclide mass was maintained throughout the CA model. Figure 9-1 provides a pictorial representation of the detailed CA Base Case Conceptual Model, considering the need to evaluate radionuclide flux (and not concentration) until the streams are reached. Table 9-1 provides descriptive information that amplifies the conceptual model depicted in Figure 9-1.

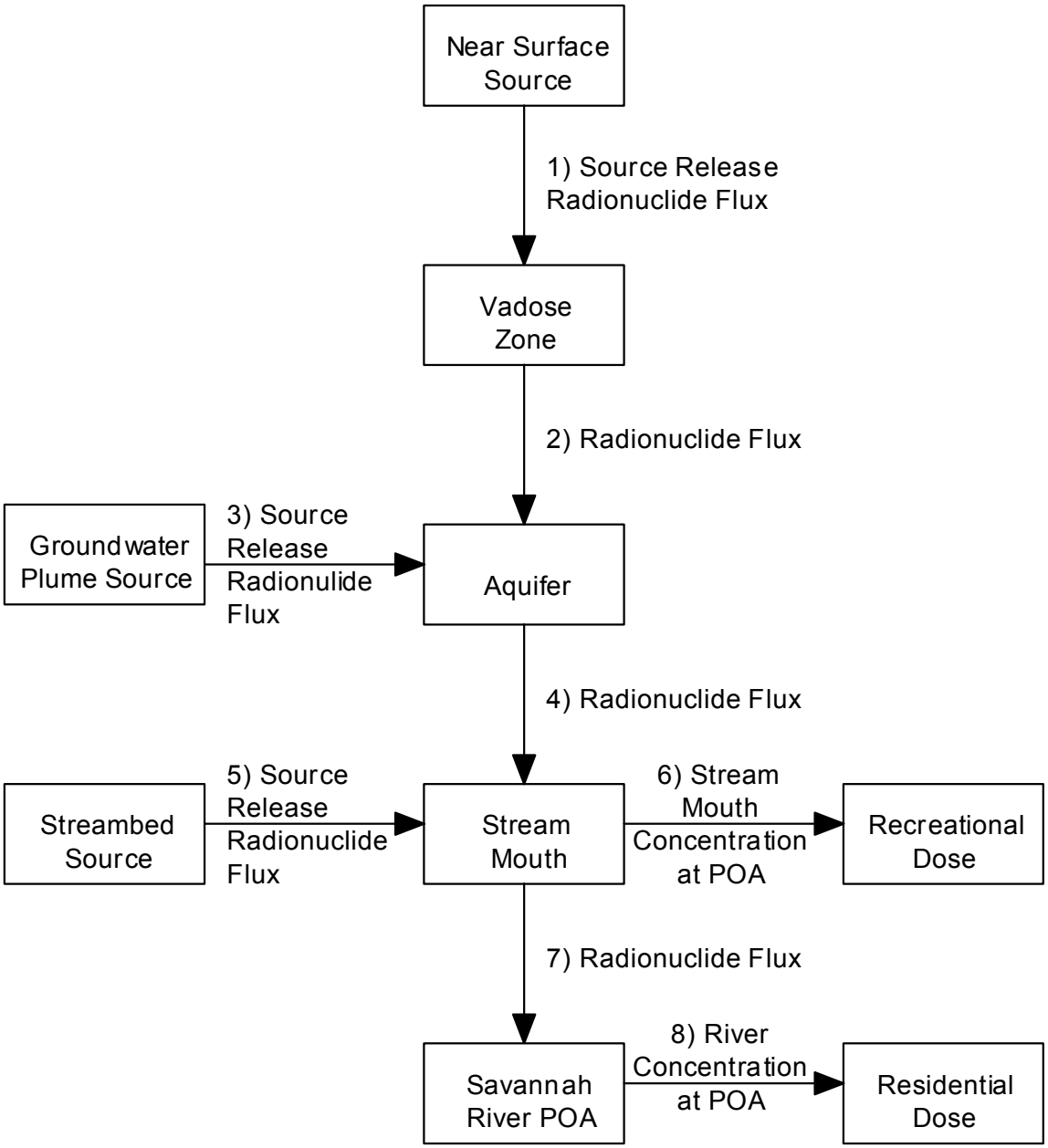


Figure 9-1. CA Base Case Conceptual Model

Notes for Figure 9-1:

(Radionuclide Flux = total curies of each radionuclide per year)
(Concentration = curies/liter [annual average])
Recreational dose is at the stream mouth prior to mixing with the Savannah River
Residential dose is on the Georgia side where the Savannah River bounds SRS and on either side of the river downstream of Steel Creek

Table 9-1. CA Base Case Conceptual Model Description

Conceptual Item	Conceptual Model Description		
1) Source Release Radionuclide Flux from Near-Surface Source to Vadose Zone	Potential Release Mechanisms: • Advection with partitioning • Dissolution or fixed release • Diffusion / leaching • Solubility limited • PA: Where published PA results are available, the PA radionuclide flux to water table will be utilized as the Source Release term within the CA	Source Matrices: • Soil K_ds: sandy or clayey sediment; cellulose degradation product (CDP) correction • Cementitious K_ds: age of oxidizing or reducing cementitious solids • Activated Metal • Surface Contamination	Infiltration (Darcy velocity) to Source: • Cementitious enclosure (in situ disposal (ISD), Vault, Tank, encapsulated, and concrete slab) • Closure cap (clay and geosynthetic) • Backfill (Low Perm Soil, Common backfill) • Background
2) Radionuclide Flux from Vadose Zone to Aquifer	Information needed to model radionuclide flux from the vadose zone to aquifer: • Source release radionuclide flux from near-surface source to vadose zone • Vadose zone thickness (bottom of near-surface source to average water table below source divided into clayey and sandy soil layers when data is available, otherwise sandy soil is assumed) • Vadose zone pore velocity (obtained from infiltration rate and average volumetric water content) • Vadose zone K_ds; it is assumed that concentrations are low enough that K_ds are applicable (i.e., linear sorption/desorption) (Radionuclide Flux = total curies of each radionuclide per year)		

Table 9-1 **CA Base Case Conceptual Model Description** - continued

Conceptual Item	Conceptual Model Description
3) Source Release Radionuclide Flux from Groundwater Plume Source to Aquifer	SRS groundwater plume radionuclide sources consist predominately of tritium. Based upon data availability the following three options exist for modeling these predominately tritium plume sources: 1) Use the estimated groundwater tritium plume inventory at a point in time as the basis to model. 2) Use the measured flux to receiving stream(s) as the basis to model. This is only applicable to plumes that are shrinking rather than expanding. 3) Use the estimated/measured annual tritium discharge to the associated source(s) as the basis to model. This is similar to the modeling conducted for near-surface sources as described above. (Radionuclide Flux = total curies of each radionuclide per year)
4) Radionuclide Flux from Aquifer to Stream	Information needed to model radionuclide flux from the aquifer to the stream mouth: • Radionuclide flux from vadose zone to aquifer • Source release radionuclide flux from groundwater plume source to aquifer • Aquifer flow length and flow time (must consider appropriate effective porosity) • Aquifer pore velocity (obtained from flow length and flow time) • Aquifer K_ds; it is assumed that concentrations are low enough that K_ds are applicable (i.e., linear sorption/desorption) (Radionuclide Flux = total curies of each radionuclide per year)
5) Source Release Radionuclide Flux from Streambed Source to Stream	Source release from streambed sources cannot be modeled as stream flow (advection) with partitioning (i.e., K_d) mechanism, since this would result in the essentially instantaneous release of all streambed sediment bound radionuclides (predominantly cesium or uranium) within the low-concentration, high flow stream water. Therefore a radionuclide groundwater-recharge-rate-limited source release model was developed for the streambeds. (Radionuclide Flux = total curies of each radionuclide per year)

Table 9-1 **CA Base Case Conceptual Model Description** - continued

Conceptual Item	Conceptual Model Description
6) Stream Mouth Concentration	Information needed to model concentration in the stream mouth: • Radionuclide flux from aquifer to stream • Source release radionuclide flux from streambed source to stream • Average total annual flow volume at stream mouth (i.e., average flow rate × year) Instantaneous transfer (i.e., no dispersion) and complete mixing assumed within the streams (consistent with Thermal Infrared over-flight and in-stream tritium data (Heffner et al. 1998)) Conservatively do not consider interaction between water borne radionuclides and streambed sediment as a means to retard radionuclide migration. (Radionuclide Flux = total curies of each radionuclide per year) (Concentration = curies/liter (annual average))
7) Radionuclide Flux from Stream Mouth to Savannah River POA	Information needed to model radionuclide flux from the stream mouth to Savannah River POA: • Summation of radionuclide flux from each stream contributing to the particular Savannah River POA (Radionuclide Flux = total curies of each radionuclide per year)
8) River Concentration at POA	Information needed to model concentration in the river at POAs: • Summation of radionuclide flux from each stream contributing to the particular Savannah River POA • Average total annual flow volume in Savannah River at POA (i.e., average flow rate × year) Instantaneous transfer (i.e., no dispersion) and complete mixing assumed within the Savannah River (consistent with Thermal Infrared over-flight and in-stream tritium data (Heffner et al. 1998)). Conservatively do not consider interaction between water borne radionuclides and streambed sediment as a means to retard radionuclide migration. (Radionuclide Flux = total curies of each radionuclide per year) (Concentration = curies/liter (annual average))

ISD = In-situ disposal

Other required information:

1) Radionuclide half-lives and decay chains; 2) Physical exposure parameters; consumption and human exposure parameters; and bioaccumulation (transfer) factors; 3) Dose conversion factors

9.1.3.1 Base Case Source Release Conceptual Models

A listing of the CA sources corresponding to SRS facilities expected to retain a significant inventory at the end state (Table 8-1 of Section 8.0) is provided in Table C-1 of Appendix C. This listing was derived from Table 8-1 by considering the individual facilities, the groundwater flow paths associated with each source, and the source release mechanism appropriate for the constituents within the source (see Section 9.3.1.6 below). Table C-1 also denotes the applicable inventory table from Appendix A and the projected end state for each source. Where appropriate, consideration was given to combining the inventories of sources for modeling purposes. Such combination of inventories depended on compatible results from the aquifer flow path modeling discussed in Section 9.1.4.1 below.

Sources described in Section 8.0 (and listed in Table C-1) were also classified into one of the following two categories:

- Sources requiring specific source release rate models, and
- Sources that can be adequately represented by a generic CA model

Classification of sources into these two categories was determined as outlined below:

1) The inventories of the 49 CA parent radionuclides and three daughters with half lives greater than three years for each of the facilities/waste sites described in Section 8.0 were run within the "generic" CA transport model with dose module, in order to determine the dose resulting from the facilities/waste site without the use of a specific source release model.
2) Those facilities/waste sites with an aggregated dose from all 52 CA radionuclides of 0.1 mrem/yr or greater were classified as requiring a specific source release model.
3) Those facilities/waste sites with an aggregated dose from all 52 CA radionuclides of less than 0.1 mrem/yr were represented by the generic CA model. The source release within the generic CA model consisted of the inventory being placed on the ground surface, with the underlying vadose zone represented as a soil matrix with an advection/partitioning release mechanism based upon background infiltration (i.e., no cover or cap) and sandy soil K_ds.
4) Streambed and groundwater plume sources required a specific source release model, since they did not fit the representation of the generic CA model described above.

See Table 9-1 for information regarding these types of specific source release models.

Models for near-surface sources classified as requiring a specific source release model were developed based upon the appropriate assignment of potential release mechanisms, source matrices, and infiltration as outlined in Table 9-1, consistent with the end states presented within Table C-1. The inventories of near-surface sources classified as suitable for representation by a generic CA model were represented by the results of the generic CA model run.

Where published PA results were available (ELLWF, FTF, and SDF PAs) the PA radionuclide flux to the water table was used as the source release term within the CA, if the PA source was classified as requiring a specific source release rate model. It was necessary, however, to account for CA radionuclides that were not addressed within the PAs. This was done by utilizing the generic CA model run results for those radionuclides not addressed within the PAs. The ELLWF PA flux results had to be modified based upon the projected inventories, since these fluxes were determined based on a unit curie basis. Since the FTF and SDF PA fluxes were based upon the anticipated inventory, no such correction was required. PA results from the H-Tank Farm (HTF), Transuranic (TRU) Pad 1, and F-Area Material Storage (FAMS) were not available for use within the CA; these sources were simulated using the CA models.

9.1.4 Input Data Requirements

During development of the CA conceptual model, the following required input data was identified:

- Facility/waste site radionuclide inventory
- Infiltration rates
- K_ds and CDP correction factors
- Vadose zone thickness lithology (i.e., sand and clay thicknesses)
- Vadose zone saturation, porosity, density, and tortuosity
- Aquifer flow path characteristics (total flow path length, clay length in flow path, flow velocity, and flow distribution at the vadose zone interface)
- Aquifer porosity, density, and tortuosity
- Average annual flow at stream mouth and in the Savannah River at POAs
- Radionuclide decay chains and branching fractions
- Radionuclide half lives and atomic weights[1]
- Bioaccumulation or transfer factors
- Human exposure parameters and consumption rates

[1] In the GoldSim™ model, the parameter representing the mass of the species of interest is designated as the molecular weight. In this CA, only radionuclides are being considered, therefore this parameter is the atomic weight. However, because the molecular weight designation is in GoldSim™ model, this parameter will henceforth be called the molecular weight instead of the atomic weight.

Since only radionuclide flux must be calculated prior to discharge to the streams, dimensional data (i.e., footprints) for the sources was not necessary within the CA Base Case model for the following release mechanisms (see Table 9-1): advection with partitioning; dissolution or fixed release rate; and diffusion/leaching. However source dimensional data (i.e., footprint) was considered in development of the inventory distributions to the aquifer flow path. Dimensional data was required for one source for which a solubility-limited release mechanism was deemed appropriate.

9.1.4.1 Aquifer Flow Path Input Data Requirements

Flow particle track modeling was conducted using existing models (i.e., PORFLOW, FACT, and MODFLOW) to determine the groundwater flow path length, clay length in the flow path, and flow velocity to the surface water outcrop(s) from each CA facility/waste site through all appropriate aquifers (Water table, Gordon, Crouch Branch, etc.). Table 9-2 provides a summary of the models that were available for potential use in establishing the flow paths.

The following was considered during use of the existing models for determination of the groundwater flow length and time (i.e., velocity):

- For the CA Base Case, groundwater divides were based upon average water table conditions (i.e., the conditions represented in the existing model flow fields).
- Facilities/waste sites on groundwater divides were noted and the proportion of the facility/waste site contributing to flow in each of the groundwater flow directions was estimated based upon the footprint of the facility/waste site. This proportion was used to appropriately apportion the inventory of such facilities to the flow path and receiving stream assuming a uniform distribution of the inventory throughout the site.
- Some large facilities/waste sites (e.g., Reactor Buildings, Burial Grounds, etc.) or facilities/waste sites with multiple subunits (e.g., Reactor Seepage Basins, F & H-Area Seepage Basin, etc.) required multiple particle tracks that are averaged based on similarity of flow paths and receiving stream. However, the bulk of the inventory associated with seepage basin sources consisting of multiple basins typically occurs within the first basin. Such facts were considered when selecting particle tracks to run.

Water table divides were based on long-term average conditions and were not considered in a probabilistic manner within the CA. Groundwater divide locations relative to facilities/waste sites was considered as part of the sensitivity analysis (Section 9.4.2).

Table 9-2. Potential Existing Models for Use in Establishing Aquifer Flow Path Input Data

Area	Model	Model Code[1]	Reference
A/M	2008 A/M Groundwater Flow Model	GMS v6.0 (MODFLOW)	GeoTrans 2009
D	2005 D-Area Expanded Operable Unit (DEXOU) Model	GMS v.5.1 (MODFLOW)	Brewer and Sochor 2002
GSA (E, F, H, S, and Z)	1999 or 2004 GSA Models	FACT (1999) or PORFLOW (2004)	Flach and Harris 1999 or Flach 2004
C, D, K, L, N, P, and TNX	1999 C, K, L, P Model	FACT	Flach et al. 1999
C	2000 C-Area Groundwater (CAG) Model	GMS v.3.0 (MODFLOW)	Bills et al. 2000
L	2004 L-Area Southern Groundwater (LASG) Model	GMS v.5.0 (MODFLOW)	WSRC 2004
R [2]	R-Area	FACT	Jones et al. 1998
R [3]	2002 R-Reactor Seepage Basin (RRSB) Model	GMS v.3.1 (MODFLOW)	WSRC 2003a
TNX	TNX Model	GMS v.3.0 (MODFLOW)	WSRC 2000

[1] GMS = Groundwater Modeling System
[2] The model domain covers both the RRSBs and the reactor building.
[3] The model domain covers only the RRSBs (i.e., reactor building is not included).

9.1.4.2 Stream Flow Input Data Requirements

Annual average stream and Savannah River flow data associated with the POAs was utilized for the CA Base Case. The average stream/river flows were based on the United States Geological Survey (USGS) period of record, excluding years that included reactor cooling water discharges. The following distributions were developed based upon data availability:

- Distribution of the 1,000-year average flow (assuming no significant climate change) (this distribution will be utilized within the uncertainty analysis)
- Distribution of the annual average flows
- Distribution of the daily average flows

Additionally the low flow 7Q10 (i.e., seven-day, consecutive low flows with a ten year return frequency) values were provided.

9.1.5 Other Considerations

The CA conceptual model development also considered:

- Low-Level Waste Disposal Facility Federal Review Group (LFRG) criteria outlined in Phifer et al. 2008a, and listed in Appendix B of this CA (particularly, Criteria 3.3.2.3, 3.3.2.4, 3.3.5.1, 3.3.6.1, 3.3.6.3, 3.3.6.4, 3.3.6.7, and 3.3.7.2)
- Applicable comments on the previous CA (WSRC 1997a and DOE 1998), outlined in Table 4 of Phifer et al. 2008a, and the corresponding "Response in Relation to Current CA" in that same table, for Program Plan Tasks 4.2.18, 4.2.20, 4.2.21, and 4.2.24
- Software quality assurance requirements outlined in CA Section 6.1, and detailed in Swingle (2006)
- Design check requirements outlined within CA Section 6.1
- The current End State Vision assumption of a 2025 end state, such that CA GoldSim™ (GTG 2007) transport modeling was initiated using the associated representative inventory data, and dose impacts were determined beginning in that year
- The possibility of double counting the contribution of tritium in cases where groundwater plumes and associated source facilities/waste sites may both account for the same tritium inventories. Steps were taken to assure that double counting did not occur, recognizing that, due to the mobility of tritium, a significant fraction of the tritium discharged to source facilities/waste sites has already migrated off the SRS site through groundwater plumes and surface water.

9.2 MODELING METHODOLOGY

The CA modeling methodology follows the approach outlined in Section 9.1 and the CA Conceptual Model report by Phifer et al. (2008c). The following discussion is derived largely from the *Savannah River Site Composite Analysis: Base Case Deterministic Calculations* report (Smith et al. 2009a).

Radionuclide fluxes are followed from the source through the vadose zone and aquifer to the stream mouth and Savannah River where doses to a member of the public are calculated. Because concentrations for CA dose modeling are needed only within surface water sources at the SRS boundary (i.e., stream mouths or Savannah River) for the base case, the CA base case model (Section 9.3) is required to consider only the flux of radionuclides from the source through the vadose and aquifer zones to the streams/river, where concentrations will be determined from the radionuclide flux and the flow rate of the stream/river.

Figure 9-2 shows a flow diagram for the general CA calculation methodology as it was implemented. From CA Section 8.0 and the CA inventory report, *Inventory of Residual Radioactive Material at the Projected Savannah River Site End State* (Hiergesell et al. 2008), 126 individual sources of residual radioactive material at SRS were identified. This total includes some cases where inventories given in the report were combined. For example, inventories were listed for individual waste tanks but, for purposes of the CA, tanks were grouped by type because tanks of the same type are located in the same general area. Of these 126 sources, 15 were subsurface sources consisting of groundwater plumes (Groundwater Operable Units, [GOUs]) and radionuclides in stream-bed sediments (Integrator Operable Units [IOUs]). Transport paths from the other 111 surface sources to seep lines into a stream or the Savannah River were determined (Hamm et al. 2009). This analysis found that releases from some of the 111 sources split to two different streams and that some transport paths showed a bimodal distribution with long and short path lengths. On the basis of this analysis, 18 of the surface sources were divided into two or three parts to better represent the transport paths. Subsequent analysis determined that the inventory from three of the surface sources [Old Radioactive Waste Burial Ground (ORWBG), Mixed Waste Management Facility (MWMF), and Naval Reactor Component Disposal Area (NRCDA) 643-7E] needed to be subdivided to model the release mechanisms creating an additional eight sources.

As a result, 137 surface sources were analyzed using what was termed a "generic" release scenario where the inventory was released directly onto the ground surface with natural background infiltration (i.e., no credit for a cap or cover) and the resulting dose was calculated assuming recreational use of stream water and residential use of river water contaminated by the source. If the total dose for a surface source from this analysis was less than 0.1 mrem/yr for all time from the release date to 10,000 years beyond the assumed site end-state date of 2025, the "generic" dose was accepted as a conservative estimate of the actual dose and no further analysis was performed for this surface source. If the maximum dose calculated by the "generic" release scenario exceeded 0.1 mrem/yr (0.1% of the 100 mrem/yr dose limit), additional analysis was performed to account for the presence of engineered barriers that would inhibit radionuclide transport or to include a more realistic treatment of the mechanism for radionuclide release. This additional analysis was termed "source release" modeling. Of the 137 surface sources analyzed using the "generic" release scenario, the maximum dose from 24 sources exceeded the 0.1 mrem/yr criterion. Six other sources were included in the subsequent more detailed source release analysis for consistency. For example, the R-Area reactor vessel was included in the source release modeling even though the total maximum dose from this source was less than 0.1 mrem/yr, because all of the other reactor vessels exceeded the limit. One source, the P-Reactor Surface Contamination source, was removed from the more detailed analysis because no single source release mechanism was immediately identifiable and it was just slightly above the cutoff criterion of 0.1 mrem/yr (see discussion in Section 9.3.7). The net result was that 29 surface sources were analyzed using more detailed "source release" modeling along with the 15 plume and stream-bed sources that also required separate analysis. Results from these 44 source release modeling cases were combined with the generic results from the other 108 cases to create the overall CA base case results for 152 individual sources. All of the base case calculations (both generic and source release) were performed using a GoldSim™ model of radionuclide transport and dose calculation. This model is described in Section 9.3.

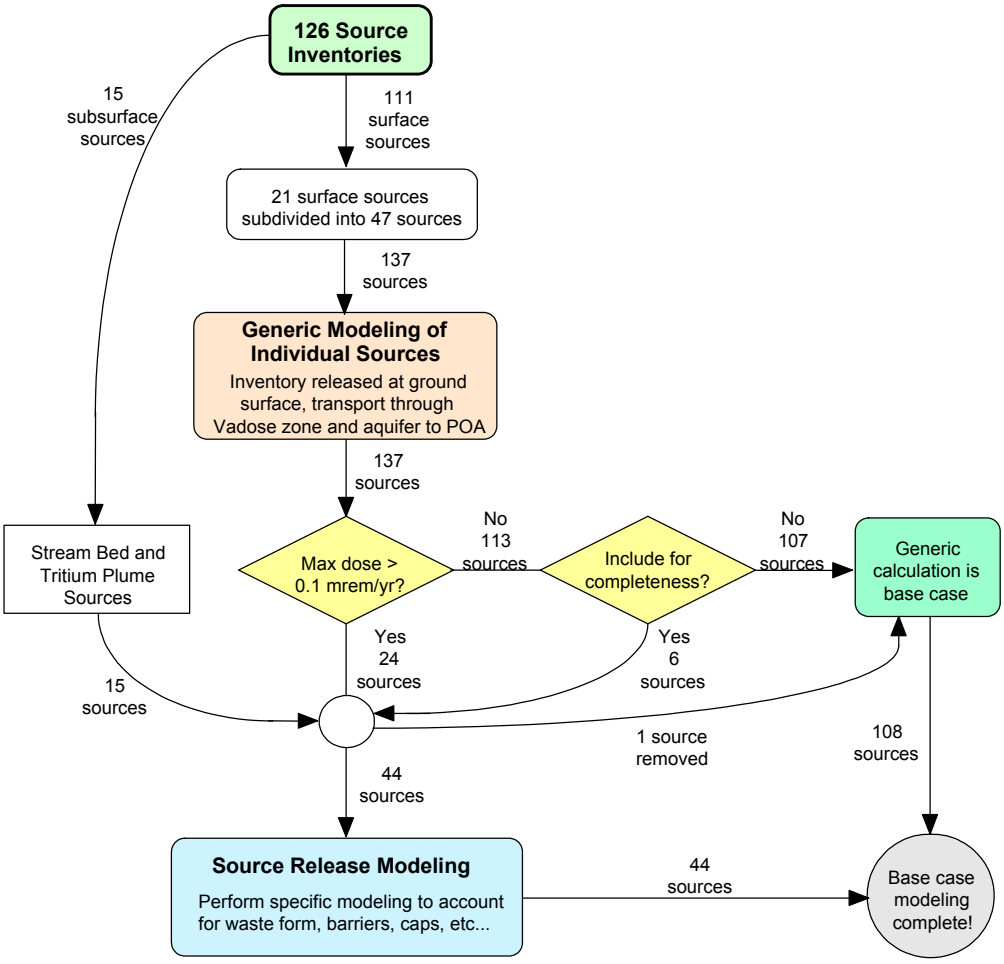


Figure 9-2. Schematic representation of CA Calculation Methodology (from Smith et al. 2009a).

The methodology for dose calculations was developed separately from the transport simulation modeling. The module used to perform dose calculations was developed under a subcontract to Neptune, Inc (Phifer et al. 2009). The dose module calculates a recreational dose from exposure by fishing (ingestion of fish), swimming, boating and shore shine based on radionuclide concentrations in the stream. A residential dose is calculated from radionuclide concentrations in the Savannah River assuming that this water is used for drinking and farming. The residential dose is based on ingestion of river water and vegetables, beef and milk raised using river water, ingestion of soil, inhalation of dust, and soil shine from absorption of irrigation water. Assuming that the resident will also recreate, the total dose from each source is the sum of the recreational and residential doses. Additional information concerning the dose module can be found in Section 9.3.2 and Phifer et al. (2009).

9.2.1 Modeling Codes and Existing Models

Radionuclide concentrations for dose modeling were needed only within surface water sources at the SRS boundary (i.e., stream mouths or Savannah River) for the base case; therefore, the CA base case model considered only the flux of radionuclides from the source through the vadose and aquifer zones to the streams/river where concentrations were determined from radionuclide fluxes and surface water flow rates. The GoldSim™ code (GTG 2007) is suitable to model radionuclide flux, both deterministically and probabilistically, from the source through the vadose and aquifer zones to the streams/river. Therefore, this code was selected for modeling the CA deterministic base case, alternate sensitivity cases, and the probabilistic (stochastic) realizations (Smith et al. 2009a, 2009b). The dose module used to perform CA dose calculations also was developed using the GoldSim™ modeling platform.

Existing SRS PORFLOW, FACT, and MODFLOW flow models were used to determine the groundwater flow path length, clay length in the flow path, and flow velocity to surface water outcrop for each facility/waste site modeled and the stream to which flow from the facility outcrops as outlined in Section 9.1.4.1. Brief descriptions of these codes follow.

PORFLOW™ is a commercially available computer code developed and marketed by Analytic & Computational Research, Inc. (ACRI 2004). It is a comprehensive mathematical model for simulation of multi-phase fluid flow, heat transfer and mass transport in variably saturated porous and fractured media. PORFLOW can simulate transient or steady-state problems in Cartesian or cylindrical geometry. The porous medium may be anisotropic and heterogeneous and may contain discrete fractures or boreholes within the porous matrix. It can also simulate chemical reactions or radioactive decay. It utilizes control volume discretization and the nodal point integration method, with all properties and state variables being defined at the center of an interior grid cell. The theoretical models within the code provide a unified treatment of concepts relevant to fluid flow and transport. In particular PORFLOW can be used to simulate groundwater flow and contaminant transport in the vadose zone and underlying aquifers. (Aleman 2007, Flach 2004, and Hang 2007)

FACT (Subsurface Flow And Contaminant Transport) is a variably-saturated 3D finite-element groundwater flow and solute contaminant transport code developed by the Savannah River National Laboratory (SRNL) (Hamm et al. 1997, Hamm and Aleman 2000). FACT version 1.0 is an outgrowth of the SAFT3D code developed jointly by HydroGeoLogic, Inc. and SRNL (Huyakorn et al. 1991). Distinguishing features of FACT include efficient memory management and numerical algorithms that make large grids feasible and user-friendly boundary conditions. For example, the combination recharge/drain boundary condition automatically determines whether a surface node should receive recharge or be discharging groundwater, based on the head solution. The software has been extensively verification and validation (V&V) tested (Hamm et al. 1997). (Flach and Harris 1999)

MODFLOW is a computer program developed and made available by the U.S. Geological Survey (Harbaugh et al. 2000). It numerically solves the three-dimensional ground-water flow equation for a porous medium by using a finite-difference method to approximate the groundwater flow equation as a set of algebraic equations in a discretized three-dimensional grid of rectangular cells in which the medium properties are assumed to be uniform. It simulates steady and nonsteady flow in an irregularly shaped flow system in which aquifer layers can be confined, unconfined, or a combination of confined and unconfined. It has a modular structure that allows it to be easily modified to adapt the code for a particular application. MODFLOW is included in the Groundwater Modeling System (GMS) (Brigham Young University 1999) software package (version 6.5; Feb. 2, 2009 build date). GMS is a standard suite of tools for modeling analyses (GeoTrans 2009).

No overall SRS flow model exists that can meet all of the CA aquifer flow data requirements; however, numerous flow models covering various portions of the SRS have been created over the years (see Table 9-2 and Hamm et al. 2009). The most reliable flow models have been 3D models that cover a modest range of the site, typically addressing several process areas simultaneously. These models typically cover a broad enough range to take advantage of surrounding seepage faces and surface water head boundary conditions. Generally, such surrounding features provide additional assurance in the reliability of a flow model and for CA purposes directly address the discharge locations of interest (i.e., the POAs). Of the available flow models, those considered to be the most reliable (and up to date) were considered for use in developing the required aquifer flow data for the CA (Hamm et al. 2009).

Four existing flow models were chosen to cover the facility/waste sites for which inventories were identified in Section 8.0, and are listed in Table 9-3. Based on the source locations and the active range of the four flow models, the sources were grouped together based on which flow model they reside within. The active region for each of the four flow models is shown in Figure 9-3, along with the corresponding aerial mesh.

Table 9-3. List of Flow Models Employed in Estimating Aquifer Parameters

Flow Model	Code	Numerical Method	Development Organization	Flow Model Reference	Users Manual & QA/QC Documentation
AM	MODFLOW	Finite difference method (Control volume method)	GeoTrans	GeoTrans 2009	Harbaugh et al. 2000; Pollock 1994; WSRC 2006 ; Zheng et al. 2001; Zheng and Wang 1999
CKLP	FACT	Finite element method	SRNL	Flach et al. 1999	Aleman 1999; Hamm and Aleman 2000
GSA	PORFLOW (FACT based)	Control volume method	SRNL	Flach 2004	Aleman 2007; ACRI 2004; Hang 2007
R	FACT	Finite element method	SRNL	Jones et al. 1998	Aleman 1999; Hamm and Aleman 2000

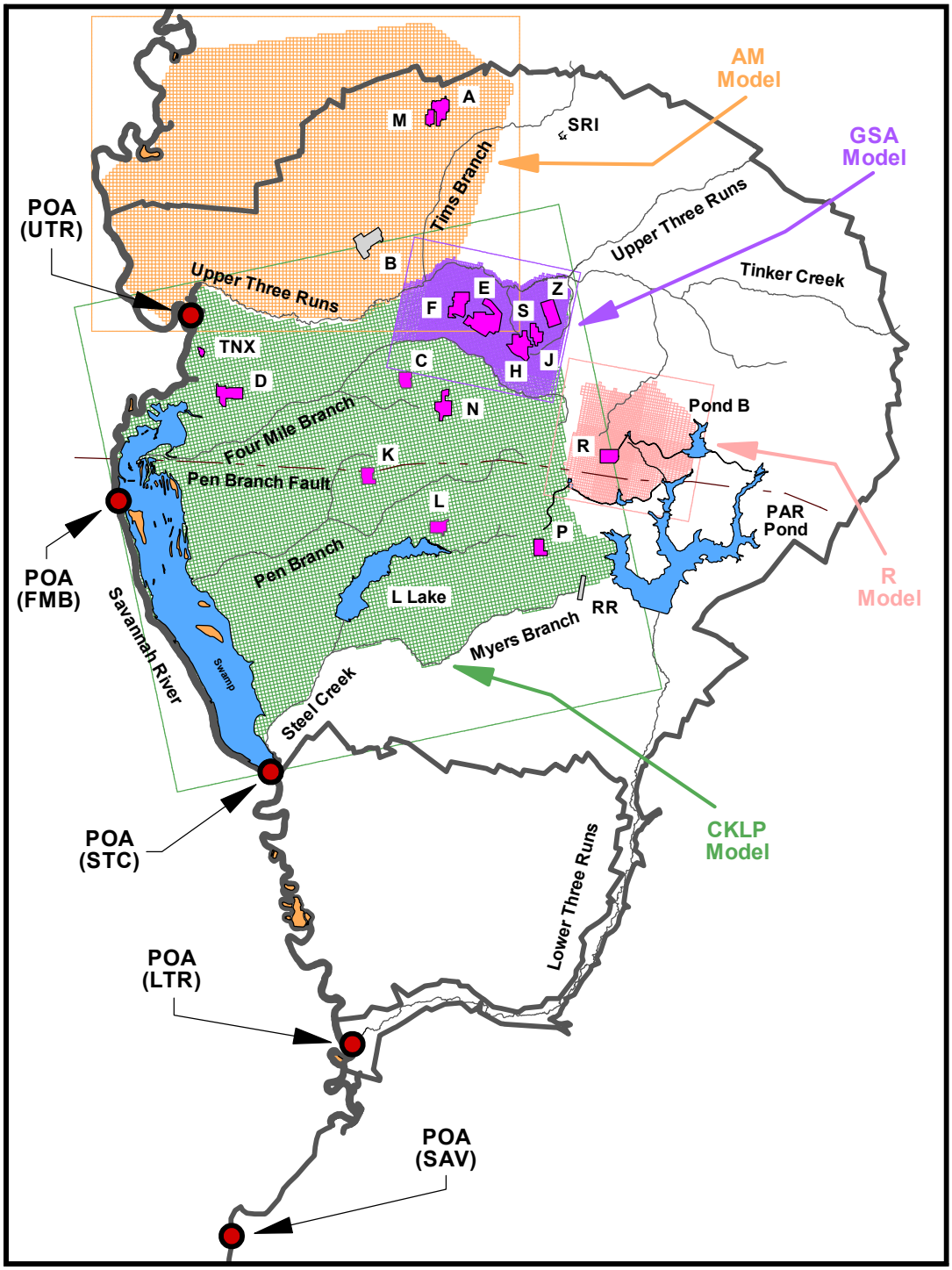
The flow field described by the MODFLOW-based 3D flow model of the AM area, which was developed by GeoTrans (2009), was utilized in the CA in the particle tracking analyses. For the CKLP area, flow-field characteristics described using a FACT-based 3D flow model, developed by Flach et al. (1999) were used. A comprehensive hydrogeological framework for the GSA was developed by Smits et al. (1997). Within this framework a FACT-based 3D flow model of the GSA area was developed by Flach and Harris (1999). This FACT-based model was later converted over to a PORFLOW-based model by Flach (2004), and the flow field arising out of this PORFLOW (Version 5.97)-based version was utilized in the CA. Finally, the flow field described by a FACT-based 3D flow model of the R area, which was developed by Jones et al. (1998), was used in the CA particle tracking analyses. References for the user's manuals and Quality Assurance (QA)/Quality Control (QC) documentation for each of the models is also provided in Table 9-3.

The actual node sizes for each flow model are provided in Table 9-4. As can be seen in Figure 9-3, there is some overlap between the CKLP and GSA Flow Model domains where CA source units of interest exist. Since it is believed that the finer meshed GSA Flow Model is more accurate and appropriate for GSA purposes, CA source units residing within the GSA are addressed using the GSA Flow Model. Table C-1 indicates which flow model each baseline CA source unit (and any refinement thereof) resides in and also indicates which POA(s) it contributes to.

Table 9-4. Number of Nodes per Direction for Each Flow Model Employed in Estimating Aquifer Parameters

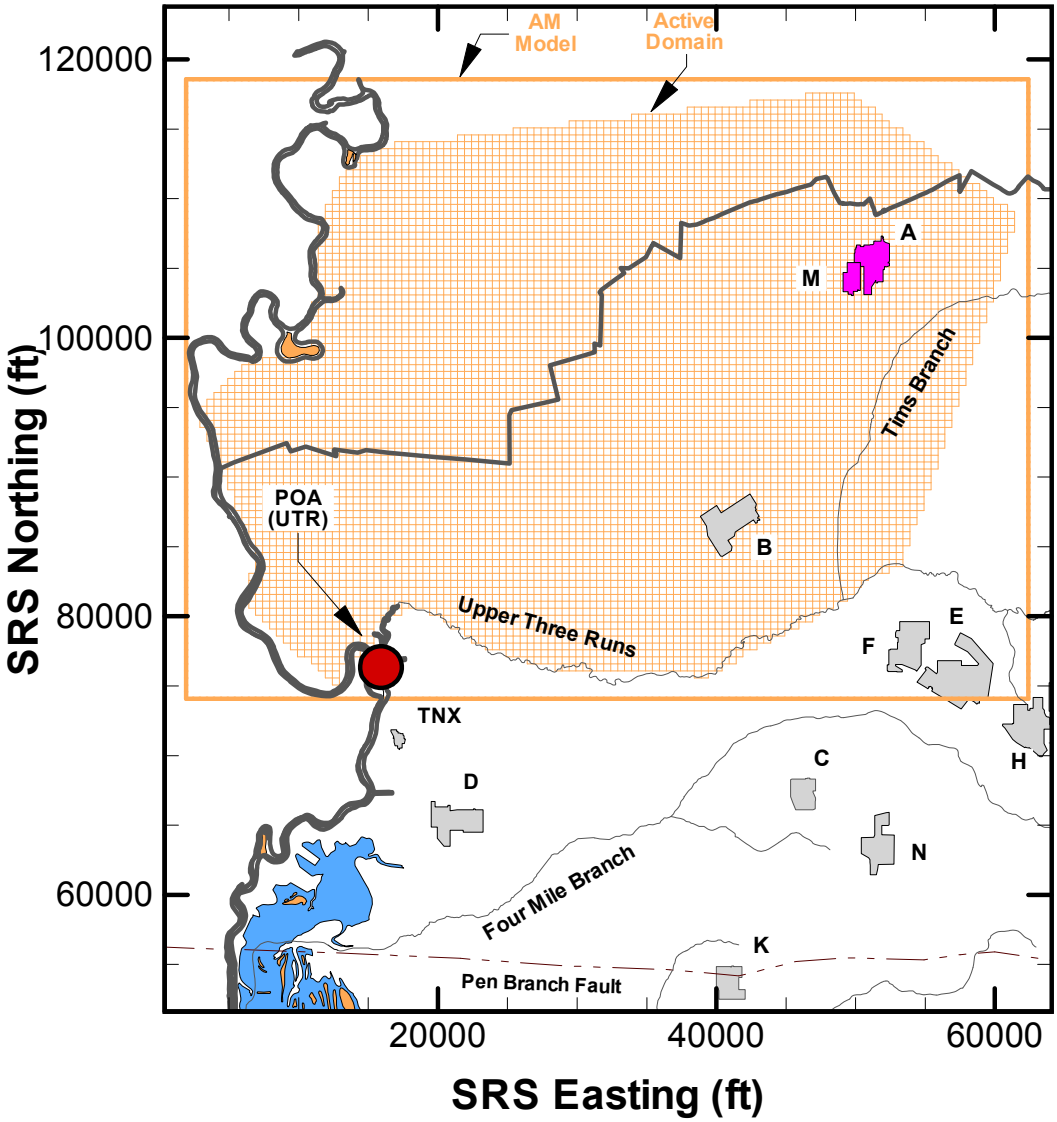
Flow Model	Number of Nodes in x Direction	Number of Nodes in y Direction	Number of Nodes in z Direction
AM	121	89	18
CKLP	141	141	9
GSA	108	77	21
R	66	63	33

To better illustrate the chosen process areas for each flow model, the process areas being addressed by each flow model are highlighted (shaded in magenta) in close-up views of the Site map in Figure 9-4 through Figure 9-7 for the flow models AM, CKLP, GSA, and R, respectively. In each figure, the outer boundary and "active" internal mesh of the specific flow model are drawn. Those process areas in which CA source units are addressed by the specific flow model for that area are shaded in magenta, while those not considered are shaded in grey.



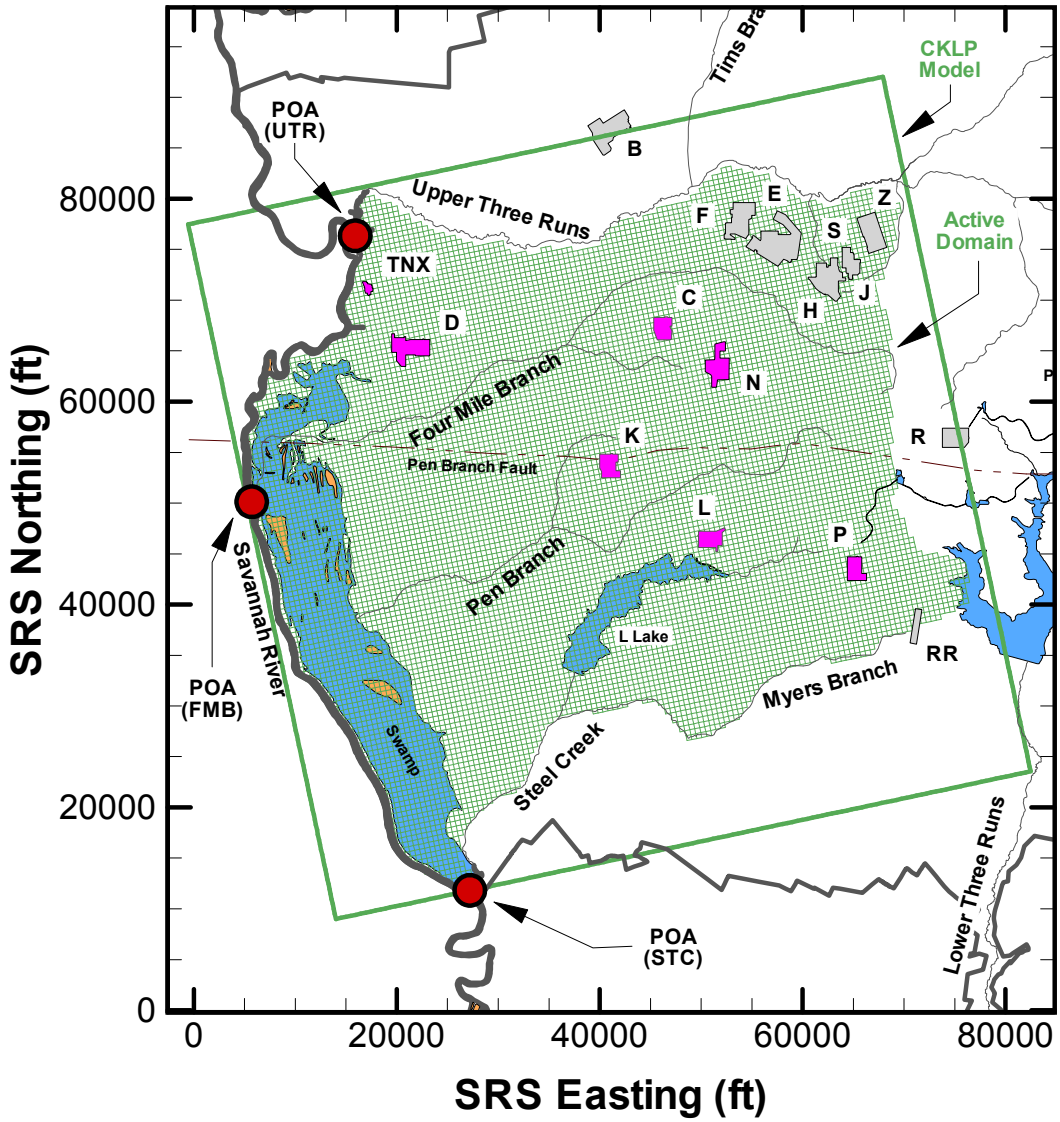
This map shows the locations of the various process areas (shaded in magenta) that contain CA Units of interest, the footprints of the four 3D flow models employed, and the five Points of Assessment.

Figure 9-3. SRS Map of 3D Model Domains



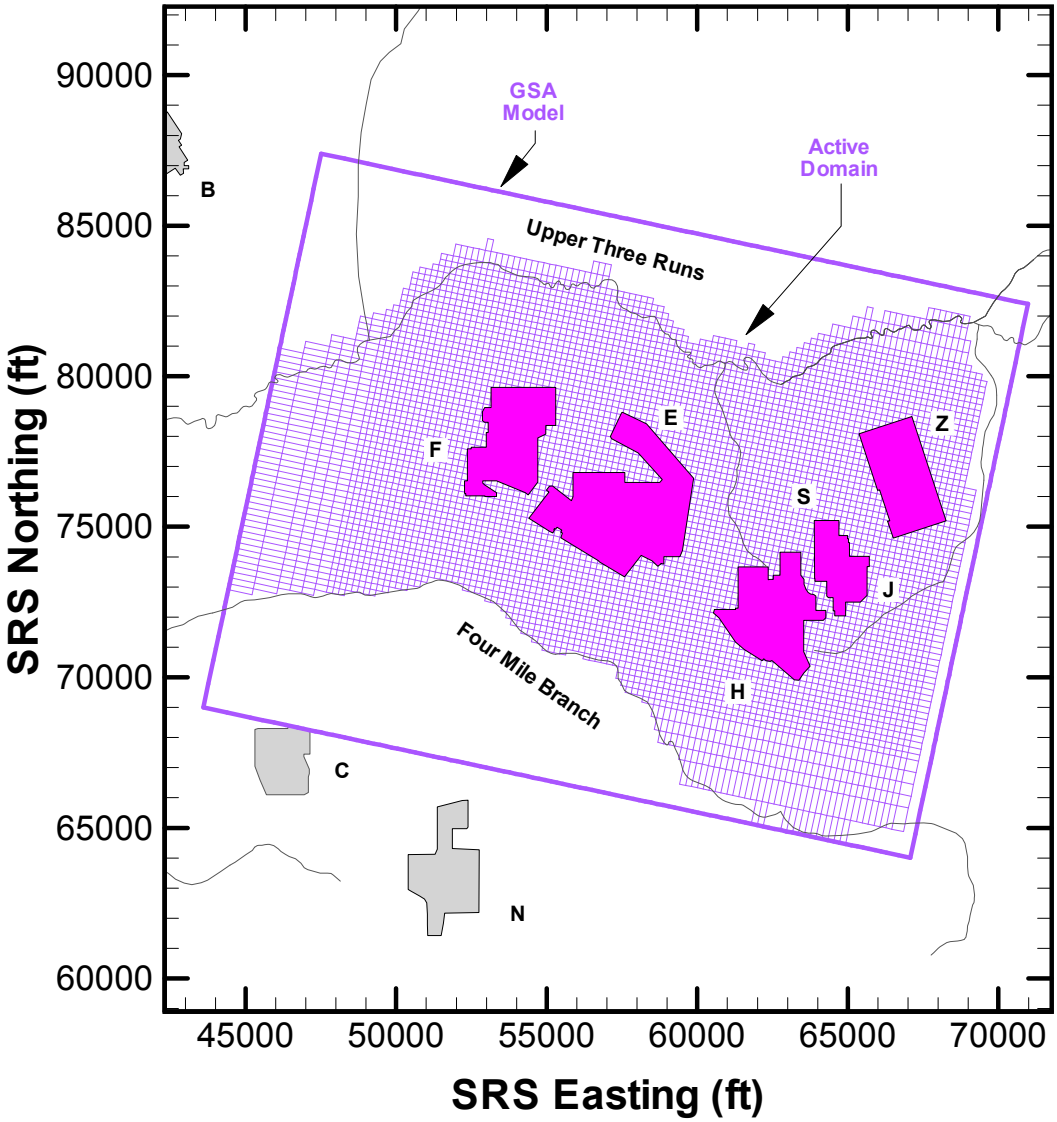
This map shows the active grid region for the AM Flow Model along with those process areas assigned to it.

Figure 9-4. Close-Up of SRS Map for the AM Flow Model



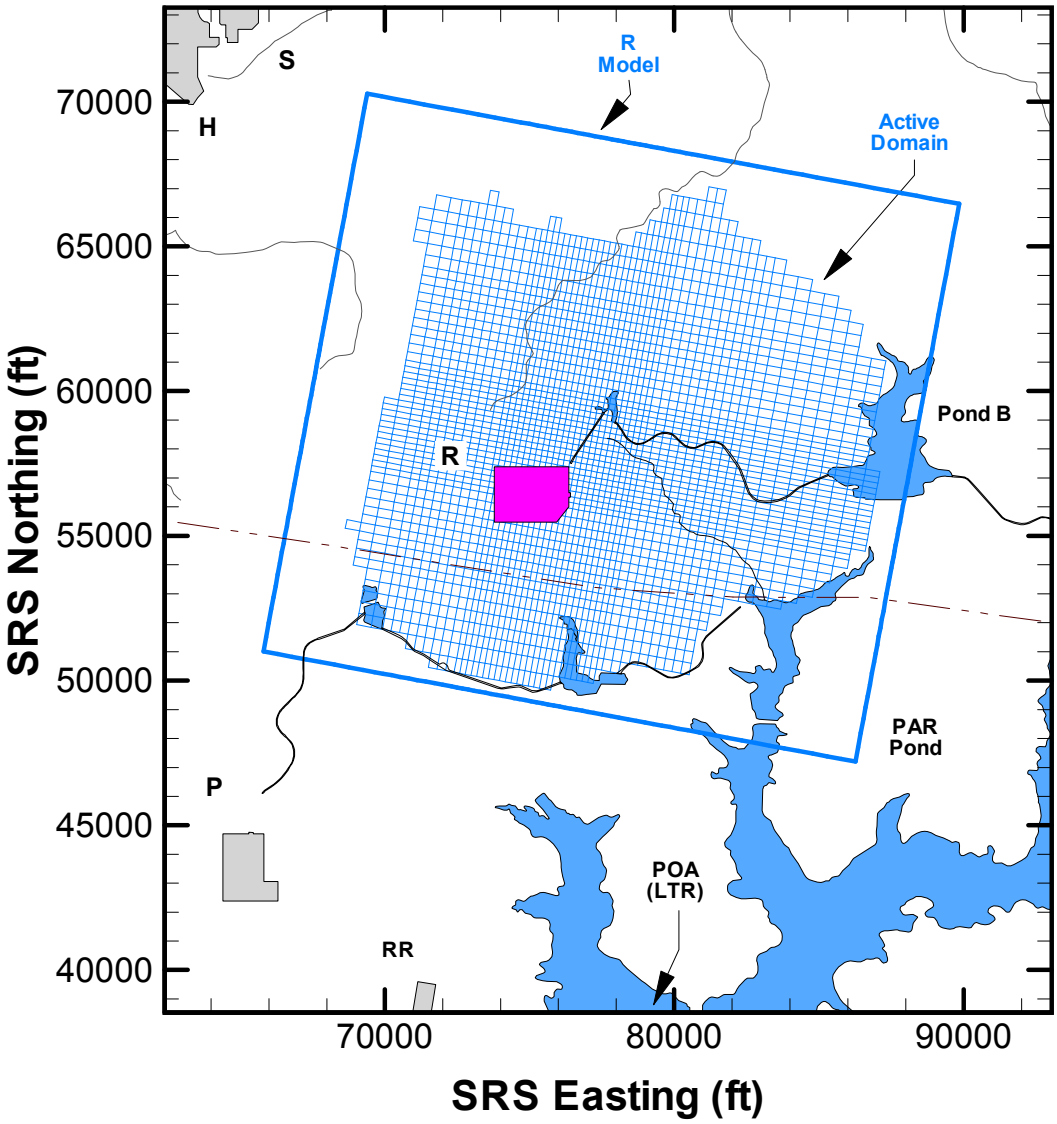
This map shows the active grid region for the CKLP Flow Model along with those process areas assigned to it.

Figure 9-5. Close-Up of SRS Map for the CKLP Flow Model



This map shows the active grid region for the GSA Flow Model along with those process areas assigned to it.

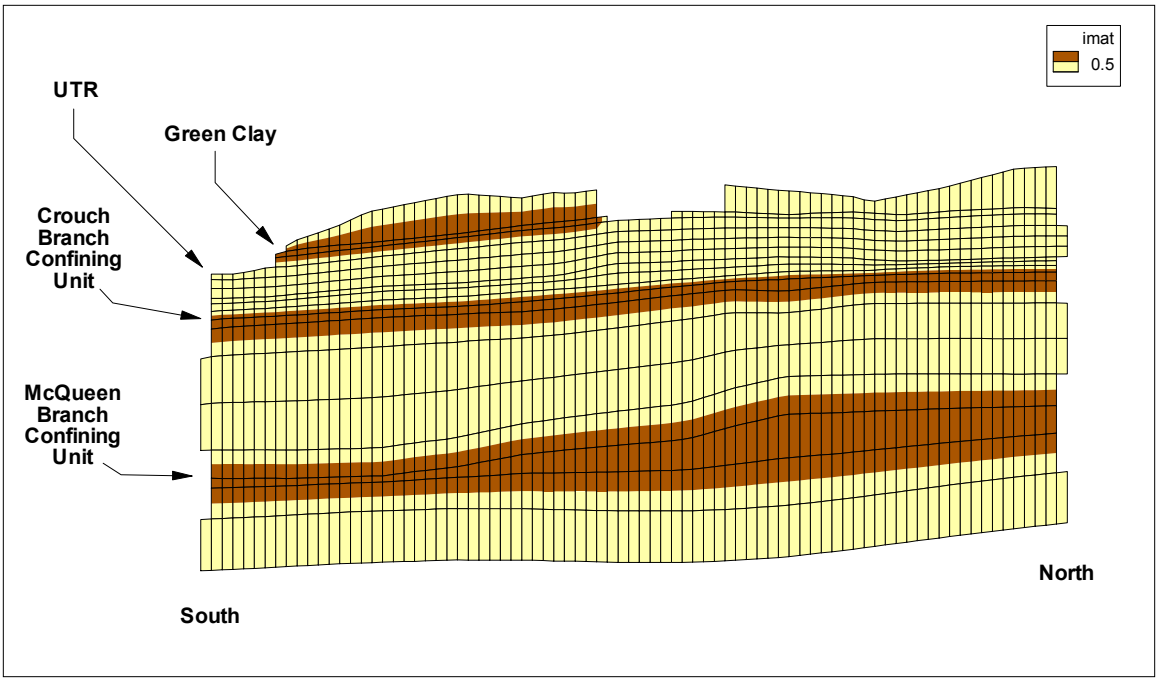
Figure 9-6. Close-Up of SRS Map for the GSA Flow Model



This map shows the active grid region for the R Flow Model along with those process areas assigned to it.

Figure 9-7. Close-Up of SRS Map for the R Flow Model

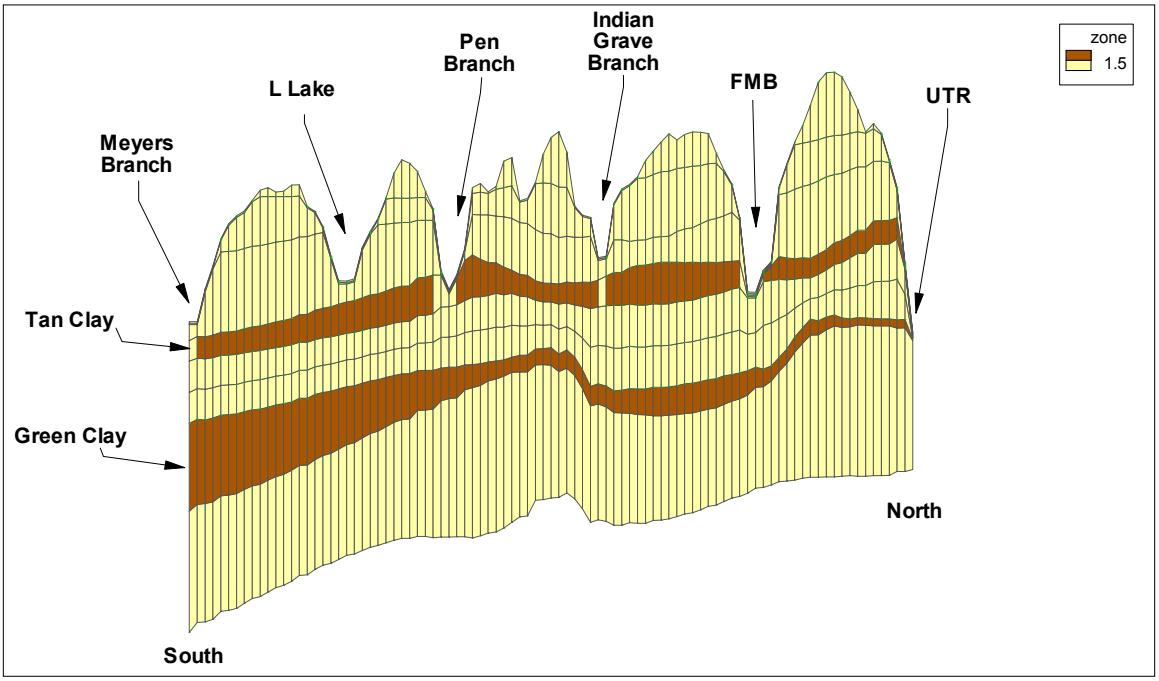
The AM Flow Model (based on the MODFLOW code) employs a finite difference technique (control volume solution technique). The 18 vertical layers within this AM Flow Model represent a vertical stack of 18 MODFLOW cells. A vertical cross-section through the mesh is provided in Figure 9-8 below. This cross-section has been vertically distorted (i.e., not to scale) to better illustrate the confining units and clay regions of interest. The vertical cross-section shown corresponds to the ith plane with index of 77. Note that for the AM Flow Model, the mesh being shown represents the vertices of the cell centers, not the vertices of the cell edges.



A typical cross-sectional slice through the control-volume mesh of the AM Flow Model highlighting its clay regions and vertical meshing (i^{th} plane =77)

Figure 9-8. A Typical Cross-Sectional Slice through the Control-Volume Mesh of the AM Flow Model

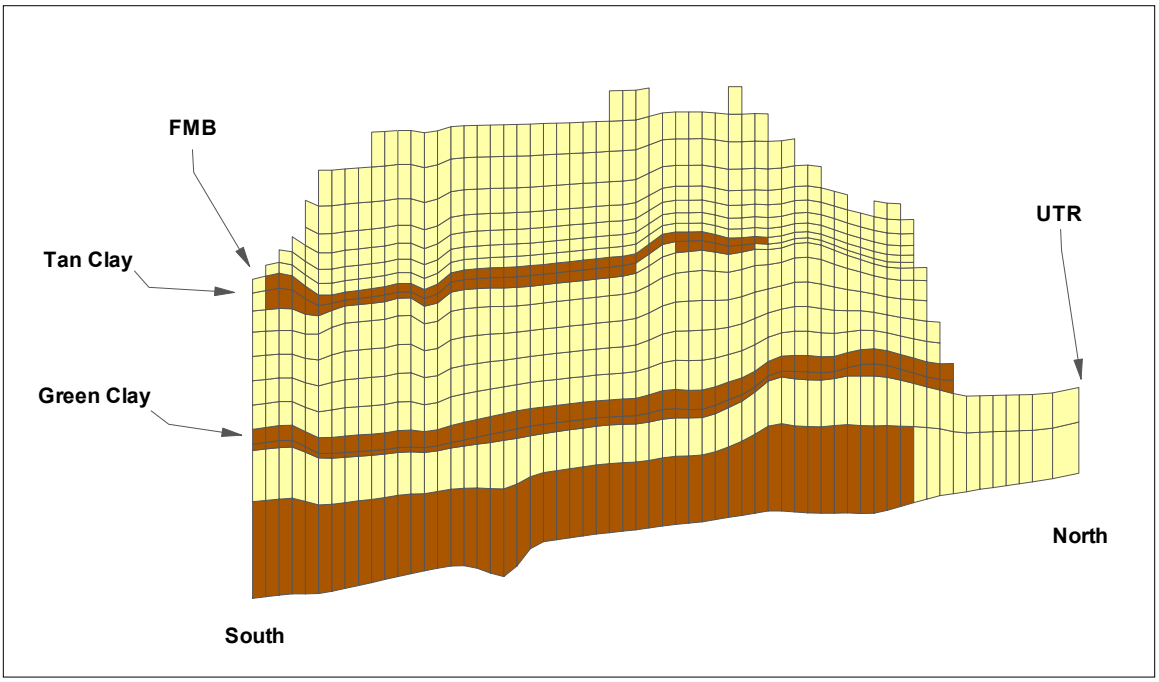
The CKLP Flow Model (based on the FACT code) employs a finite element solution technique. The eight vertical layers within this CKLP Flow Model represent a vertical stack of eight FACT finite elements. A vertical cross-section through the mesh is provided in Figure 9-9 below. This cross-section has been vertically distorted (i.e., not to scale) to better illustrate the confining units and clay regions. The vertical cross-section shown corresponds to the ith plane with index of 87.



A typical cross-sectional slice through the finite-element mesh of the CKLP Flow Model highlighting its clay regions and vertical meshing (i^{th} plane =87)

Figure 9-9. A Typical Cross-Sectional Slice through the Finite-Element Mesh of the CKLP Flow Model

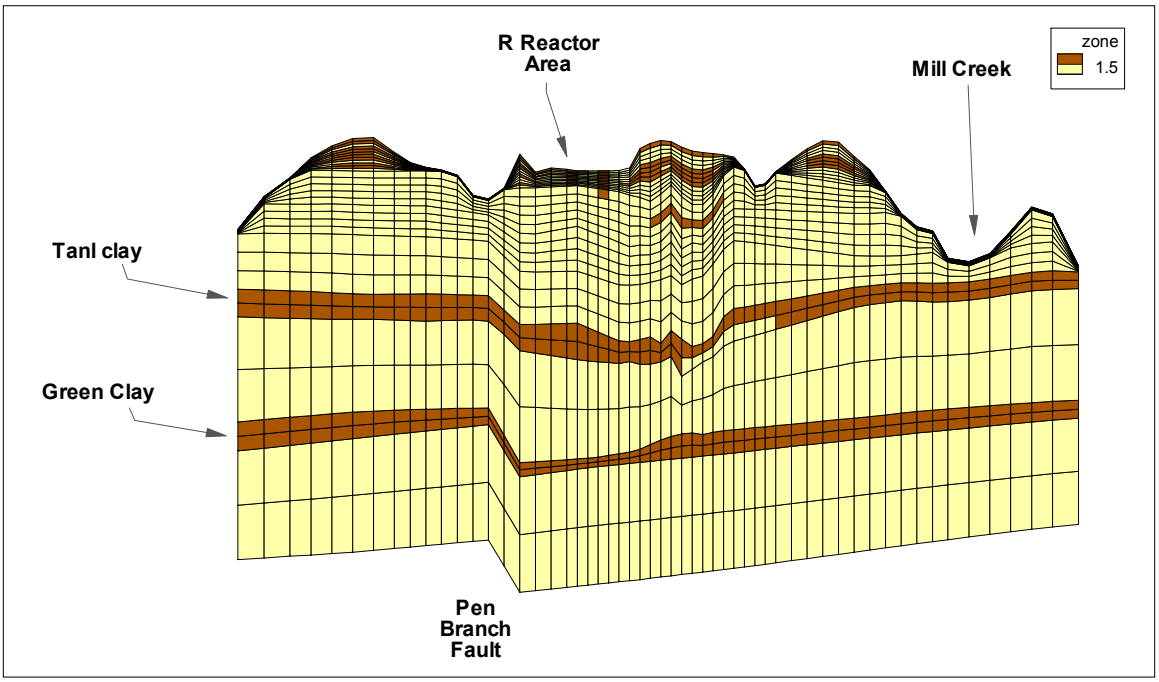
The GSA Flow Model (based on the PORFLOW code) utilizes a finite volume solution technique. The 21 vertical layers within this GSA Flow Model represent a vertical stack of 21 PORFLOW cells. A vertical cross-section through the mesh is provided in Figure 9-10 below. This cross-section has been vertically distorted (i.e., not to scale) to better illustrate the confining units and clay regions. The vertical cross-section shown corresponds to the ith plane with index of 50.



A typical cross-sectional slice through the finite-volume mesh of the GSA Flow Model highlighting its clay regions and vertical meshing (i^{th} plane =50)

Figure 9-10. A Typical Cross-Sectional Slice through the Finite-Volume Mesh of the GSA Flow Model

The R Flow Model (based on the FACT code) employs a finite element solution technique, as noted earlier. The 32 vertical layers within this R Flow Model represent a vertical stack of 32 FACT finite elements. A vertical cross-section through the mesh is provided in Figure 9-11 below. This cross-section has been vertically distorted (i.e., not to scale) to better illustrate the confining units and clay regions. The vertical cross-section shown corresponds to the ith plane with index of 25.



A typical cross-sectional slice through the finite-element mesh of the R Flow Model highlighting its clay regions and vertical meshing (ith plane =25)

Figure 9-11. A Typical Cross-Sectional Slice through the Finite-Element Mesh of the R Flow Model

9.2.2 GoldSim™ Composite Analysis Model Overview

GoldSim™ (GTG 2007) was used to create a one-dimensional model of radionuclide transport from a source location through structures (representing the waste material and engineered barriers), the vadose zone, and the saturated aquifer into designated SRS streams and the Savannah River (see Section 9.3.1). GoldSim™ was also used in implementing recreational and residential dose calculations, which are described in more detail in Section 9.3.2.

9.2.2.1 Transport Application

Version 9.60 of the commercial GoldSim™ software (GTG 2007), which provides capabilities for simulating the release and transport of radioactive materials through the environment, was used as the software for CA modeling. GoldSim™ is a graphically oriented modeling environment where the user builds models by dragging and dropping functions into the working area, entering data and/or equations into the functions, and connecting the functions to perform the desired calculation. The model was organized by collecting related functions into GoldSim™ containers. GoldSim™ provides plotting features, built-in connections to Excel spreadsheets, and capabilities for performing Monte Carlo probabilistic simulations. GoldSim™ performs unit conversions automatically, which is a significant help in maintaining correct calculations. GoldSim™ also provides a model development environment that makes it easy to include documentation of calculations within the model. All of the base case deterministic calculations described in Section 9.3 and the sensitivity calculations described in Section 9.4 were performed using the same GoldSim™ CA model (Model Version 1.05).

Figure 9-12 shows a view of the top level of the GoldSim™ CA model. Nine containers were created at this level to collect related modeling functions. Table 9-5 gives a brief summary of the functionality in each of these containers. Only the first six containers listed in Table 9-5 contain significant computational modeling functions, which are described in more detail in Section 9.3.

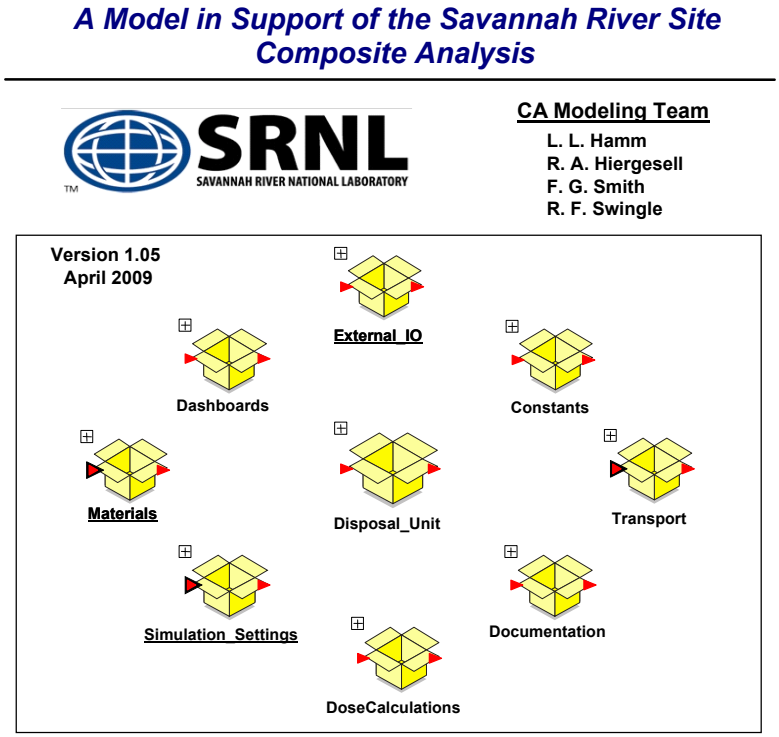
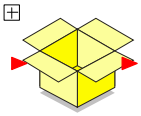
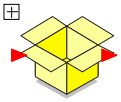
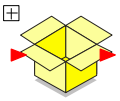
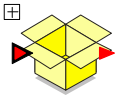
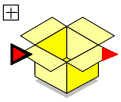
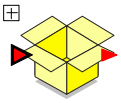
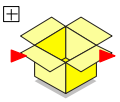
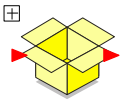
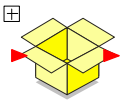


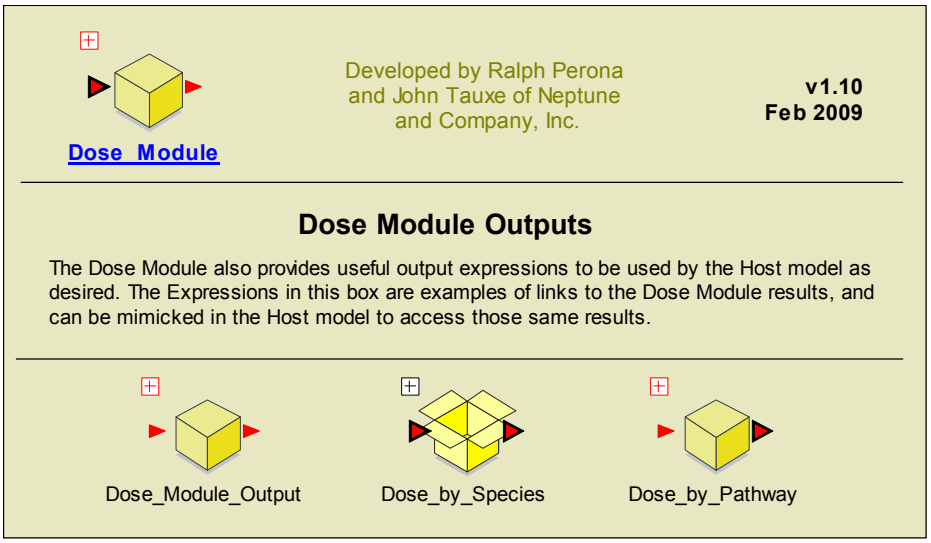
Figure 9-12. Top-Level View of GoldSim™ CA Model (from Smith et al. 2009a).

Table 9-5. Functionality in Top Level CA Model GoldSim™ containers (from Smith et al. 2009a).

 Disposal_Unit	Contains the models describing release of source inventory to the environment and transport through waste, barriers, vadose zone and aquifer to the stream and river where dose calculations are performed. Described further in Section 9.3.1 and its subsections.
 DoseCalculations	Contains the module developed by subcontractor Neptune, Inc. that performs dose calculations with some additional output functionality included by SRNL. The dose module was checked and documented by Phifer et al. (2009) and is discussed in Section 9.3.2.
 External_IO	Contains links to Excel spreadsheets used to read input data and fixed parameters into the model and to collect output from the model calculations. Contents are described further in Section 9.3.3.
 Materials	Contains definitions of materials used to create cells representing porous media through which the contaminant transport takes place. Contents are described further in Section 9.3.4.9.
 Simulation_Settings	Contains fixed settings for the simulation starting year (1950), end state year (2025) and period of assessment (1,000 yrs). Calculates the simulation time in calendar years and year of cap failure and sets the inventory release time. Contains functions that determine the data collection frequency used in container External_IO.
 Transport	Contains infiltration rates and stream and river flow distributions. Contents are described further in Section 9.3.1.
 Constants	Defines a few constants such as vectors of ones and zeros dimensioned by species and elements used throughout the model. Does not contain any computational models and will not be described further.
 Dashboards	Contains links to two dashboards that can be used to run the simulation. Does not contain any computational models and will not be described further.
 Documentation	Contains files that briefly document the model. This report is more comprehensive than the internal model documentation. This container does not contain any computational models and will not be described further.

9.2.2.2 Dose Application

Figure 9-13 shows a top-level view of the contents of GoldSim™ container DoseCalculations. Containers Dose_by_Species and Dose_by_Pathway were added to the model to collect the particular data of interest to the CA and have no impact on the actual dose calculations. The three changes described in the change log text in the lower part of Figure 9-13 were made to the Dose_Module as-received from Neptune, Inc. The most significant change was to decrease the bioaccumulation factor for C14 in fish from 50,000 L/kg to 3 L/kg. The value of 3 L/kg is based on the study by Hinton et al. (2009) that models carbon in the environment using data specific to the Savannah River Site. Because this is such a large change from C14 bioaccumulation factors used previously, the impact was investigated in model sensitivity studies. Additional information concerning the dose module can be found in Section 9.3.2.



Change Log for Container Dose_Module

In data element \Fish_Concentrations\Fish_Water_ratios:

The bioaccumulation factor for Carbon was changed from 50,000 L/kg to 3 L/kg based on SRNS-TR-2009-00049 "Systems Model of Carbon Dynamics in Four Mile Branch on the Savannah River Site", T. Hinton, D. Kaplan, D. Fletcher, J. McArthur, C. Romanek, February 12, 2009.

The bioaccumulation factor for Ca was changed from 400 L/kg to 40 L/kg based on a review of the data that revealed an error.

The function \Water_Dose\Drinking_Pathways_Dose\CheckSum_WaterIngDose was removed because of a divide by zero error.

Changes made 2/25/09 by F. G. Smith SRNL.

Figure 9-13. Top-level contents of container DoseCalculations (from Phifer et al. 2009)

9.3 COMPOSITE ANALYSIS BASE CASE DETERMINISTIC MODEL

This section of the CA describes the model used to perform deterministic base case calculations for the CA and presents results from that analysis, and is largely taken from *Savannah River Site Composite Analysis: Base Case Deterministic Calculations* (Smith et al. 2009a). This modeling effort was conducted to fulfill the objectives of Tasks 4.2.18 Radionuclide Source Release Rates, 4.2.19 Conduct Initial Transport Modeling Simulations, 4.2.21 Conduct Transport Modeling, and 4.2.22 Conduct Dose Calculations/Modeling of the Composite Analysis Program Plan (Phifer and Cook 2007). The objectives of these tasks have been fulfilled by the generic release, source release, and base case deterministic modeling and results, all of which include dose modeling, presented herein.

The Disposal Unit Module and Dose Module are described in detail in Sections 9.3.1 and 9.3.2. Model input and output, and a description of input data, are described in Sections 9.3.3 and 9.3.4, respectively. Results of model validation efforts are provided in Section 9.3.5, and the remainder of this section (i.e., 9.3.6, 9.3.7, 9.3.8, and 9.3.9) is devoted to a discussion of modeling results.

9.3.1 Disposal Unit Module

GoldSim™ was used to create a one-dimensional model of radionuclide transport from a source location through structures (representing the waste material and engineered barriers), the vadose zone, and the saturated aquifer into a designated stream and the Savannah River. Recreational dose calculations were performed using the stream concentrations of radionuclides and residential doses were calculated using radionuclide concentrations in the river as discussed in Section 9.1.3.

Figure 9-14 and Figure 9-15 show a schematic representation of the GoldSim™ model in two configurations and Figure 9-16 shows the structure within container Disposal_Unit where this model is implemented. In general, as shown in Figure 9-15, the contaminant transport path is composed of a waste zone containing five mixing cells, a barrier region below the waste with three cells, a vadose zone having 20 cells, and an aquifer path having 200 mixing cells. The model was applied in two modes: 1) Modeling all CA sources using a generic release of the inventory at the surface followed by transport to the stream and river and 2) More detailed source release modeling that accounts to some extent for engineered barriers and inventory release mechanisms within the waste material followed by transport to the stream and river. When run for generic release scenarios, the model operated as shown in Figure 9-14 where inventory transport began at the top of the vadose zone. For most CA sources this simpler modeling was sufficient. If more detailed modeling was required (i.e., the generic release scenario resulted in a dose > 0.1 mrem/yr; see Section 9.3.1.6.1), the model was run as illustrated in Figure 9-15 where more options for source release modeling were available. Release mechanisms available in the model for introducing the inventory at different points along the transport path are listed in Table 9-6. Input ID 4 for surface release from Table 9-6 is the release mechanism used for the 108 sources that only required the "generic" release scenario discussed in Section 9.3.1.6.1. The other release mechanisms listed in Table 9-6 were utilized for the remaining 44 sources, as appropriate, that required specific source release modeling as discussed in Sections 9.3.1.6.3 through 9.3.1.6.8. Not listed in Table 9-6, contaminant fluxes at the water table obtained from PA calculations can also be used as input to the CA model. The input IDs shown in Figure 9-14 and Figure 9-15 correspond to those listed in Table 9-6.

The contaminant transport model is designed to run a single source at a time. Automation methods were developed to sequentially run a series of CA source simulations without the need for user intervention. The model structure shown in Figure 9-15 was unchanged when modeling different sources. Cell sizes in the model will vary between different sources but the number of cells and general model structure do not. Values for parameters such as source inventory, source release mechanism, infiltration, transport path length, transport velocity, materials, and event timing are all entered into the model through the input.

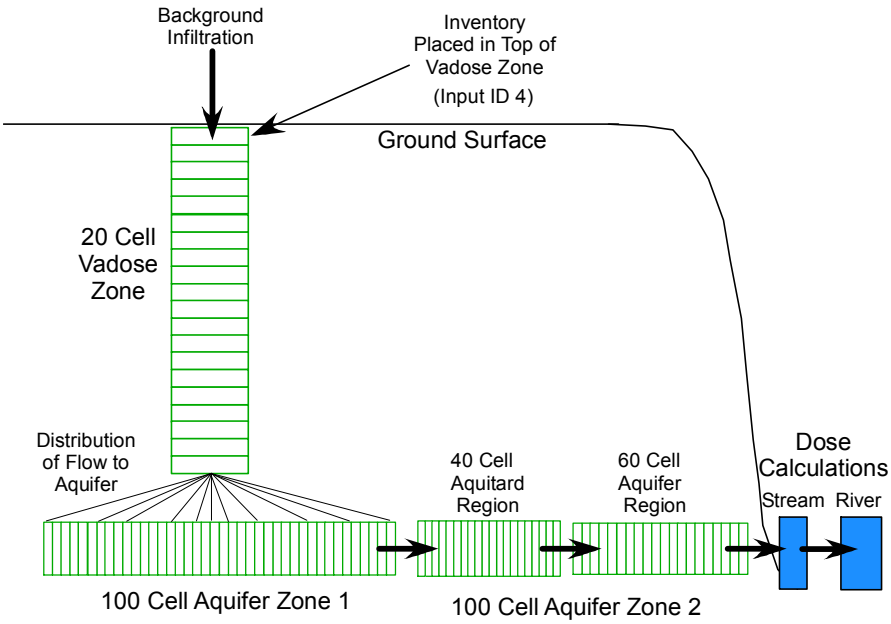


Figure 9-14. Schematic Representation of GoldSim™ Transport Model Used for Generic Releases

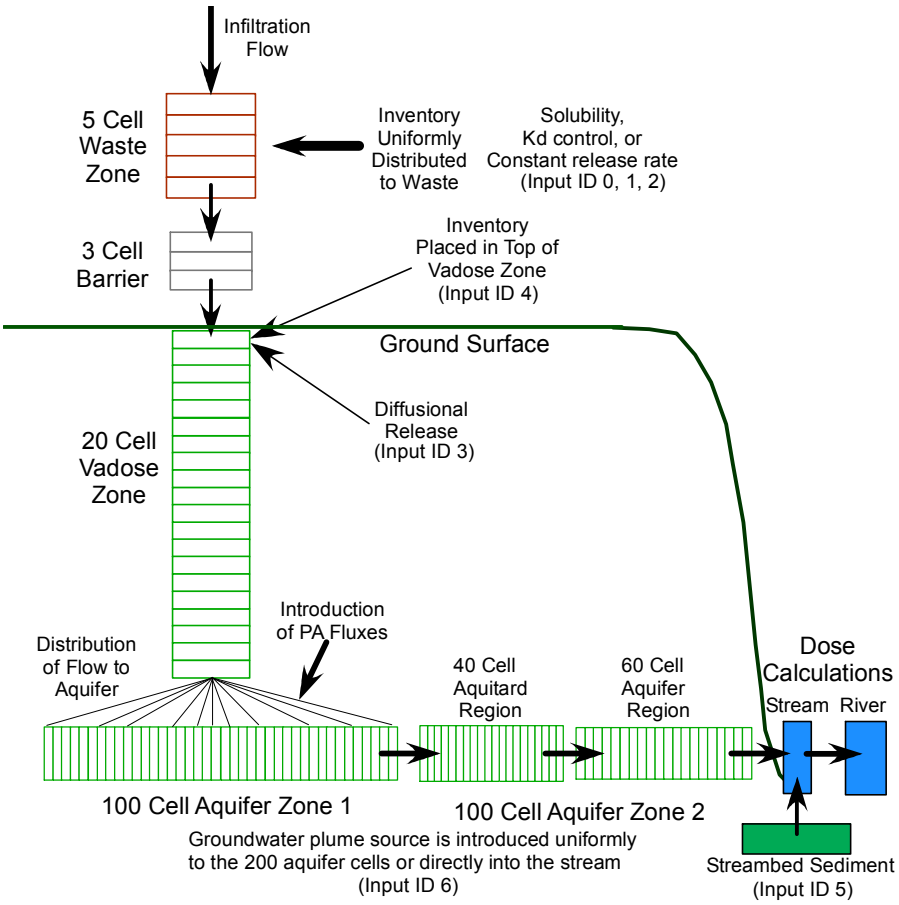


Figure 9-15. Schematic Representation of Entire GoldSim™ Contaminant Transport Model

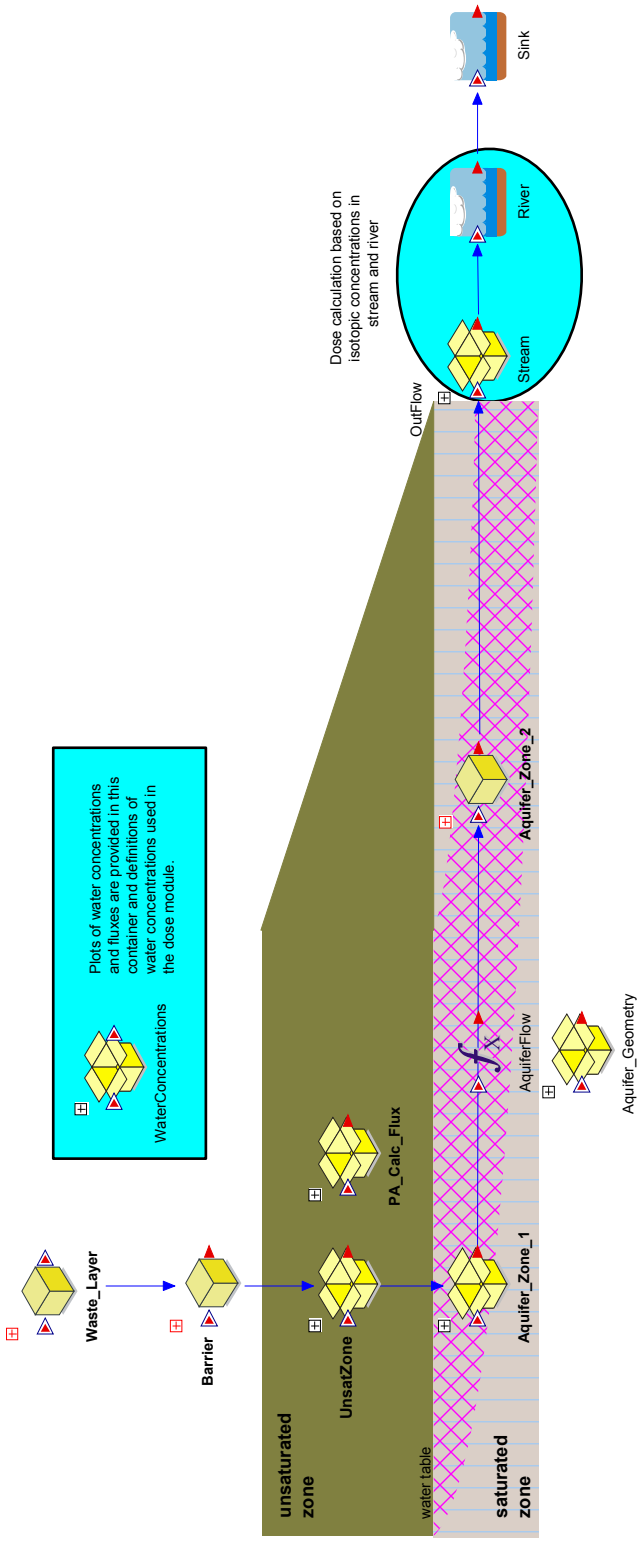


Figure 9-16. GoldSim™ CA Model Representation of Contaminant Transport Pathway

Table 9-6. GoldSim™ CA Model Release Mechanisms

Release Mechanism	Input ID	Description
Solubility controlled	0	Source inventory is released into the waste cells, K_ds are set to zero within the waste zone and the radionuclide concentration in the water phase within the waste cells is determined by solubility.
K_d controlled	1	Source inventory is released into the waste cells, waste K_ds are set by input distributions and water solubility is assumed to be infinite.
Fixed release rate	2	Source inventory is released into the waste cells at a constant rate specified through model input, K_ds are set to zero within the waste zone and solubility is assumed to be infinite.
Diffusion controlled	3	Source inventory is released into the top cell of the vadose zone through diffusion. Water solubility is assumed to be infinite.
Surface release ("generic" release scenario)	4	Source inventory is released into the top cell of the vadose zone. Transport through the unsaturated and saturated zones is K_d controlled. Water solubility is assumed to be infinite.
Streambed sediment source	5	Source inventory is released into the sediment cell connected to the stream. Water solubility is assumed to be infinite.
Groundwater plume source	6	Source inventory is released uniformly into the aquifer cells or directly into the stream. Water solubility is assumed to be infinite.

9.3.1.1 Waste Zone

The waste material is modeled in GoldSim™ container Waste_Layer with a one-dimensional set of five mixing cells. When source release mechanisms invoking solubility, K_d, or fixed release rate control are selected (Table 9-6, release mechanisms 0, 1, 2), the inventory is released uniformly into the five waste zone cells. To use more than one release mechanism for a single source, the source inventory must be divided into separate portions that will use particular release mechanisms and each portion run separately. The rate of water flow through these cells is controlled by the infiltration rate. The cell composition is calculated as a mixture of sandy soil and clayey soil properties where the clay fraction is specified through model input. This gives the user some limited flexibility in specifying the waste physical properties. Because we are not concerned with the concentrations of contaminants in the groundwater and only need to calculate the rate of transport from the source to the stream, the waste zone typically has an arbitrary area. The exception to this is solubility-controlled releases where the entire volume of water must be accounted for and hence the actual area of the waste zone must be used.

9.3.1.2 Barrier

Three mixing cells were placed between the waste zone and the unsaturated soil (vadose zone) to simulate the presence of a concrete base that the waste may have been placed on. Material properties in these cells are specified as a mixture of concrete and clayey soil properties. Therefore, if there is no barrier present, the impact of this layer can be effectively eliminated by specifying clayey soil properties and a small layer thickness.

9.3.1.3 Vadose Zone

The unsaturated soil layer is modeled using 20 mixing cells connected in a one-dimensional flow path. Material properties in these cells are specified as a mixture of clayey soil and sandy soil properties. This feature was used to model some number of cells at the top of the vadose zone as clay and the lower cells as sandy soil. The number of clay cells can be zero. The total length of the vadose zone and the length of the clay region are specified through model input.

Release mechanism 4 (Table 9-6) places the source inventory directly into the top cell in the vadose zone. This is termed the “generic release” mechanism and was used for all of the CA sources to obtain an initial evaluation of the impact from the source. Those sources having a significant impact on dose were then rerun using more detailed modeling of the source release and transport.

Release mechanism 3 (Table 9-6) specifies diffusion controlled release. This was only used to calculate release of tritium from building slabs remaining after the decommissioning of certain tritium facility buildings. Because these slabs would be in direct contact with the top of the vadose zone, diffusion controlled releases were also linked directly to the top cell in the vadose zone.

9.3.1.4 Aquifer

As shown in Figure 9-15, the one-dimensional aquifer region was divided into two parts with 100 mixing cells in each part. Flow out of the vadose zone can be distributed over the first 100 aquifer cells to represent the footprint of the source unit above the aquifer. The distribution is entered through the model input as an array of 100 fractions, any of which can be zero. It is up to the user to ensure that the sum of the fractions is one. The first 100 cells in the aquifer are modeled as saturated sandy soil. The flow rate in the aquifer is specified through the model input.

The second 100 cells were subdivided into two regions. The first 40 cells allow the user to specify the solid material as either saturated clayey soil or saturated sandy soil. This allows modeling the presence of an aquitard layer in the aquifer flow path. All of the clay identified in the aquifer flow path is collected in these cells so the model presents a somewhat simplified picture of the actual physical situation. The cell size in this region can be different from that used in the other aquifer regions to better represent the thickness of the clay layer, if any. For added flexibility, zero to 40 of these cells can be designated as clay, as required. Flow through the clay layer can also be different from that through the saturated sandy soil. The last 60 cells in the second aquifer region are assumed to be saturated sandy soil. These cells can be different in size from those in the other aquifer regions. All aquifer parameters are specified through the model input.

All of the aquifer cells have a cumulative flux input that can be used to introduce some inventory directly into each cell. This feature was used to model groundwater plumes (Table 9-6, release mechanism 6). For the first 100 aquifer cells this feature was also used to introduce contaminant fluxes extracted from PA calculations directly into the cells. The PA flux was distributed to the cells using the flow distribution fractions described above.

9.3.1.5 Stream and River

Flow from the aquifer empties into a mixing cell in the Stream container. This is the first cell where a dose calculation is made and, therefore, the first cell where the contaminant concentration must be calculated. Stochastic distributions of flows for the streams of interest to the CA are available in the Transport container shown in Figure 9-12 of Section 9.2.2. A list of the streams along with the identifying number used in model input is given in Table 9-7. Flow distribution parameters are loaded from an Access database. The flux of contaminant out of the aquifer (e.g., Ci/yr) is divided by the stream flow (e.g., m^3/yr) to obtain a contaminant concentration (e.g., Ci/m^3) that is then passed to the dose module where the recreational dose is calculated. For the deterministic base-case calculations, mean stream and river flow rates were used.

Table 9-7. Streams and Rivers Included in CA.

Stream or River	Acronym	Model ID
Upper Three Runs	UTR	1
Lower Three Runs	LTR	2
Fourmile Branch	FMB	3
Steel Creek/Pen Branch	SC/PB	4
Savannah River	SR	5

The stream cell is also connected to a sediment cell. When release mechanism 5 (Table 9-6) is specified, the inventory is placed into the sediment cell and from there transferred to the stream as explained in the streambed source release modeling discussion in Section 9.3.1.6.8. Flow from the stream passes to the river cell where another concentration is calculated based on the specified river flow. The river concentration is passed to the dose module for the residential dose calculation.

9.3.1.6 Source Release Models

Embedded within the GoldSim™ Disposal_Unit container are the various source release mechanisms listed in Table 9-6. Some of these release mechanisms have already been alluded to in the preceding discussion. The following subsections describe each of these in more detail.

9.3.1.6.1 Generic Release

All waste sites and facility CA sources (does not include streambed and tritium plume sources) were initially evaluated by releasing the inventory directly into the first cell in the vadose zone. We have termed this the "generic" release mechanism. The purpose of this was to identify sources that did not produce significant doses. If the total dose (i.e., dose from all radionuclides and both exposure scenarios) produced by the generic release was less than 0.1 mrem/yr, no additional modeling was performed for this source and the dose calculated from the generic release was used for the assessment. If the total dose was greater than 0.1 mrem/yr, more detailed modeling of the source was performed to account for actual release mechanisms and transport barriers.

9.3.1.6.2 Infiltration Rate Calculations

As a part of the source release modeling, infiltration through the waste zone was in some cases reduced to account for the presence of engineered caps over the waste. Tables of infiltration rate as a function of time for background infiltration and infiltration through clay, geosynthetic, and concrete caps were provided by Phifer and Dixon (2009) as shown in Appendix C, Table C-5. Average cap infiltration rates could be introduced into the GoldSim™ CA model as either data tables that the model would use to linearly interpolate infiltration values at each calculation time step or as fits to functions that the model would use to directly calculate infiltration rates at each time step. The approach of fitting infiltration rates to empirical functions was used in the CA model.

9.3.1.6.2.1 Infiltration Model Development

Equation 9-1 was found to provide a good fit to the average infiltration rate data for all three caps. Originally the power on the scaled time was also used as a fitting parameter. However, in all cases, the best fit power was close to three so it was decided to simplify the functional form by keeping the power constant. Average infiltration rates were then fit using the single coefficient K. Results of the fitting are summarized in Table 9-8 and the fits are compared to the model generated points in Figure 9-17 through Figure 9-19. As can be seen from the figures, the functional form in Equation (9-1) gives a very good fit to all of the infiltration rates:

$$Q = Q_0 + \left(Q_f - Q_0\right)\left[\frac{\left(t/100\right)^3}{K + \left(t/100\right)^3}\right] \tag{9-1}$$

where

Q	Average infiltration rate, in/yr
Q_0	Average infiltration rate at time 0
Q_f	Average infiltration rate at final time
t	Time, yr
K	Fitting coefficient, yr^3

Table 9-8. Cap Infiltration Models Obtained by Minimizing the Sum of Absolute Errors Squared

Cap Type	Q_0	$(Q_f - Q_0)$	K	$\Sigma(err)^2$
Clay	0.362	13.731	135.23	0.341
Geosynthetic	0.00088	11.669	549.97	0.679
Concrete	0.061	13.691	1272.25	1.521

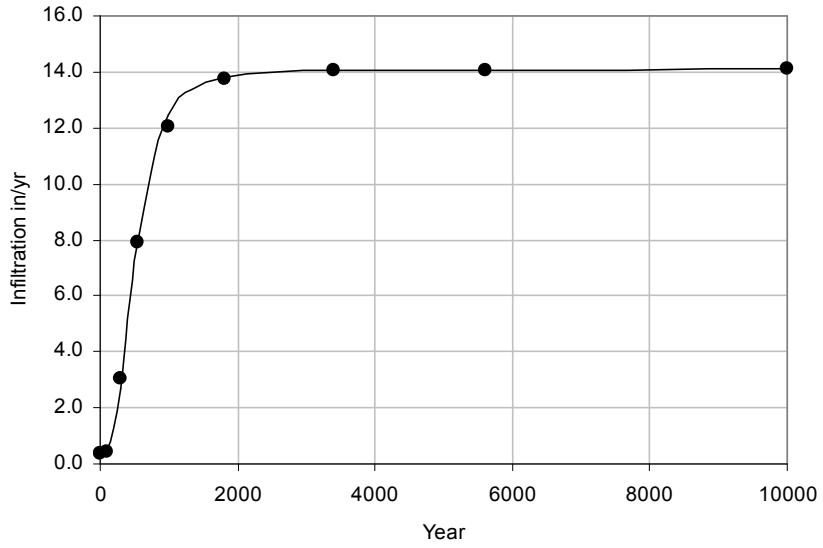


Figure 9-17. Fit of Model-Generated Infiltration for a Clay Cap to Empirical Function.

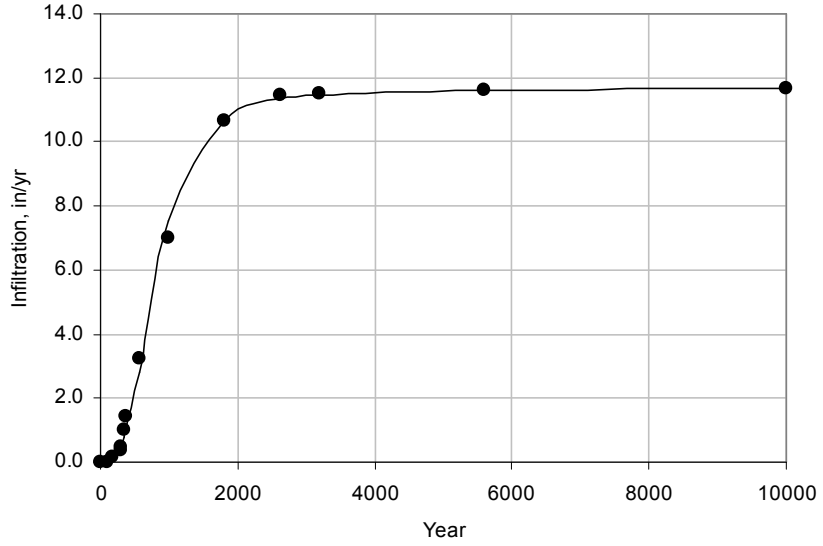


Figure 9-18. Fit of Model-Generated Infiltration for a Geosynthetic Cap to Empirical Function

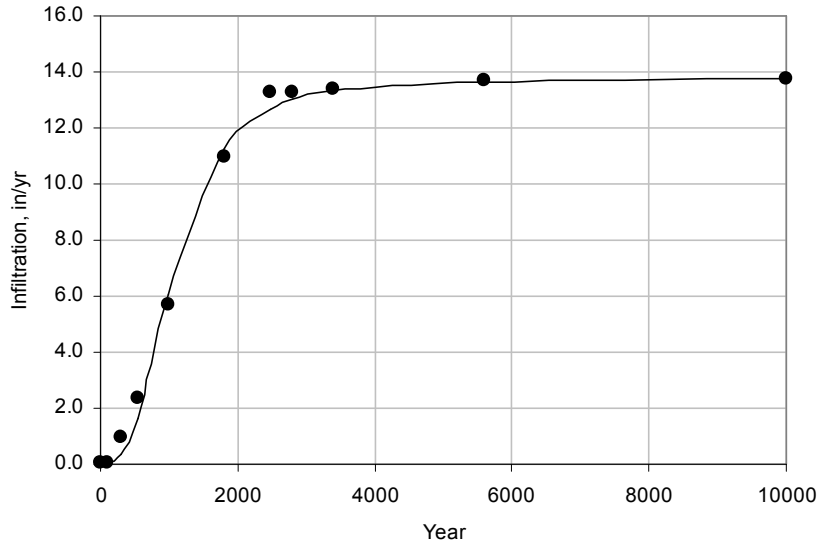


Figure 9-19. Fit of Model-Generated Infiltration for a Concrete Cap to Empirical Function

9.3.1.6.2.2 Infiltration Model Implementation

The infiltration calculations are collected in container \Transport\Infiltration in the GoldSim™ CA model. Contents of this container are shown in Figure 9-20. Functions ClayCap, GeoSynCap and ConcreteCap contain specific versions of Equation (9-1) with the parameters set for the indicated cap types. Function CapTime calculates the time from cap placement scaled by dividing by 100 and function CapTime_3 raises the scaled time to the third power. The cap type and the time of placement are set through the model input. Switch CapInfiltration selects the infiltration rate used in model calculations.

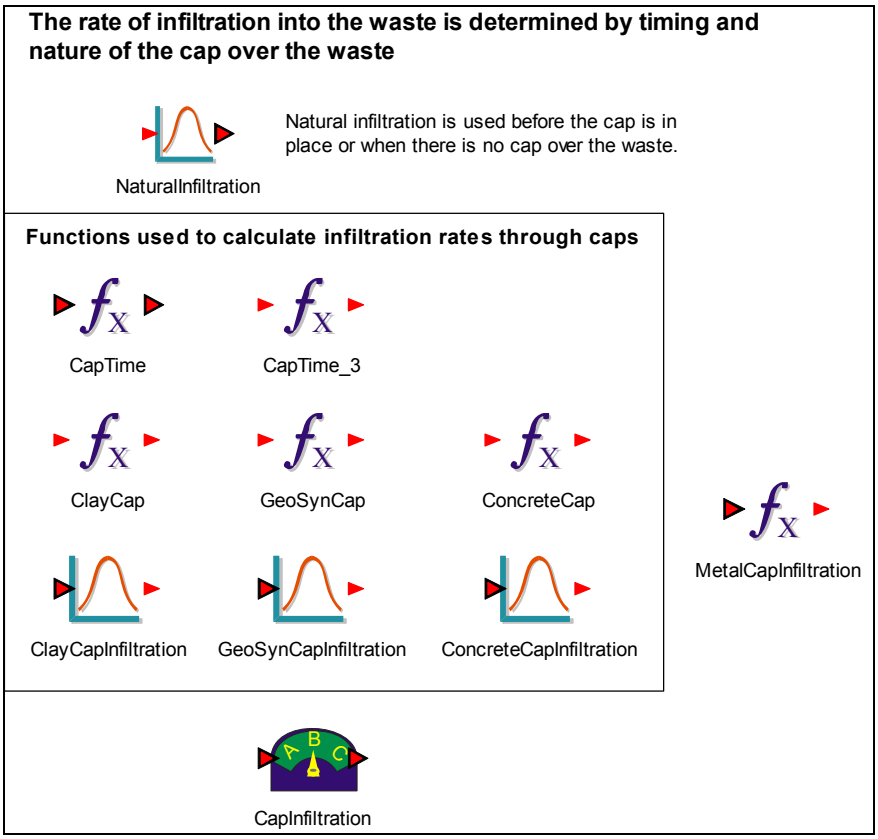


Figure 9-20. Contents of Infiltration Container

The function MetalCapInfiltration was used for source release modeling of NRCDA 643-7E where the inventory is contained in a metal storage cask buried beneath a geosynthetic cap. The storage cask is expected to maintain its integrity for 750 years during which there would be no infiltration through the waste material (WSRC 2008a). The MetalCapInfiltration function sets the infiltration rate to zero for the first 750 years following cap placement and then reports the infiltration rate calculated by the GeoSynCap function.

For probabilistic calculations, which are documented in Section 9.4.3, distributions for the infiltration rates are provided in stochastic elements NaturalInfiltration, ClayCapInfiltration, GeoSynCapInfiltration, and ConcreteCapInfiltration. All of these elements are assumed to be normal distributions truncated at $\pm 3\sigma$. Table 9-9 (also provided in Appendix C, Table C-5 along with infiltration distributions for infiltration through clay, geosynthetic, and concrete caps with time) gives the average and standard deviation for the distribution of the 1,000-year average background infiltration rates obtained from an analysis of historical data (Shine 2008). It was assumed that the standard deviation is the same fraction of the average value for all infiltration rates. With a constant ratio of standard deviation to mean value, the stochastic elements for the infiltration rates are easily created by simply multiplying the mean value by the factors shown in the last three columns of Table 9-9. That is, for all infiltration distributions: the standard deviation is equal to the mean value multiplied by 0.011, the minimum is equal to the mean value multiplied by 0.966, and the maximum is equal to the mean value multiplied by 1.034. Because the mean value is a function of time the stochastic distributions will change with time.

Table 9-9. Background Infiltration Rate (Shine 2008)

Average Infiltration (in/yr)	1,000-year Mean Standard Deviation (in/yr)	Minimum (-3σ) (in/yr)	Maximum (+3σ) (in/yr)	*σ/μ*	*min/μ*	*max/μ*
15	0.17	14.48	15.52	0.011	0.966	1.034

9.3.1.6.3 Diffusion Controlled Releases

As a part of source release modeling for the CA program, it was necessary to model the diffusion controlled release of tritium from concrete slabs. This model was applied to tritium retained in concrete slabs remaining after decommissioning of buildings 232-H, H-Area New Manufacturing (HANM), H-Area Old Manufacturing (HAOM), Tritium Extraction Facility (TEF) and 264-2H. The final end state for these buildings is either in situ disposal as a massive grout filled concrete structure or demolishing the building to the concrete slab. In either case, it is assumed that the residual tritium contamination is distributed uniformly in the original building slab and is released by diffusion from the bottom surface of the concrete into the surrounding soil.

9.3.1.6.3.1 Diffusion Model Development

Initially it was planned to model the building slab as a finite solid of thickness x as illustrated in Figure 9-21. The approach taken was to neglect diffusion from the top and sides of the slab. The top surface of the slab would be exposed to the atmosphere or grout so neglecting tritium loss from the top increases the tritium concentration within the slab and increases the flux into the soil from the bottom surface. In reality, there will also be some tritium loss from the sides of the slab which may be at least partially buried in the soil. This loss was conservatively neglected in the model.

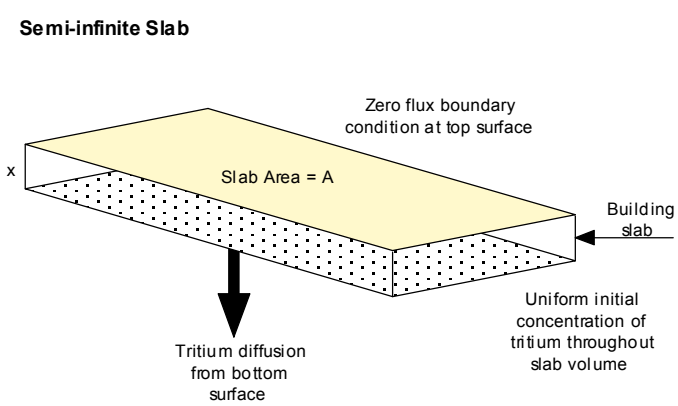


Figure 9-21. Illustration of a Building Slab of Finite Thickness

Assuming diffusion from only the lower face of a slab of thickness x with a zero flux boundary condition at the upper face, the solution for the cumulative fractional release of tritium is given by the infinite series (Sherwood et al. 1975):

$$CFR = 1 - \frac{8}{\pi^2} \sum_{n=1}^{N} \frac{e^{-(2n-1)^2 \beta}}{(2n-1)^2} \tag{9-2}$$

In Equation (9-2) the dimensionless parameter β is:

$$\beta = \left(\frac{\pi}{2x} \right)^2 D_{H3} t \tag{9-3}$$

In Equation (9-3), x is the slab thickness (m), D_{H3} is the diffusion coefficient for tritium in concrete (m^2/yr), and t is time in years from the start of the release. The instantaneous fractional release rate (yr^{-1}) of tritium is obtained by taking the derivative of Equation (9-2) with respect to time which gives the expression:

$$\dot{Q}_{H^3} = \frac{2 D_{H3}}{x^2} \sum_{n=1}^{N} e^{-(2n-1)^2 \beta} \tag{9-4}$$

The slab thickness ranges from 0.33 ft to 6 ft for the five tritium buildings of interest. The convergence of Equation (9-4) was tested over this range of thicknesses over times ranging from 0.1 to 10,000 years. During the testing it was observed that the series solution converged very slowly for large thicknesses and small times. Therefore, it was decided to also test the model of diffusion from a semi-infinite slab. That is, the building slab was modeled as a solid region of infinite thickness with the same uniform initial concentration of tritium as that in the finite slab. The finite and semi-infinite slab models are illustrated in Figure 9-22.

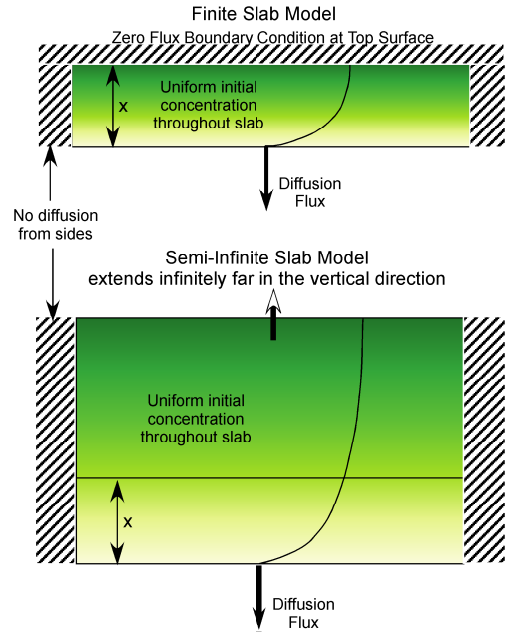


Figure 9-22. Illustration of Modeling Building Slabs as Regions of Finite or Infinite Thickness

The analytical solution for the instantaneous diffusion flux (N_{H3}, Ci/m^2-yr) from a semi-infinite slab is (Sherwood et al. 1975):

$$N_{H3} = -c_0 \, \varepsilon \sqrt{\frac{D_{H3}}{\pi \, t}} \tag{9-5}$$

In Equation (9-5), ε is the concrete porosity and c_0 is the initial concentration of tritium in the slab which is calculated from the total tritium inventory (I_{H3}) and the volume of the actual building slab (with horizontal face A) as:

$$c_0 = \frac{I_{H3}}{A \, x \, \varepsilon} \tag{9-6}$$

Combining Equations (9-5) and (9-6), the instantaneous fractional release of tritium is calculated as:

$$\dot{Q}_{H3} = -\frac{N_{H3} \, A}{I_{H3}} = \frac{1}{x} \sqrt{\frac{D_{H3}}{\pi \, t}} \tag{9-7}$$

The solutions for the rate of tritium release obtained by applying Equations (9-4) and (9-7) are compared in the following figures (Figure 9-23 through Figure 9-25) for three values of the slab thickness. In the figure legends, N is the number of terms used in the series solution in Equation (9-4).

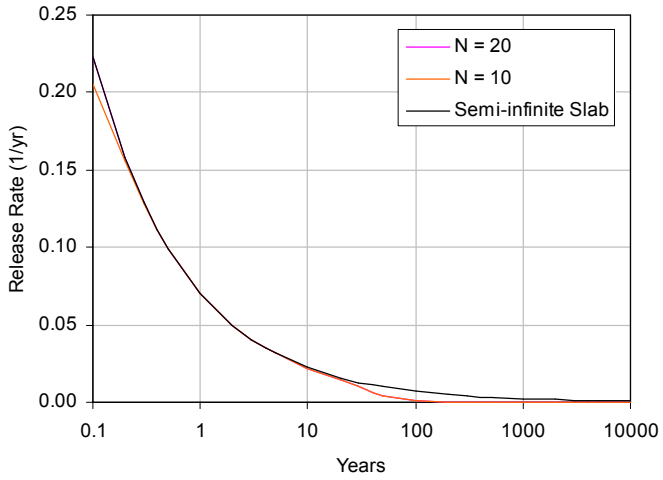


Figure 9-23. Tritium Release Rate Predictions for a 0.33 ft Slab

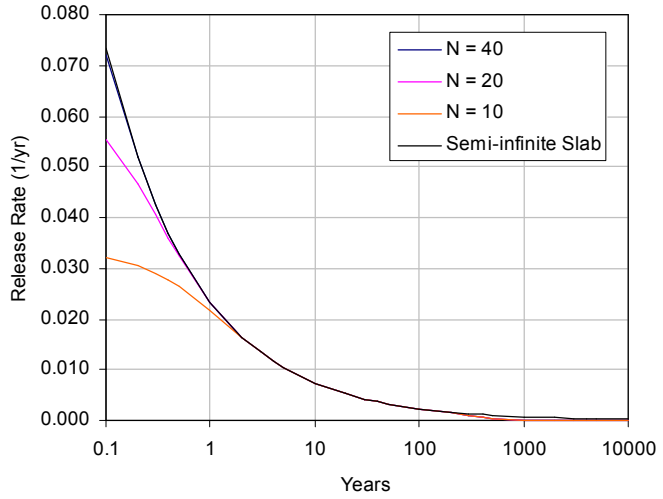


Figure 9-24. Tritium Release Predictions for a 1.0 ft Slab

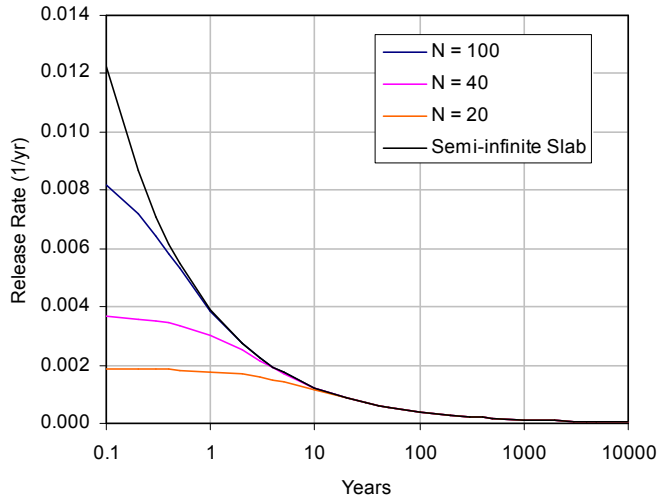


Figure 9-25. Tritium Release Predictions for a 6.0 ft Slab

In all cases, the semi-infinite slab solution predicts either a higher release rate than from the finite slab solution or the difference between the two is negligible. At long times, the series solution gives a more accurate prediction of actual diffusion controlled release rates but the difference between the two solutions is relatively small. At short times with thicker slabs the series solution is slow to converge. As shown in Figure 9-25, over 100 terms of the series solution must be used to obtain an accurate solution at short times for a six foot slab thickness. Although it is not possible to see because the curves overlap, in Figure 9-23, the 20 term series solution follows the semi-infinite slab solution at short times and then the 10 term series solution at long times. The slow series convergence of the finite slab model (Equation 9-4) results in long model run times which are not warranted due to the good approximation obtained using the semi-infinite slab model (Equation 9-7). Based on these results, it was decided to use the semi-infinite slab model to calculate tritium diffusion from concrete slabs. From the comparisons shown above, this approach is demonstrably conservative.

9.3.1.6.3.2 Diffusion Model Implementation

The diffusion calculation was implemented in GoldSim™ container \Disposal_Unit\UnsatZone\Diffusion_Calc which has the structure shown in Figure 9-26. The tritium diffusion coefficient is fixed in the data element to be 5.0E-08 cm^2/s (Flach et al. 2007). Function DiffusionRate calculates the fractional release rate of tritium using Equation (9-7). The source tritium inventory is placed in cell DiffusionCell at the specified release time. A direct-transfer-mass-flux-link is made between the DiffusionCell and the first cell in the vadose zone. The rate of transfer between the cells is the fractional release rate calculated in function DiffusionRate. This structure allows the GoldSim™ model to see the entire inventory at the initial time of release so that tritium decay is correctly accounted for.

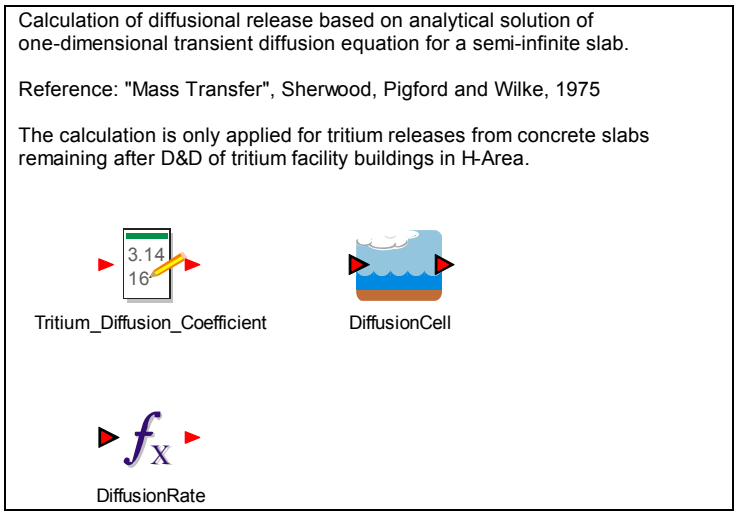


Figure 9-26. Contents of GoldSim™ Diffusion_Calc Container

9.3.1.6.4 Metal Corrosion

The release of radionuclides from activated metals is modeled as corrosion of the metal at a constant rate. For CA modeling, this was applied to releases from the reactor vessels (C, K, L, P and R) and from materials stored on NRCDA 643-7E. The modeling is in fact relatively simple. The inventory is first placed into a holding cell where decay and progeny in-growth occur and is released from the cell at a constant rate to represent corrosion of the metal holding the inventory. Material released from the holding cell is distributed uniformly to the five waste zone cells. This conceptual model is illustrated in Figure 9-27.

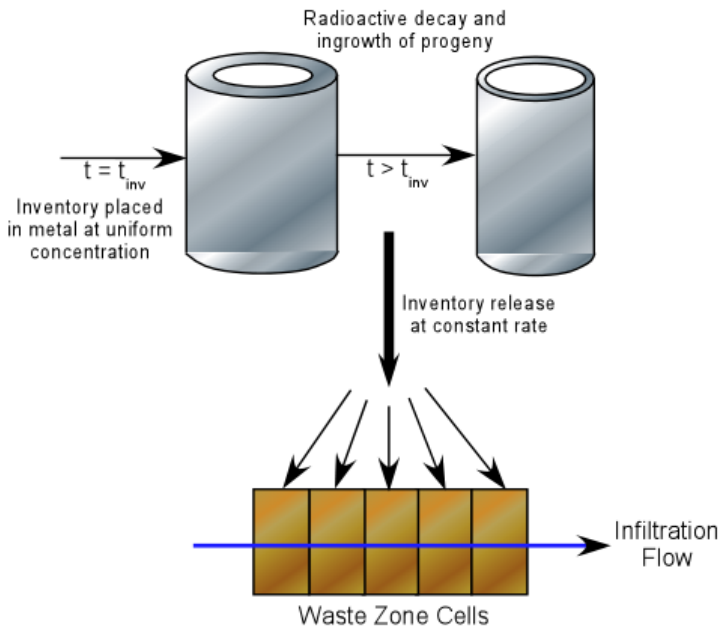


Figure 9-27. Illustration of Corrosion Model

9.3.1.6.4.1 Corrosion Model Implementation

The corrosion model is implemented in GoldSim™ container \Disposal_Unit\Waste_Layer\Source_Calc. The contents of this container are shown in Figure 9-28. The inventory for corrosion release is placed in cell "CorrosionCell" at the initial inventory time specified in the model input. The sole purpose of container "CorrosionCell" is to hold the inventory so that decay and in-growth are accounted for. Material is transferred from the "CorrosionCell" using direct-transfer-mass-flux-links to each of the five cells in the waste zone with an equal fraction placed in each cell. The fractional corrosion rate is assumed to be constant and is entered into the model through the input. The function "CorrosionRate" converts this constant release rate into a vector of dimension species so that each radionuclide is released at the same rate.

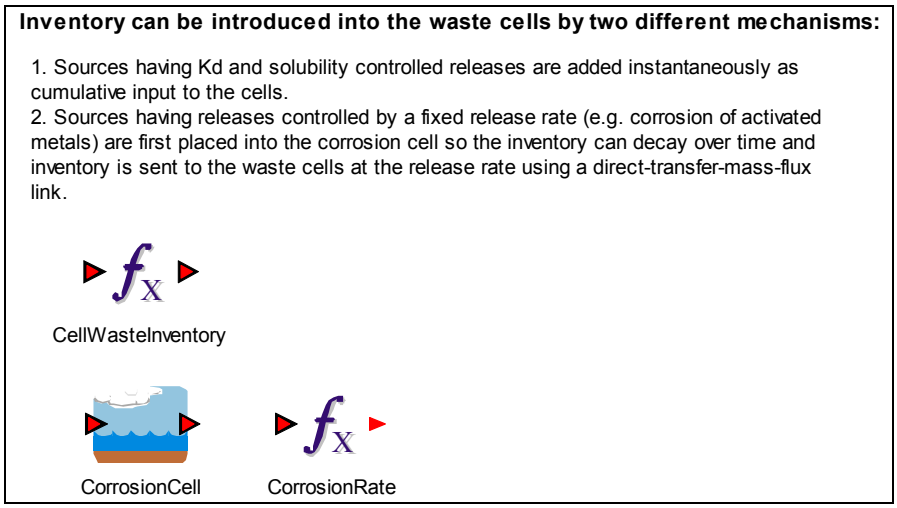


Figure 9-28. Contents of \Disposal_Unit\Waste_Layer\Source_Calc Container

9.3.1.6.4.2 Corrosion Parameters

Following the method described in the ELLWF PA Chapter 5 (WSRC 2008a), material contained in NRCDA 643-7E was modeled as six different metal components. The report by Yu et al. (2002, Table 3.1-1) gives a breakdown of the fraction of metal in each part of the NRCDA 643-7E material, the effective part thickness, and the fractional corrosion rate. This information is reproduced in Table 9-10 below. Based on Hanford data (1993), the corrosion rates of Inconel and Zircaloy were conservatively estimated to be 1.0E-05 in/yr and 1.0E-06 in/yr, respectively. Accounting for corrosion from both sides of the metal part doubles this rate as shown in Table 9-10. The fractional corrosion rate is obtained by dividing the two-sided corrosion rate by the effective thickness. The report by Yu et al. (2002, Table 2.3-1) also provides a distribution of certain isotopes between the metals on Pad 1. This information, which was used to distribute the total inventory among the metal parts, is reproduced in Table 9-11 below. CA isotopes not included in this distribution were assumed to be 100% in the faster corroding Inconel. The approach to modeling NRCDA 643-7E described above was used in the ELLWF PA (WSRC 2008a).

Table 9-10. Corrosion Parameters for Inconel and Zircaloy in NRCDA 643-7E

Component Part	Metal	Effective Thickness [in]	Metal in Part [fraction]	Corrosion Rate [in/yr]	Fractional Corrosion Rate [yr^{-1}]
1	Inconel	0.543	0.6467	2.0E-05	3.68E-05
2	Inconel	0.095	0.1617	2.0E-05	2.11E-04
3	Inconel	0.714	0.1557	2.0E-05	2.80E-05
4	Inconel	0.726	0.0359	2.0E-05	2.76E-05
5	Zircaloy	0.375	0.4496	2.0E-06	5.33E-06
6	Zircaloy	0.150	0.5504	2.0E-06	1.33E-05

Table 9-11. Radionuclide Inventory Distribution for NRCDA 643-7E

Radionuclide	Inventory Fraction in Zircaloy
C14	0.88
Ni59	0.17
Zr93	1.00
Nb94	0.27
Mo93	0.05
Tc99	0.26
Pu239	0.99
Pu240	1.00

The corrosion rate for the stainless steel reactor vessels was taken from the report by Council (2009, pp 23-24). This report gives a corrosion rate for 304 Stainless Steel in a grout environment as 0.0006 lb/yr/ft^2 and a steel density of 0.285 lb/in^3. The reported rate is for corrosion from both sides of a steel plate. Assuming an average steel thickness of 0.5 inch (Council 2009 and WSRC 2008c) the values translate into a fractional corrosion rate of 0.43 μm/yr. This corrosion rate was applied for all five reactor vessels (C, K, L, P and R).

9.3.1.6.5 <u>I129 on Berl Saddles Wasteform</u>

In at least two of the sources (MWMF and ORWBG), it has been noted that much of the I129 is captured on silver nitrate impregnated berl saddles (WSRC 1997b). Because iodine readily reacts with silver to form silver iodide, it is believed that the I129 on the berl saddle waste form is almost entirely in this form. Silver iodide is very insoluble and therefore considerably less mobile than most I129. Hawkins (1983) gives the solubility of AgI as 2.80E-06 g/L in cold water. This value represents a solubility for I129 associated with the AgI of 1.53E-09 g/mL. In WSRC 1997b, the solubility of AgI in SRS groundwater conditions was estimated to be 2.2E-10 moles/L which represents a solubility of 2.80E-11 g/mL for I129 associated with AgI. The larger value of 1.53E-09 g/mL was conservatively chosen for the CA. This release mechanism was only applied in the ORWBG because an estimate for the split between I129 on berl saddles and generic I129 could not be accurately determined for the MWMF.

9.3.1.6.5.1 I129 Special Wasteform Implementation

If waste-form solubility is available in the Parameters.xls workbook for a given nuclide, the CA model will use it preferentially over the K_d when the "Release Mechanism" flag in the Disposal_Units workbook for the given unit is set to "0", signifying a solubility-controlled release. Therefore, a source cannot be run with a nuclide that has both K_d-controlled and solubility controlled components. Consequently, the ORWBG was split into two disposal units for the purposes of the CA. One source contained the assumed solubility-controlled I129 on berl saddles, and the second source contained the remaining K_d-controlled I129, as well as the remainder of the inventory for all other CA nuclides. The dose results from these two units can be added together to give a final ORWBG dose. Using the release mechanism flag of 0 changes the water solubility of elements in the waste zone from the default of infinite solubility to solubility values specified in the Parameters.xls file. Currently, only the solubility for iodine has a unique value specified.

For the purposes of K_d-controlled releases, an arbitrary value for the footprint of the disposal units is used in the CA model. This approach is acceptable for K_d-controlled releases because the important parameter for transport is the ratio of solid matrix material to groundwater. The exact amount of each material is not important in the CA model for transport purposes. This conclusion was verified in the model by changing disposal unit footprints and noting that doses do not change. However, for solubility-controlled releases, the total amount of water present is important and to make sure that the correct amount of water flow into the waste zone is used, the correct disposal unit footprint (309624 m^2 for the ORWBG) must be used. Because solubility control is currently used only for I129 in the ORWBG, that is the only disposal unit affected by this issue.

9.3.1.6.6 <u>Performance Assessment Water Table Flux</u>

For PA facilities requiring source release modeling, isotope fluxes to the aquifer taken directly from the PA calculations were used in the CA model. This method ensures that the CA calculation is consistent with the PA as much as possible. Use of the PA flux accounts for source release factors in the CA without specific modeling, because the source release has already been taken into account when calculating the PA fluxes. Time histories of species fluxes from the PA calculation are entered into the CA model where contaminant species enter the aquifer. This method was used for the E-Area Intermediate Level Vault (ILV), E-Area Low Activity Waste Vault (LAWV) and the three components of Z-Area Vault 2. WSRC (2008a) and LWO (2009) are the E-Area and Z-Area PA documents, respectively.

9.3.1.6.6.1 Performance Assessment Flux Implementation

Time histories of species fluxes to the aquifer from PA disposal units requiring source release modeling were collected in Excel spreadsheets. The PA fluxes were either based on a unit Curie inventory or based on the projected disposed inventory. In either case, the fluxes were proportioned to the CA inventory. Flux data, obtained from PORFLOW© output for the pertinent sources and adjusted for the CA inventory, were transferred into the PA_Flux.xls workbook. Each flux table has a separate worksheet within the workbook. Within each worksheet, Column A contains the elapsed time in years, and the remaining columns are fluxes for the 63 isotopic species in the GoldSim™ model master species list (Phifer et al. 2009). Fifty-two of the 63 isotopic species in the master species list are used for CA calculations (Taylor et al. 2008) and none of the non-radioactive species in the master species list are considered in the CA. Fluxes for non-CA species were set to zero in the PA_Flux worksheets.

The GoldSim™ CA model will read and ultimately add together water table fluxes originating from the CA inventory and any flux that is directly input to the model. Therefore, inventories of species that appear in the flux tables must be removed. The GoldSim™ CA model reads the flux time series from the FluxTable worksheet in the PA_Flux.xls workbook. Because there are a number of facilities which have PA fluxes, this worksheet must be rewritten for every facility that is run. For automated runs, this step is accomplished as part of an outside Visual Basic for Applications (VBA) macro that runs the GoldSim™ CA model for a series of sources.

As outlined in Table 9-22 the use of PA fluxes was implemented for the E-Area ILV, E-Area LAWV, and Z-Area Vault 2s. The Saltstone Disposal Facility (SDF) PA species corresponded exactly with the CA species. However, for both the E-Area ILV and E-Area LAWV the following 11 CA radionuclides had CA inventories but did not appear in the PA flux tables: Am242m, Cf249, Cm246, Cs135, Cs137, K40, Ni63, Pb210, Pu240, Sr90 and Th230. In these cases, the inventory not accounted for in the PA fluxes was included using the CA generic release mechanism (see Section 9.3.6). These species, based on the CA generic release mechanism, tend to be less important to the overall dose, and variation in the overall dose from this approximation is likely to be conservative because the contaminants will reach the dose assessment point faster than if the waste form or disposal unit were modeled.

9.3.1.6.7 <u>Groundwater Plumes</u>

Plumes of contaminants exist within the aquifers beneath SRS and are the result of releases to the subsurface of mobile contaminants associated with SRS disposal activities. The most prevalent contaminants in these groundwater plumes are trichloroethylene (TCE) and tritium (H3), both of which were extensively handled in SRS operations and which resulted in process waste that was disposed in near surface facilities. Groundwater plumes exist in all of the major operational areas at SRS, as indicated in Figure 9-29. The plumes illustrated in this figure do not distinguish between TCE and H3 and, in fact, plumes of the two contaminants are co-mingled in many instances.

9.3.1.6.7.1 Location and Extent of SRS plumes

In Figure 9-29, all of the main operations areas are illustrated with an explanation of the principal contaminants associated with groundwater contamination. Of the two principal contaminants, only H3 is of further interest in this CA because the purpose of the CA is to evaluate the potential for an individual to receive a dose from the residual radionuclide contamination that might remain at SRS when it reaches its end-state. The primary locations where H3 is extensively present in the groundwater include the GSA (F-, H- and E-Areas) and the reactor areas, which include C-, K-, L-, P- and R-Areas. Some H3 in the groundwater is also found at TNX and D-Areas.

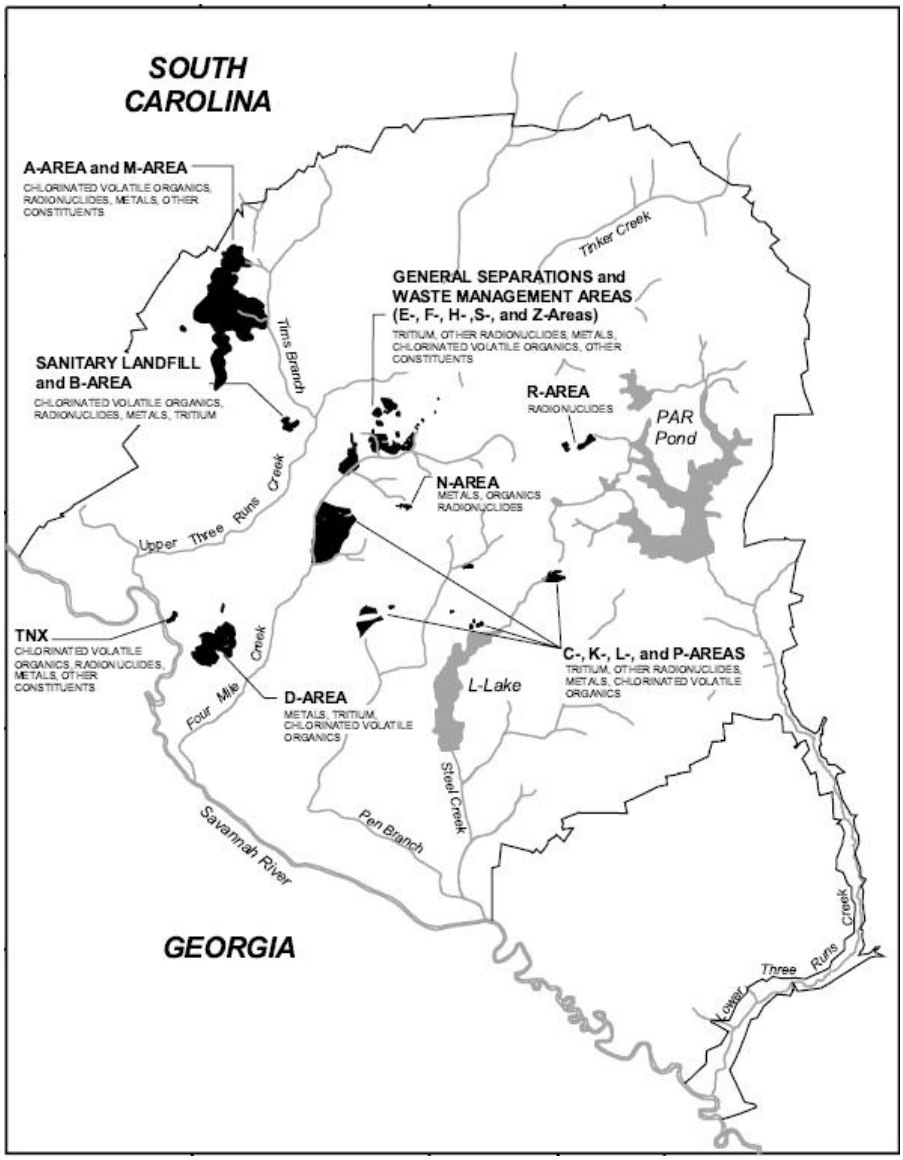


Figure 9-29. Location of Groundwater Plumes at SRS (WSRC 2007b)

Tritium Plumes in General Separations Areas (F-, E- and H-Areas)

More detailed maps of the outline of tritium plumes which occur in the GSA are shown in Figure 9-30. Primarily, these plumes are emanating from the F-Area and H-Area Seepage Basins (FSB and HSB), the ORWBG, and the MWMF. Additionally, two other plumes have been identified and are referred to as the GSA Western plume (near F-Area) and the GSA Eastern plume (near H-Area).

It should be noted that a local groundwater divide runs from east to west and passes through F-Area, the MWMF and H-Area. North of this divide, groundwater is migrating slowly to the north and discharging into Upper Three Runs (UTR) while south of this divide, groundwater is migrating slowly southward and discharging into Fourmile Branch (FMB).

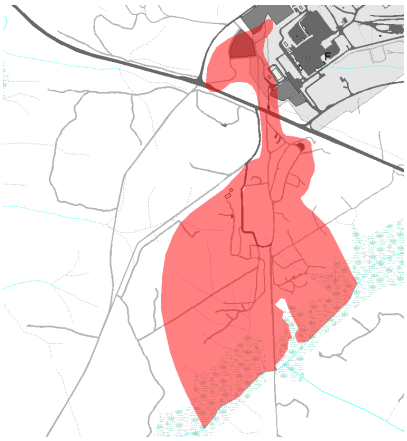
Tritium Plumes in Reactor and Other SRS Operational Areas

Tritium plumes are also known to exist in each of the reactor areas at SRS. At each of these locations, the H3 source may be either the reactor disassembly basin or the seepage basin(s) which received discharge water during the operations period. Illustrations are provided in Figure 9-31 which indicate the aquifer plume outlines where the plumes may be co-mingled TCE and H3, both of which are mobile in the subsurface. Each of these plumes indicates the direction of movement of shallow groundwater and each discharge into the closest perennial stream.

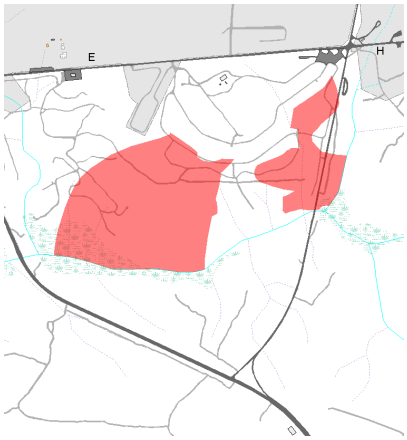
In addition to the reactor plumes, groundwater plumes also exist in D-Area, and the former TNX Area. Plumes in these areas are indicated by the pink shaded areas in Figure 9-32. Tritium in D-Area is in the central portion of the shaded areas and is migrating with the groundwater, moving from the upper right toward the Savannah River. Likewise, the small plume containing H3 beneath TNX-Area is migrating toward the Savannah River.

Tritium Inventories Associated with Groundwater Operable Units

The primary source for tritium groundwater total activity was ERD 2001a, as reported in the CA Inventory Report (Hiergesell et al. 2008). An exception to this is for the L-Area GOU, which has an estimate of H3 in the "Corrective Measures Study/Feasibility Study for the L-Area Southern GW Operable Unit" (WSRC 2007a). This estimate supersedes the estimate that was provided in the ERD 2001a document. The revised H3 estimate represents the inventory for both the Northern and Southern GOU in the vicinity of L-Area. A second exception is the MWMF GOU which was inadvertently omitted in Rev. 0 of the CA Inventory report. Tritium inventories associated with each GOU and utilized in this analysis are presented in Table 9-12 (also see Appendix A Table A-77). The GSA-Western and Eastern plumes are small and tritium inventories have not been developed for them.

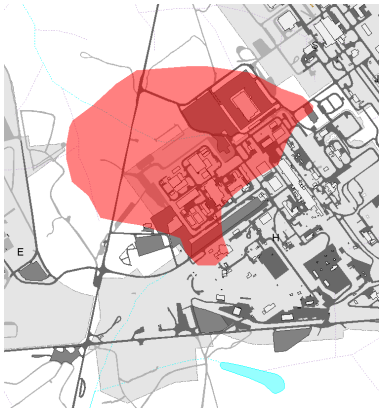


F-Seepage Basin Plume

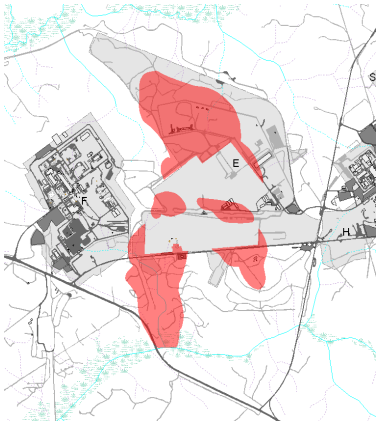


H-Seepage Basin Plume

GSA Area Western Plume



GSA Area Eastern Plume



MWMF Plumes

Figure 9-30. Groundwater Plumes Associated with the GSA

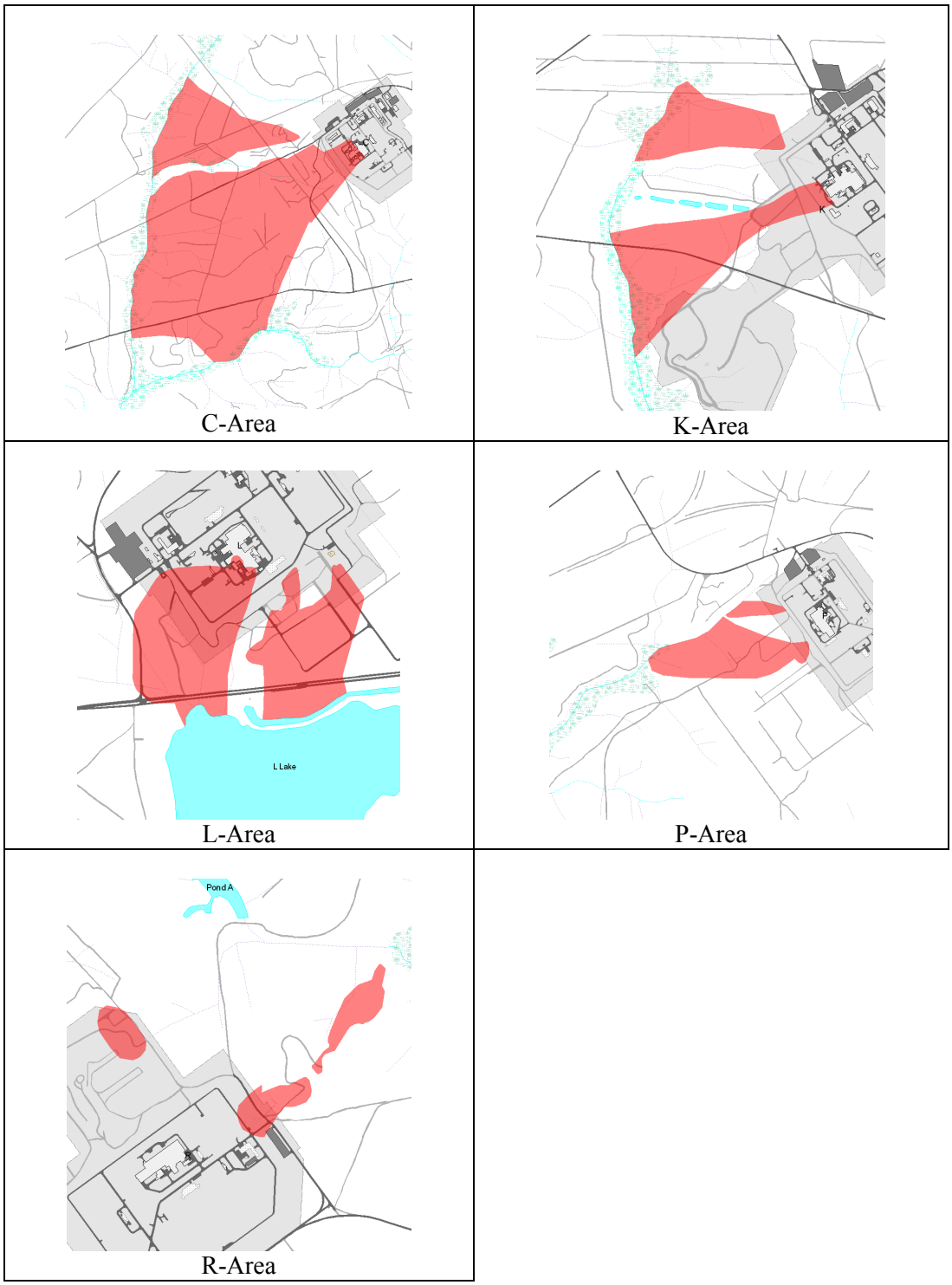
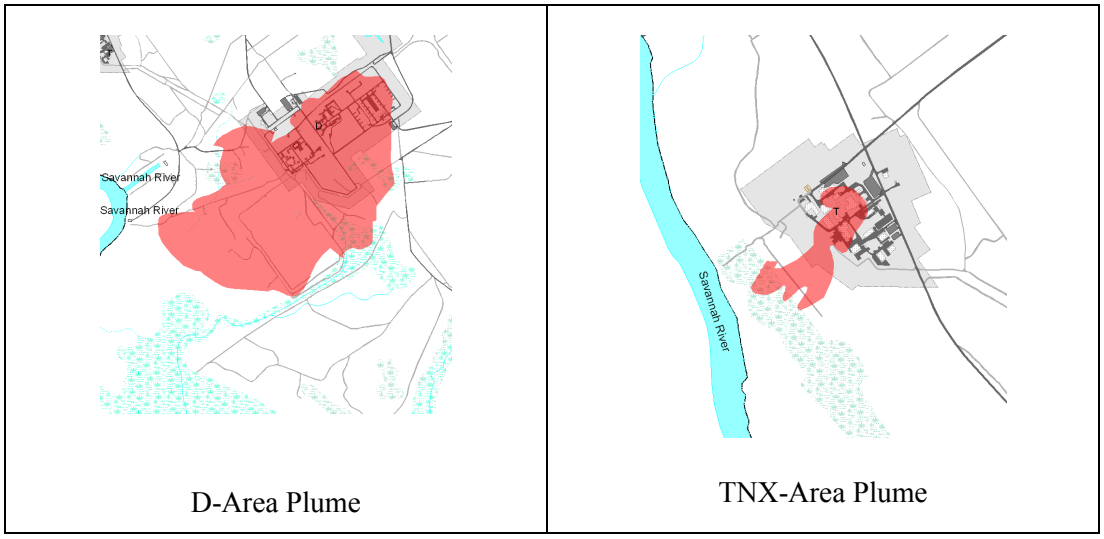


Figure 9-31. Groundwater Plumes Associated with SRS Reactor Areas



D-Area Plume

TNX-Area Plume

Figure 9-32. Groundwater Plumes Associated with D-Area and TNX Area

Table 9-12. Groundwater Operable Unit Tritium Plume Inventory Information

Operable Unit	Tritium (Ci)	Source(s)
F-Area	9.0E+02	F-Area Seepage Basins, IPSL
H-Area	2.4E+01	H-Area Seepage Basins, IPSL
C-Area	1.1E+03	Seepage Basin, Reactor Bldg.
K-Area	2.4E+05	Seepage Basins, Reactor Bldg.
L-Area [1]	3.6E+03	Seepage Basin, Reactor Bldg.
P-Area	1.6E+03	Seepage Basins, Reactor Bldg.
R-Area	1.6E+00	Seepage Basin, Reactor Bldg.
D-Area	3.6E+02	Heavy Water Facilities, Process Bldgs.
TNX-Area	2.0E-01	Process Bldgs
MWMF	4.2E+05	MWMF/LLRWDF/ORWBG
GSA-Western	--	Tank Farms, other
GSA-Eastern	--	Tank Farms, other

[1] L-Area Northern and Southern GOUs are combined into the Southern GOU.

9.3.1.6.7.2 Strategy for Simulating Groundwater Plumes

The general approach for simulating groundwater tritium plumes was to take the calculated inventory, developed for each GOU, and place it within the aquifer portion of the CA GoldSim™ model. Because several potential sources of H3 exist within each of the GOUs, the shortest of the candidate groundwater flow paths for these facilities was selected as the appropriate one for simulating the existing tritium plumes. This strategy is thought to introduce a slight bias toward overestimating the H3 activity reaching the stream, because a shorter flow path and the associated shorter travel times mean that there is less time for H3 to decay. The estimated total tritium inventory was then distributed uniformly along the full length of the selected aquifer flow path, from directly beneath the surface facility to the discharge location at the adjacent surface stream. This approach was utilized for groundwater plumes in the reactor areas, except for part of C-Area, and in D-Area and TNX. The tritium inventory was removed from surface facilities that were the original source of these groundwater tritium plumes because sampling programs indicate that the entire tritium inventory has already leached into the subsurface.

This general approach was not utilized for plumes located in the GSA or in C-Area that discharge into FMB. The majority of the groundwater contamination in the GSA exists south of the groundwater divide and will eventually discharge into FMB. The extensive well monitoring network in this locality provides ample evidence that these plumes are no longer expanding and that groundwater tritium concentrations are no longer increasing, due primarily to the steady or diminishing releases from surface disposal facilities and the relatively short half-life of H3. Stream concentration data from surface water samples collected in FMB at Station A7 are also consistent with these trends, indicating that the rate of H3 being discharged into FMB is diminishing. Station A7 is located downstream from all GSA groundwater tritium sources but upstream from C-Area and therefore reflects the contribution of all plumes discharging into the stream from the GSA. Stream tritium concentrations from Station A7 were normalized versus the rate of stream flow when the samples were collected to estimate the total tritium activity being discharged from the GSA into FMB (Noffsinger 2009). The calculated H3 flux data are shown below in Figure 9-33, with a best-fit regression line extending through the data points and extrapolated to the year 2015. With the exception of a handful of elevated tritium flux data points that occur from late 2005 and into early 2006, and which are linked to a specific construction-remediation project in the GSA, the data points on this graph conform relatively closely to the indicated trend line.

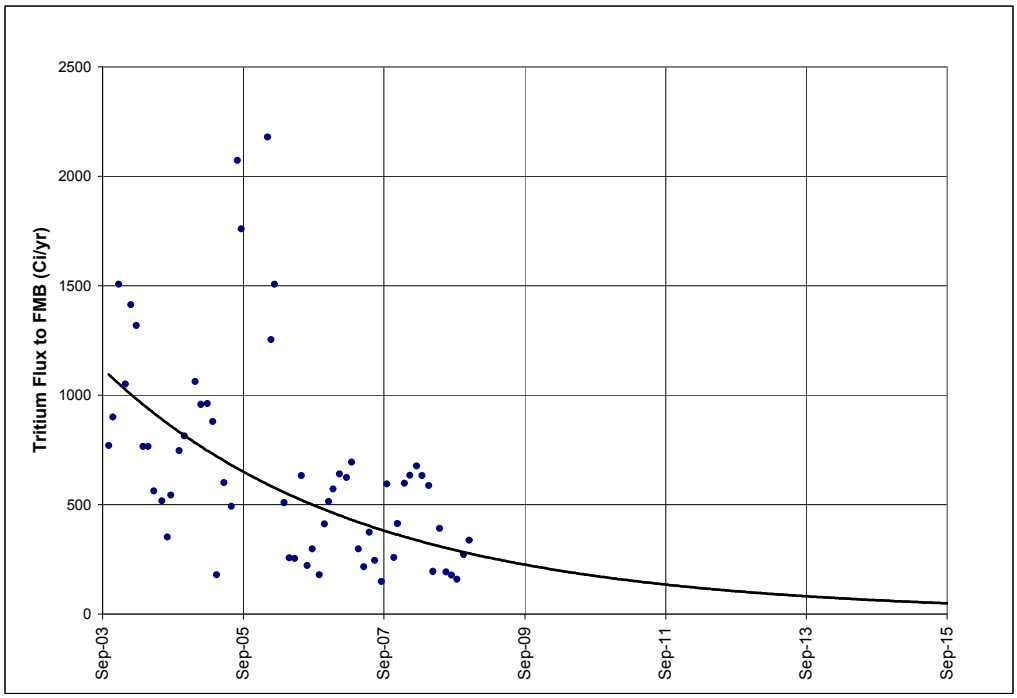


Figure 9-33. Tritium flux to FMB – Based on Stream Concentration at FMB-A7

Because dependable data exist to quantify the rate of H3 discharge from the GSA into the surface water of FMB, sources of H3 that originate in surface disposal facilities south of the divide were estimated collectively, based on the median discharge rate to FMB calculated for 2008 and discharged directly into FMB. The median flux rate, based on 12 samples collected in 2008, is 363 Ci/yr. A spreadsheet quantifying this calculation was provided by Soil and Groundwater Closure Project (SGCP) and the estimate was formally reported in an email from D. C. Noffsinger to R. A. Hiergesell, dated 2/19/2009 and later documented in Noffsinger 2009. For the purpose of simulating the contribution of these sources into the future, the groundwater contribution is assumed to decline from 363 Ci/yr in proportion to the half-life of H3 (12.3 years).

The only other H3 groundwater plumes contributing to FMB originate from the C-Reactor Buildings and seepage basins. An estimate was made in the report "Groundwater Transport Modeling for Southern TCE and Tritium Plumes for the C-Area Groundwater Operable Unit" (WSRC 2001) of the H3 mass discharge to FMB and a tributary stream named Castor Creek. The estimate of the total maximum annual contribution from this groundwater plume is 145 Ci/yr (see also ERD 2001b), with about 120 Ci/yr discharging into FMB and about 25 Ci/yr discharging into Castor Creek. There is also a northern H3 plume emanating from C-Area and discharging into FMB. The report "Groundwater Transport Modeling for the Northern Tritium Plume in the C-Area Groundwater Operable Unit" (WSRC 2003b) describes these plume discharges as a maximum of 3 Ci/yr. For the purpose of simulating future releases to FMB in the CA model, this discharge rate was conservatively assumed to be representative of 2008 and was added to the 363 Ci/yr that enters FMB from the GSA. The total flux to FMB from both C-Area and the GSA is therefore estimated to be 511 Ci/yr in 2008.

In applying this approach to the GSA facilities south of the groundwater divide that contributed H3 to existing tritium plumes discharging to FMB and those surface facilities in C-Area that also contribute H3 to existing tritium plumes discharging to FMB, any H3 in the estimated remaining inventory of these surface facilities was removed so that H3 being discharged to FMB was not "double counted". The strategy for simulating groundwater plumes in the GSA that fall north of the groundwater divide was as follows. The GSA-Eastern GOU and GSA-Western GOU plumes are small and no H3 inventory has been developed. Because analysis conducted by SGCP indicates that these plumes are not expanding and will not reach UTR, they were not included in the CA. In evaluating the MWMF surface releases, the estimated inventory associated with the entire MWMF was apportioned to either the northward groundwater path (UTR) or the southward groundwater path (FMB), based on the percentage of surface area of the disposal facility estimated to fall north and south of the groundwater divide. That portion of the MWMF radionuclide inventory that fell north of the groundwater divide was allowed to retain its H3 inventory when the MWMF-UTR segment was evaluated while that portion of the inventory associated with the southward release path had its H3 inventory removed to avoid "double counting" the H3 released to FMB.

9.3.1.6.7.3 Groundwater Plume Model Development

The approach for developing the analytical model for simulating the migration of H3 in groundwater plumes at SRS within GoldSim™ involved the use of GoldSim™'s internal functional elements to represent the strategy outlined in Section 9.3.1.6.7.2. The logic was implemented using the standard GoldSim™ building blocks, namely mixing cells and function elements, to direct the placement of H3 inventory within the appropriate aquifer flow path or to enable H3 discharge directly into FMB such that the transport and decay computations necessary to calculate stream concentrations and associated doses resulting from the release of H3 could be performed.

Within the Generic CA GoldSim™ model, the aquifer portion of the flowpath associated with each disposal unit is represented by 200 uniform width mixing cells, through which groundwater migrates toward the discharge zone adjacent to the nearest stream, as is shown in Figure 9-14 and Figure 9-15. Plumes of contaminants typically extend from a point directly beneath the surface source to the discharge locality adjacent to nearby streams hence the introduction of tritium in the GoldSim™ model was made to all 200 aquifer mixing cells. Additionally, because insufficient data exist to justify any particular concentration variation along the length of the plume, the inventory was introduced uniformly amongst all 200 cells. For example, the L-Area plume map only delineates regions thought to have tritium concentrations greater then 20 μCi/L. Finally, the introduction of groundwater tritium into the GoldSim™ model was made such that it conformed to the date that the tritium inventory estimate reflected.

Many of the existing tritium plumes at SRS are actually co-mingled plumes from two or more sources located in the general vicinity of one another. This is typically the case for releases from basins, reactor buildings or both. Rather than arbitrarily apportioning the total H3 inventory estimate among the aquifer flowpaths associated with several surface disposal facilities, the shortest aquifer flowpath was selected for introduction of the entire H3 inventory. Examples where this approach was applied are P-Area, L-Area, and K-Area tritium plumes. This approach is expected to introduce a slight bias towards overestimating the tritium concentration in the stream water and also, therefore, the human doses resulting from the use of this stream water.

9.3.1.6.8 Streambed Sediments

Integrator Operable Units are defined as surface water bodies (e.g., streams and lakes) and associated wetlands including sediment/soil, sediment and related biota. Integrator Operable Units have received discharges of radionuclides throughout SRS's operating history as a result of periodic releases from ongoing operations and from historical disposal actions. All sources of residual radioactive material at the SRS end state, including certain closure units, decommissioned facilities and Low-Level Waste (LLW) disposal facilities which contain significant quantities of radioactive constituents will eventually release contaminants to the subsurface and will discharge to these integrating units over the CA period of assessment (2025 to 3025).

Integrator Operable Units in the CA are taken to mean the sediments associated with the main channel of SRS streams and not the watersheds associated with each stream. The seven IOUs at the SRS, which correspond to the surface water drainages, are:

- FMB IOU
- Lower Three Runs (LTR) IOU (includes PAR Pond)
- Pen Branch (PB) IOU
- Savannah River and Floodplain Swamp IOU
- Steel Creek (SC) IOU
- Tims Branch and Steed Pond IOU
- UTR IOU

The locations of each SRS stream and associated watershed are shown in Figure 9-34. In the case of several IOUs, discharge canals are included as a part of the IOU. Examples of this inclusion are C-Reactor Discharge Canal being considered part of the FMB IOU, and the R-Reactor Discharge Canal being considered as part of the LTR IOU. PAR Pond and the other pond located between PAR Pond and R-Reactor are all considered to be a part of the LTR IOU. On the other hand, while Tims Branch and Steed Pond (located on Tims Branch) discharge into UTR, they are not considered to be a part of the UTR IOU, but rather are combined and considered to be a separate IOU.

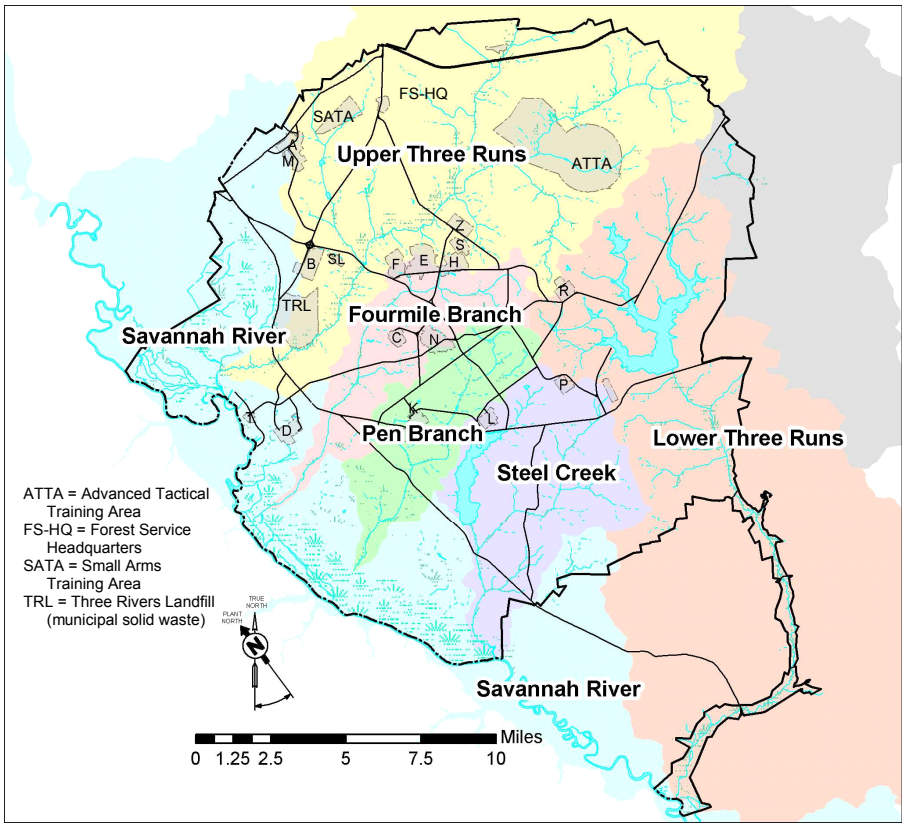


Figure 9-34. Savannah River Site Streams and Associated Watersheds

Significant sources of residual radionuclides expected to be present at the SRS end state are in the sediments associated with the individual SRS site streams. These radionuclides are adsorbed to the streambed sediment and soils of the shallow floodplains that lie immediately adjacent to stream channels. Sediment sampling has been conducted for each of the IOU's by SGCP and the analytical results of these data collection efforts are reported in the Periodic Update reports associated with each IOU. While numerous radionuclides have been released to the site streams at different times since operations began at SRS, most have either decayed away or been flushed out of the system by groundwater and surface flow. The most significant radionuclide in most SRS stream sediment is Cs137 which tends to be adsorbed onto streambed sediments and several uranium isotopes (U238, U235 and U234) which are present in Steed Pond and Tims Branch sediments.

9.3.1.6.8.1 Strategy for Simulating Stream Sediment Sources

The strategy for incorporating the IOUs into the CA Base Case GoldSim™ model embraced the logic illustrated in Figure 9-35. This schematic reflects groundwater moving vertically upward through the streambed sediments and discharging upward into the open water of the stream itself. The water moves vertically upward and discharges into the surface water. The concentration of the radionuclides leached out of the stream sediment is calculated at the POA for each stream just prior to discharging into the Savannah River and then water of that concentration is utilized to evaluate doses received by members of the public. A conceptual model of how this logic is related to an individual stream is illustrated in Figure 9-36.

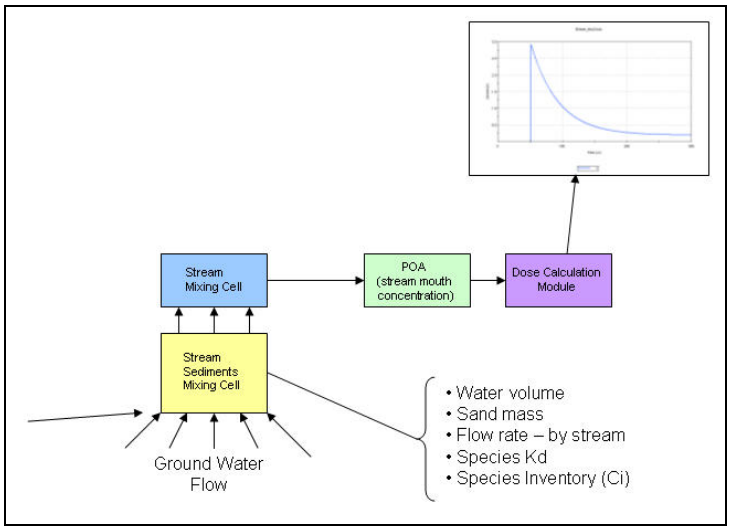


Figure 9-35. Logic for Simulating Streambed Sediment Radionuclide Sources

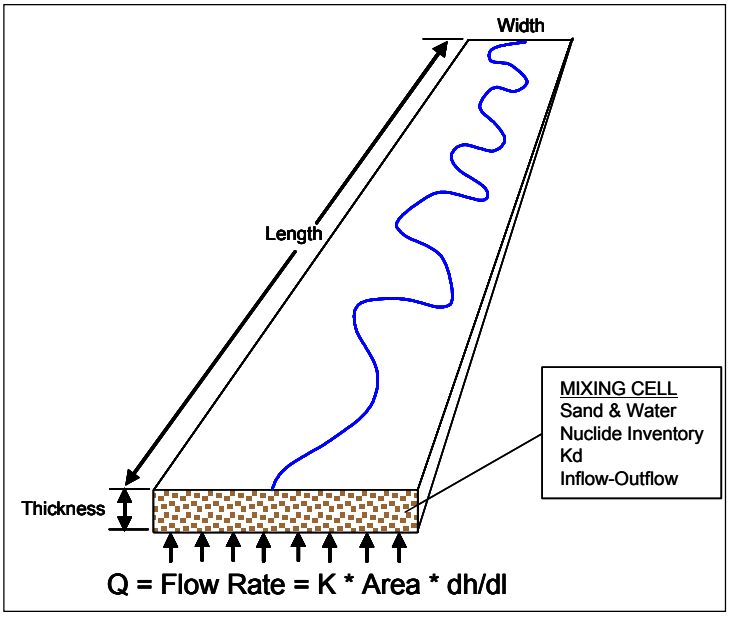


Figure 9-36. Conceptual Model of Flow System in the Vicinity of SRS Streams

The rate of groundwater movement through the stream sediment (Q) is based on typical vertical hydraulic gradients (dh/dl) and vertical hydraulic conductivity (K_v) for the sediment (sand) observed in natural settings like those found along SRS streams as well as an estimate of the cross-sectional area of flow associated with each SRS stream. The cross-sectional area of flow was estimated for each SRS stream as shown in Figure 9-36, where the linear length of contaminated sediments was estimated using the ArcMap program and the appropriate GIS layers needed to delineate the contaminated sediment lengths of SRS streams. A unit stream sediment width was assumed to define the cross-sectional area of flow.

9.3.1.6.8.2 Implementation of Sediment Model

Stream sediments for an entire stream were represented by a single GoldSim™ Cell, which is the basic element in the GoldSim™ program, using the appropriate dimensions, materials, water flux terms and radionuclide inventories. Model input was developed to define those factors unique to each SRS stream which are needed to perform the sediment release calculations and then read into the model. Input items were then directed to the appropriate mixing cell for performing streambed release calculations. The radionuclide flux released from the Streambed Sediment Cell was then connected to the Stream Mixing Cell, established in the CA Base Case model, to perform stream water concentration calculations as well as the associated doses.

9.3.1.6.8.3 Sediment Model Input

Model input related to simulation of streambed sediment releases includes certain items that are needed, in addition to other global parameters (material or transport properties) required by the Base Case GoldSim™ model, to activate the streambed sediment source release. Because sand is the material presumed to be representative of stream sediment, the material and flow properties for saturated sand as well as radionuclide K_ds for sand were used in the sediment model. The vertical hydraulic conductivity (K_v) of sand is 88 m/yr (Phifer et al. 2006) which, along with the vertical hydraulic gradient (dh/dl), were used in computing the groundwater flow rate (Q) through the streambed sediments. This calculation was performed external to the GoldSim™ model and the Q term utilized as input to the GoldSim™ model. Vertical hydraulic gradients ranging from 2.0E-3 to 2.0E-04 were utilized in this calculation and are thought to be reasonable and valid for streambed settings. K_ds for Cs and U isotopes are globally defined within the CA GoldSim™ base case model and are 50 and 200 ml/g, respectively. Table 9-13 contains various parameters that are defined individually for IOUs or sub parts of IOUs.

Table 9-13. Parameters Used in Sediment Flow Calculations

IOU	Length of Contaminated Sediment (m)	Channel Width (m)	Surface Area (m^2)	Flow Rate (Q) (m^3/yr)
UTR	1.5E+04	5	7.7E+04	1.6E+03
FMB	1.6E+04	5	7.9E+04	1.6E+03
Pen Branch	1.2E+04	5	5.7E+04	1.2E+03
Steel Ck	1.6E+04	5	8.0E+04	7.0E+02
LTR	4.5E+04	5	2.3E+05	8.3E+03
Tims Branch	3.5E+03	2	7.0E+02	1.1E+03
Steed Pond	4.8E+02	15	7.2E+02	1.3E+03
SR-A	1.2E+04	950	1.1E+07	2.3E+05
SR-B	1.2E+04	750	9.0E+06	1.8E+05

9.3.2 Dose Module

The overall dose module development process, starting with the screening of transport pathways, exposure scenarios, and radionuclides and ending with the final dose calculations is summarized in Figure 9-37.

As discussed in Section 9.2, the module used to perform CA dose calculations was developed under a subcontract to Neptune and Company, Inc. using the GoldSim™ (GTG 2007) modeling platform. The dose module utilized for the CA is version 1.1. Neptune and Company, Inc. personnel also performed QA checking of the module itself and Savannah River National Laboratory (SRNL) performed the design check of the input data utilized.

The implementation of the dose calculations and associated results are organized within different containers in the Dose Module. The various containers in the Dose Module and their contents are described in the following bullets. Details of the calculations in each container are presented in Sections 9.3.2.2.1 and 9.3.2.2.2.

- The processed input concentrations from the transport model (Species → DoseSpecies) are in InputConcs_DoseSpecies.

- Dose Module inputs are mostly in Behaviors, DoseConversionFactors, and subcontainers within ExposureConcs.

- Exposure concentrations in all derivative media, based on the water input concentrations, are in ExposureConcs.
- The results of the dose assessment are in WaterDose, and a summary of these results is provided in DoseResultsSummary.

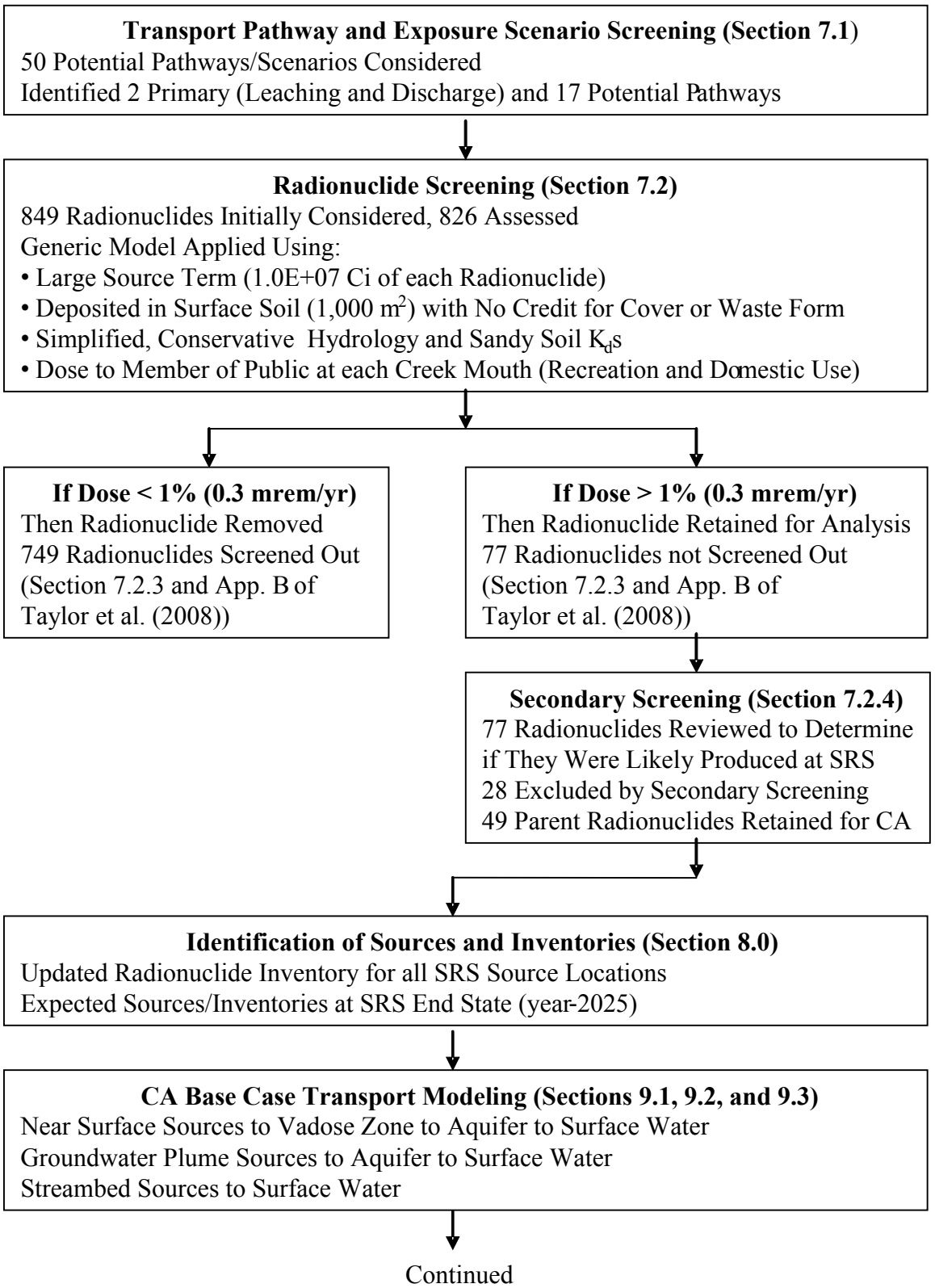


Figure 9-37. Development Process for CA Dose Calculations

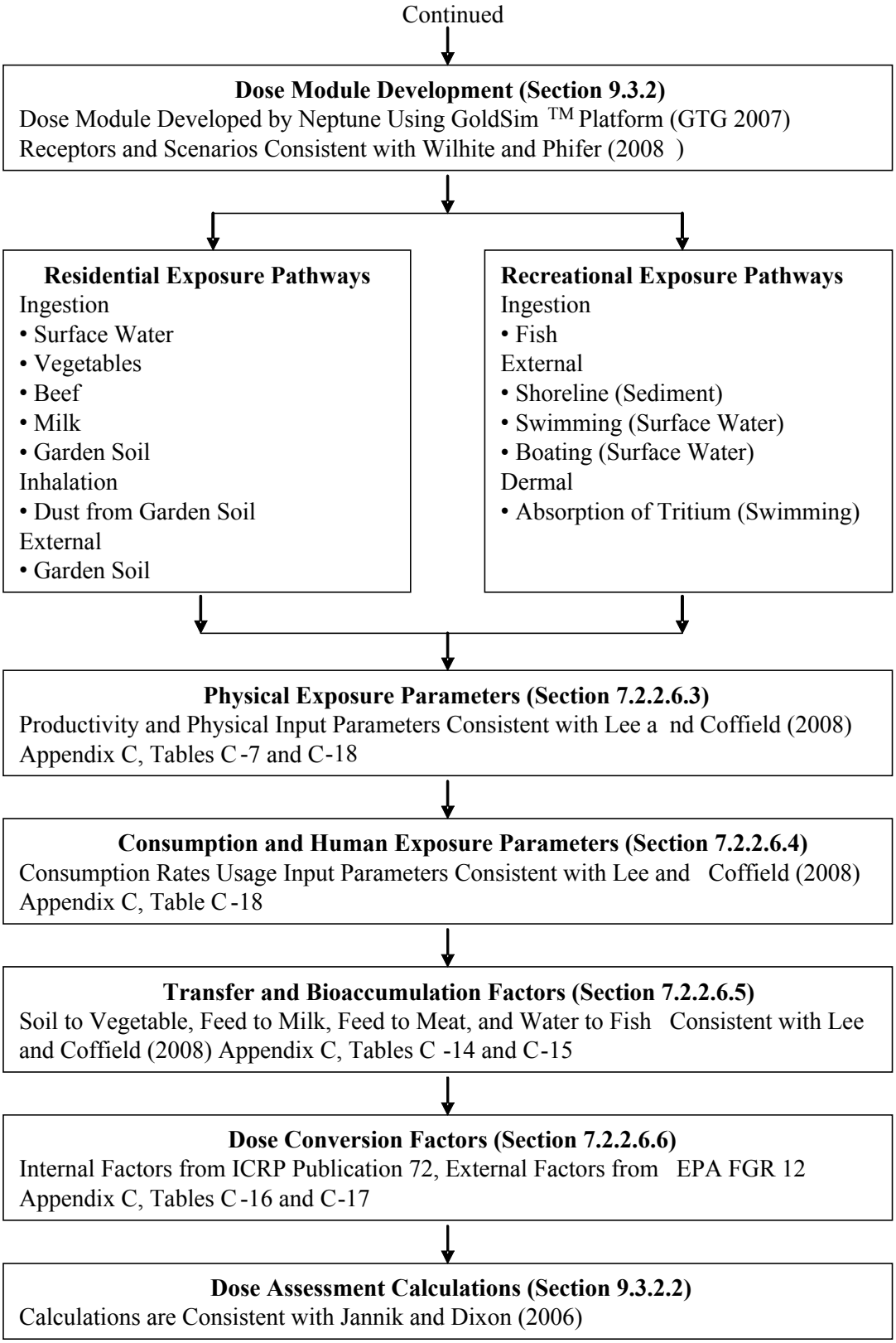


Figure 9-37. Development Process for CA Dose Calculations -- continued

9.3.2.1 Receptors and Scenarios

The CA dose module was used to evaluate dose according to pathways and exposure scenarios identified in Wilhite and Phifer (2008), and summarized in Section 7.1. The two pathways identified of primary importance to radionuclide transport resulting in dose to the public are leaching of the source resulting in contaminated groundwater and transport through the subsurface to surface streams.

Exposure scenarios were developed by considering the potentially significant pathways to human exposure. The initial exposure pathway list was generic in nature, and the significance of each pathway was evaluated on a site-specific and CA-specific basis to develop relevant scenarios. Many exposure pathways were removed from consideration in developing scenarios for the CA because of a negligible contribution to public exposure. The rationale for removing each of these pathways is provided in Section 7.1.2.

Composite Analysis POAs are assumed to be at the mouths of the SRS streams and in the Savannah River (Section 4.2.5), and therefore only exposure scenarios involving contact with, and use of, contaminated surface water (i.e., stream or river water) were considered. Two exposure scenarios were judged to bound exposures; a recreational scenario and a residential scenario (see Section 7.1.3).

Consistent with Wilhite and Phifer (2008), the dose module was used to calculate a residential dose from radionuclide concentrations in the river assuming that this water is used for drinking and farming. The residential dose calculations include the following exposure pathways.

- Direct ingestion of river water
- Consumption of vegetables, beef, and milk produced as a result of using river water for irrigation and watering livestock
- Ingestion of irrigated garden soil
- Inhalation of dust from irrigated garden soil
- External exposure due to soil shine from irrigated garden soil

Also consistent with Wilhite and Phifer (2008), a recreational dose was calculated, and included the following exposure pathways.

- Ingestion of fish
- External exposure as a result of swimming, boating, and shore shine
- Dermal absorption during swimming (tritium only)

The total dose is calculated by summing the recreational and residential doses (Smith et al. 2009a). The exposure pathways included in the Dose Module and the scenarios to which they apply are shown in Table 9-14.

Table 9-14. Exposure Pathways and Scenarios Used in the Dose Module

Exposure Pathway	Exposure Scenario
Ingestion of surface water	Residential
Ingestion of vegetables	Residential
Ingestion of beef	Residential
Ingestion of milk	Residential
External; garden soil	Residential
Ingestion of garden soil	Residential
Inhalation of dust; garden soil	Residential
Ingestion of fish	Recreational
External; shoreline	Recreational
External; swimming	Recreational
Dermal absorption of tritium; swimming[1]	Recreational
External; boating	Recreational

[1]Combined into a total dose from swimming reported in calculation results.

9.3.2.2 Dose Assessment Calculations

The dose assessment calculations for many of the exposure pathways in the Dose Module are an implementation within GoldSim™ of the methodology described in Jannik and Dixon (2006), which is a modification of LADTAP XL© software used at SRS (Simpkins 2004). The equations implemented in these calculations, presented below, are organized according to equations that are relevant to the residential scenario (Section 9.3.2.2.1), and those relevant to the recreational scenario (Section 9.3.2.2.2). Each equation must be evaluated for each radionuclide of interest at each time of interest. The total dose at each time is the sum of the individual radionuclide doses at that time. Descriptions of the exposure parameters used in these equations are provided in Section 7.2.2. Parameter values necessary to conduct these calculations, and specific to this module, are discussed in Section 9.3.4.10 and listed in Tables C-7, and C-14 through C-18. The internal DCFs used in these calculations (Tables C-16 and C-17) are consistent with ICRP Publication 72 values (ICRP 1995), when available. In the absence of ICRP Publication 72 values for several short-lived radionuclides, a value of zero was used, in accordance with DCFs provided in NCRP Report No. 123, Volume I (NCRP 1996). The external DCFs used, and listed in Tables C-16 and C-17, were taken from EPA Federal Guidance Report No. 12 (EPA 1993).

In the equations that follow, L is used to denote units of length, M is used to denote units of mass, and T is used to denote units of time.

9.3.2.2.1 Dose Calculations for the Residential Scenario

9.3.2.2.1.1 Direct Ingestion of River Water

The equation for calculating radiation dose (dose/T) from ingestion of drinking water containing radionuclide i is based on the equation documented in Jannik and Dixon (2006):

$$D_{water\ ingestion,i} = C_{drink_water,i} \times IR_water \times DCF_ing_i$$

where

$D_{water\ ingestion,i}$ = river water ingestion dose for radionuclide i (dose/T)
$C_{drink_water,i}$ = drinking water concentration of radionuclide i (activity/L^3)
IR_water = ingestion rate of drinking water (L^3/T)
DCF_ing_i = dose conversion factor for ingestion of radionuclide i (dose/activity)

There is no holdup time assumed for the river water before ingestion.

9.3.2.2.1.2 Ingestion of Vegetables

The equation for vegetable ingestion radiation dose (dose/T) is:

$$D_{vegetable\ ingestion,i} = C_{veg,i} \times IR_veg \times F_{veg} \times DCF_ing_i\ ,$$

where

$D_{vegetable\ ingestion.i}$ = vegetable ingestion dose for radionuclide i (dose/T)
$C_{veg,i}$ = concentration of radionuclide i in vegetables (activity/M)
IR_veg = ingestion rate of vegetables (M/T)
F_{veg} = fraction of vegetables grown using contaminated agricultural water (–)
DCF_ing_i = dose conversion factor for ingestion of radionuclide i (dose/activity)

Both root and leaf uptake contributed to the radionuclide concentration in vegetables ($C_{veg,i}$). For vegetable contamination via root uptake, the concentration is calculated from:

$C_{root\ uptake,i}$ = plant concentration by root uptake of radionuclide i (activity/M) = $C_{soil,i}$ (activity/M) × B_{iv} ,

where

$C_{soil,i}$ = concentration of radionuclide i in soil (activity/L^3)
B_{iv} = plant-soil concentration ratio for radionuclide *i* (–)

The concentration in soil ($C_{soil,i}$) was calculated assuming equilibrium partitioning of radionuclides from agricultural water to bulk soil, based on the retardation factor R. The retardation factor equation is:

$$R_i = \frac{K_{d,i} \times \rho_{soil}}{\theta_w},$$

where

R_i	=	retardation factor for radionuclide i (–)
ρ_{soil}	=	soil bulk density (M/L^3)
$K_{d,i}$	=	soil/water distribution coefficient of radionuclide i (L^3/M)
θ_w	=	volumetric water content of the soil (–)

The volumetric soil water content equation is:

$$\theta_w = \eta \times S,$$

where

θ_w	=	volumetric water content of the soil (–)
η	=	soil porosity (–)
S	=	soil saturation (–)

This simple model is applied based on the assumption that 100% of water added to the soil during the growing season is irrigation water. Enough water is presumed to be added to reach the K_d equilibrium, and there is no accounting of dilution by precipitation. The concentration in soil is calculated as:

$$C_{soil,i} = \frac{C_{water,i} \times R_i}{\rho_{water}},$$

where

$C_{soil,i}$	=	activity of radionuclide i in agricultural soil (activity/M)
$C_{water,i}$	=	activity of radionuclide i in agricultural water (activity/L^3)
ρ_{water}	=	water density, assumed to be 1 g/mL (M/L^3)

The concentrations of radionuclides in vegetables related to uptake from the leaves are calculated following Jannik and Dixon (2006), as follows:

$$leaf\ uptake_i = \frac{r_i \times F_w \times \left(1 - e^{-\lambda_i t_{irr}}\right)}{Y\lambda_e},$$

where

r_i	=	retention of radionuclide i on plant leaf surface (–)
F_w	=	washing factor; differs for leafy and nonleafy plants (–)
λ_i	=	decay constant for radionuclide i (T^{-1})
t_{irr}	=	irrigation time (T)
Y	=	vegetable production yield (M/L^2)
λ_e	=	weathering and radiological decay constant (T^{-1})

Leaf uptake has dimensions of $L^2 \cdot T/M$.

The equation to calculate vegetable concentrations as a function of both leaf and root uptake is:

$$C_{veg,i} = \left[\left(C_{ag_water,i} \times Irr \times \ leaf_uptake_i\right) + C_{root_uptake,i}\right] \times e^{-\lambda_i \cdot t_hold_veg},$$

where

$C_{veg.i}$	=	concentration of radionuclide i in garden vegetables (activity/M)
$C_{ag_water.i}$	=	concentration of radionuclide i in agricultural water (activity/L^3)
Irr	=	irrigation rate ($L^3/L^2 \cdot T$)
$leaf_uptake_i$	=	plant uptake by leaves of radionuclide i ($L^2 \cdot T/M$)
$C_{root_uptake,i}$	=	concentration of radionuclide i in plant from root uptake (activity/M)
λ_i	=	decay constant for radionuclide i (T^{-1})
t_hold_veg	=	holdup time between harvest and ingestion of vegetables (T)

9.3.2.2.1.3 Ingestion of Beef

Beef cattle can uptake radionuclides in the residential scenario by directly ingesting river water or ingesting fodder that has been irrigated with river water. The equation used for calculating beef ingestion radiation dose rate (similar to Jannik and Dixon 2006) is:

$$D_{beef\ ingestion,i} = C_{beef,i} \times IR_beef \times F_{beef} \times DCF_ing_i \ ,$$

where

$D_{beef\ ingestion,i}$	=	beef ingestion dose for radionuclide i (dose/T)
$C_{beef,i}$	=	concentration of radionuclide i in beef (activity/M)
IR_beef	=	ingestion rate of beef (M/T)
F_{beef}	=	fraction of beef from cattle raised on affected pasture (–)
DCF_ing_i	=	dose conversion factor for ingestion of radionuclide i (dose/activity)

The concentration of radionuclides in beef is calculated as:

$$C_{beef,i} = TF_{beef,i} \times \left[\left(C_{fodder,i} \times IR_{fodder} \times F_{pasture}\right) + \left(C_{ag_water,i} \times IR_{water} \times F_{water}\right)\right] \times e^{-\lambda_i . t_hold_beef} ,$$

where

$C_{beef,i}$	=	concentration of radionuclide i in beef (activity/M)
$TF_{beef,i}$	=	beef transfer factor for radionuclide i (T/M)
IR_{fodder}	=	cattle ingestion rate of fodder (M/T)
$F_{pasture}$	=	fraction of pasturage irrigated with agricultural water (–)
$C_{ag_water,i}$	=	concentration of radionuclide i in agricultural water (activity/L^3)
IR_{water}	=	cattle ingestion rate of agricultural water (L^3/T)
F_{water}	=	fraction of water for cattle from agricultural water (–)
λ_i	=	decay constant for radionuclide i (T^{-1})
t_hold_beef	=	holdup time between slaughter and ingestion of beef (T)

The equation used for calculating concentrations in animal fodder is very similar to that used for calculating concentrations in vegetables (Section 9.3.2.2.1.2). The only difference is the assumption that all animal exposure is via grazing and, therefore, the exponential decay for holdup time for vegetables is omitted. The equation is:

$$C_{fodder,i} = \left(C_{ag_water,i} \times Irr \times leaf_uptake_i\right) + C_{root_uptake,i} ,$$

9.3.2.2.1.4 Ingestion of Milk

Like beef cattle, dairy cows can also uptake radionuclides in the residential scenario by directly ingesting river water or ingesting fodder that has been irrigated with river water. The equation used for calculating the milk ingestion radiation dose (similar to Jannik and Dixon 2006) is:

$$D_{milk\ ingestion,,i} = C_{milk,i} \times IR_milk \times F_{milk} \times DCF_ing_i ,$$

where

$D_{milk\ ingestion,i}$	=	milk ingestion dose, (dose/T)
$C_{milk,i}$	=	concentration of radionuclide i in beef (activity/ L^3)
IR_milk	=	ingestion rate of milk (L^3/T)
F_{milk}	=	fraction of milk from cows raised on affected pasture (–)
DCF_ing_i	=	dose conversion factor for ingestion of radionuclide i (dose/activity)

The equation for concentrations of radionuclides in milk is very similar to that for beef (after Jannik and Dixon):

$$C_{milk,i} = TF_{milk,i} \times \left[\left(C_{fodder,i} \times IR_{fodder} \times F_{pasture}\right) + \left(C_{ag_water,i} \times IR_{water} \times F_{water}\right)\right] \times e^{-\lambda_i . t_hold_milk} ,$$

where

$C_{milk,i}$	=	concentration of radionuclide i in milk (activity/M)
$TF_{milk,i}$	=	milk transfer factor for radionuclide i (T/M)
$C_{fodder,i}$	=	concentration of radionuclide i in fodder (activity/M)
IR_{fodder}	=	milk cow ingestion rate of fodder (M/T)
$F_{pasture}$	=	fraction of pasturage irrigated with agricultural water (–)
$C_{ag_water,i}$	=	concentration of radionuclide i in agricultural water (activity/L^3)
IR_{water}	=	milk cow ingestion rate of agricultural water (L^3/T)
F_{water}	=	fraction of water for milk cows from agricultural water (–)
λ_i	=	decay constant for radionuclide i (T^{-1})
t_hold_milk	=	holdup time between milking and ingestion of milk (T)

9.3.2.2.1.5 Soil Ingestion

The radiation dose equation for soil ingestion was based on examination of a similar equation for sediment, documented in Jannik and Dixon (2006):

$$D_{soil\ ingestion,i} = C_{soil,i} \times IR_soil \times T_garden \times DCF_ing_i ,$$

where

$D_{soil\ ingestion,i}$	=	soil ingestion dose for radionuclide i (dose/T)
$C_{soil,i}$	=	concentration of radionuclide i in soil (activity/M)
IR_soil	=	mass ingestion rate of soil (M/T)
T_garden	=	fraction of time spent in the garden / pasture (–)
DCF_ing_i	=	dose conversion factor for ingestion of radionuclide i (dose/activity)

9.3.2.2.1.6 External Irradiation from Soil

The radiation dose equation for external irradiation from soil was based on examination of a similar equation for sediment, documented in Jannik and Dixon (2006):

$$D_{external\ soil,i} = C_{soil,i} \times \rho_{soil} \times T_garden \times DCF_ext_i ,$$

where

$D_{external\ soil,i}$ = external irradiation dose from radionuclide i in soil (dose/T)
$C_{soil,i}$ = concentration of radionuclide i in soil (activity/M)
ρ_{soil} = soil bulk density (M/L^3)
T_garden = fraction of time spent in the garden / pasture (–)
DCF_ext_i = dose conversion factor of radionuclide i for external irradiation (dose/T per activity/L^3)

9.3.2.2.1.7 Inhalation of Dust

The equation for dust inhalation has a similar form to that for ingestion, but employs an inhalation DCF and a factor that describes the mass of fine soil particulates suspended in breathing zone air:

$$D_{dust\ inhalation,i} = C_{soil,i} \times dust_loading \times InhR \times T_garden \times DCF_inh_i$$

where

$D_{dust\ inhalation,i}$ = dose inhalation dose for radionuclide i (dose/T)
$C_{soil,i}$ = concentration of radionuclide i in soil (activity/M)
$dust_loading$ = mass of suspended soil per unit volume of ambient air (M/L^3)
$InhR$ = inhalation rate (L^3/T)
T_garden = fraction of time spent in the garden / pasture (–)
DCF_inh_i = dose conversion factor of radionuclide i for inhalation (dose/activity)

9.3.2.2.2 <u>Dose Calculations Related to Recreational Surface Water</u>

Similar to some of the residential exposure pathways, refinements to some of the exposure calculations described in Jannik and Dixon (2006) were also introduced for recreational exposure pathways. A simple soil/water equilibrium partitioning calculation identical to that described for soil in Section 9.3.2.2.1.2 was used to calculate sediment concentrations. External radiation dose related to sediment exposure is calculated using a 15-cm depth DCF that is used for calculating soil external dose.

9.3.2.2.2.1 Ingestion of Fish

The radiation dose from ingestion of fish is calculated from radionuclide concentrations in recreational water as:

$$D_{fish\ ingestion,i} = C_{rec_water,i} \times B_{if} \times IR_fish \times DCF_ing_i \times e^{-\lambda_i \cdot t_hold_fish} ,$$

where

$D_{fish\ ingestion,i}$ = fish ingestion dose for radionuclide i (dose/T)
$C_{rec_water,i}$ = concentration of radionuclide i in recreational water (activity/L^3)
B_{if} = fish-water concentration ratio for radionuclide i (–)
IR_fish = ingestion rate of fish (M/T)
DCF_ing_i = dose conversion factor for radionuclide i for ingestion (dose/activity)
λ_i = decay coefficient for radionuclide i (T^{-1})
t_hold_fish = holdup time between catching and eating fish (T)

9.3.2.2.2.2 External Irradiation from Sediment

The equation for sediment dose from external irradiation used is very similar to that described for soil. Sediment concentrations are calculated from concentrations in recreational water using the same retardation and equilibrium partitioning equations described for soil in Section 9.3.2.2.1.2. The only difference is that sediment and recreational water take the place of garden soil and agricultural water. These equations portray a worst-case condition in so far as the partitioning does not account for the dynamic nature of sediments that will result in some dilution of affected sediments over time.

The external radiation dose from sediment exposure (dose/T) is calculated from radionuclide concentrations in sediment as:

$$D_{external\ sediment,i} = C_{sediment,i} \times sed_dens \times T_shore \times F_{shape} \times DCF_ext_i ,$$

where

$D_{external\ sediment,\ i}$ = external irradiation dose from sediment for radionuclide i (dose/T)
$C_{sediment,i}$ = concentration of radionuclide i in sediment (activity/M)
sed_dens = sediment dry bulk density (M/L^3)
T_shore = fraction of time spent on the shoreline (–)
F_{shape} = shape factor for external dose from shoreline deposits (–)
DCF_ext_i = dose conversion factor for radionuclide i for external irradiation (dose/T per activity/L^3)

9.3.2.2.2.3 External Exposure While Boating

The external radiation dose received during boating is calculated assuming 2π geometry with no shielding considerations (Jannik and Dixon 2006). This is accomplished by applying a geometry factor of 0.5 and using a submersion dose conversion factor. Shielding provided by the boat's hull is not considered. The external dose from boating in recreational water is calculated from:

$$D_{external\ boating,i} = C_{rec_water,i} \times T_boating \times F_{geo,boat} \times DCF_sub_i$$

where

$D_{external\ boating,i}$ = external irradiation dose from radionuclide i while boating (dose/T)
$C_{rec_water,i}$ = concentration of radionuclide i in recreational water (activity/L^3)
$T_boating$ = fraction of time spent boating (–)
$F_{geo,\ boat}$ = geometry shape factor for external dose during boating (–)
DCF_sub_i = dose conversion factor for radionuclide i for external irradiation when submerged in the exposure medium (dose/T per activity/L^3)

9.3.2.2.2.4 Exposure While Swimming

The external radiation dose from swimming assumes spatially-uniform radionuclide concentrations and complete submersion (4π geometry) (Jannik and Dixon 2006) and is calculated from:

$$D_{external\ swimming,i} = C_{rec_water,i} \times T_swimming \times F_{geo,swim} \times DCF_sub_i$$

where

$D_{external\ swimming,I}$ = external swimming dose from radionuclide i (dose/T)
$C_{rec_water,i}$ = concentration of radionuclide i in recreational water (activity/L^3)
$T_swimming$ = fraction of time spent swimming (–)
$F_{geo,\ swim}$ = geometry shape factor for external dose during swimming (–)

The radiation dose from tritium absorption while swimming is calculated from the tritium concentration in recreational water as:

$$D_{tritium\ swimming} = C_{rec_water,i} \times T_swimming \times F_{absorption} \times DCF_ing_i$$

where

$D_{tritium\ swimming}$ =dose from tritium absorption while swimming (dose/T)
$C_{rec_water,i}$ = concentration of radionuclide i in recreational water (activity/L^3)
$F_{absorption}$ = tritium absorption rate during swimming (L^3/T)
DCF_ing_i = dose conversion factor for radionuclide i for ingestion (dose/activity)

The external swimming dose and the tritium swimming dose are added together to give a total swimming dose that is reported as a recreational dose in the results from model calculations.

9.3.3 Model Input and Output

GoldSim™ container External_IO is used to read the input data required for model simulations and write results from the model calculations to output files. The CA GoldSim™ model was made very flexible by separating the input data from the model to a large extent. Model input parameters are contained in three files:

1. Access database CA_db.mdb is used to input parameters for stochastic distributions used in the model. The Access database is a simple GoldSim™ database and uses the tbl_Parameter table to hold the distribution input. This structure was used because it provides an easily readable format to list the distribution parameters and is easier to control and maintain than constants embedded within the model. Values stored in the database must be downloaded into the model. When a value in the database is changed, the download must be performed to update the model. GoldSim™ provides a built-in function that will automatically update all entries from the database which the user can run to be certain that the latest values in the database are stored in the model.
2. Excel workbook Parameters.xls is used to input molecular weights and half lives of radionuclides into the CA model and deterministic values for K_ds, solubilities, and CDP factors for elements in the model. K_d values are input for sandy and clayey soil and for young, middle aged, and old oxidizing cement. Solubilities in the waste material and in cement are input. The spreadsheet is automatically read when the GoldSim™ simulation starts. The advantage of using the spreadsheet is the ability to clearly label the input. As with the Access database, the intent of collecting fixed model input parameters into a spreadsheet is easier for control and maintenance of the data than is possible when the values are embedded within the model.
3. Excel workbook Disposal_Units.xls is used to input parameters for individual sources. A partial listing of an input file for one source is shown in Figure 9-38. where the first 30 lines of input are displayed. Input entries in columns A, B and C are explained in Table 9-15. Columns E and F are used to specify clay in the vadose zone. Entering 1 in column F sets the properties of the corresponding vadose zone cell to those of clayey soil while an entry of 0 sets the properties to sandy soil. A fraction between 0 and 1 will cause the property values to be calculated as a mixture of clayey and sandy soil where the fraction represents the volume fraction of clayey soil in the cell. However, this feature of using mixed soil properties was never used in the CA calculations. Columns H and I extend 63 lines to include an inventory entry for all species in the model. Columns K and L extend 100 lines to enter the fractional distribution of the output from the vadose zone to the first 100 aquifer cells. Columns N and O extend 40 lines to allow specification of clay in the 40 aquitard cells in a manner identical to that used for the vadose zone in columns E and F. Data from the spreadsheet is automatically read into the model when the GoldSim™ simulation starts. Multiple spreadsheets for individual sources are stored in a single Disposal_Units.xls workbook.

	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O
1	***Disposal Unit Parameters***				***Unsaturated Zone***			***Inventory***			***Footprint Distribution***			***Aquitard Composition***	
2	***Input***	***Value***	***Units***		***Cell***	***Clay***		***Species ID***	***Curies***		***Cell***	***Fraction***		***Cell***	***Clay***
3	***Inventory Year***	1962			1	0		Ac227	0		1	0.0002395		1	1
4	***Cap Placement Year***	2007			2	0		Ag108m	0		2	0.0007281		2	0
5	***Cap Time to Failure***	20000	yr		3	0		Al26	0		3	0.0012168		3	0
6	***Cap Type***	2			4	0		Am241	2.20E+02		4	0.0017055		4	0
7	***Waste Thickness***	20	ft		5	0		Am242m	0		5	0.0021942		5	0
8	***Fraction Sand in Waste***	1			6	0		Am243	0		6	0.0026829		6	0
9	***Barrier Thickness***	1	ft		7	0		Bi210m	0		7	0.0031715		7	0
10	***Volume Fraction Soil in Barrier***	1			8	0		C14	4.00E+03		8	0.0036602		8	0
11	***Vadose Zone Thickness***	33	ft		9	0		Ca41	0		9	0.0040626		9	0
12	***Aquifer Total Path Length***	4682.592	ft		10	0		Cf249	0		10	0.0046276		10	0
13	***Length of First Aquifer Zone***	2289.852	ft		11	0		Cf251	0		11	0.0051164		11	0
14	***Length of Aquitard Zone***	1087.524	ft		12	0		Cl36	0		12	0.0056051		12	0
15	***Aquifer Pore Velocity***	132.072	ft/yr		13	0		Cm243	0		13	0.0060938		13	0
16	***Aquitard Pore Velocity***	14.786	ft/yr		14	0		Cm244	8.13E+03		14	0.0065825		14	0
17	***Stream ID***	3			15	0		Cm245	8.41E-04		15	0.0070711		15	0
18	***Release Mechanism***	1			16	0		Cm246	8.41E-04		16	0.0075597		16	0
19	***Source Release Rate***	0	1/yr		17	0		Cm247	0		17	0.0080484		17	0
20	***CDP Activation Switch***	1			18	0		Cm248	0		18	0.0083649		18	0
21	***Use PA Flux***	0			19	0		Co60	0		19	0.009016		19	0
22	***Stream Bed Water Volume***	1	m^3		20	0		Cs135	0		20	0.0095047		20	0
23	***Stream Bed Flow***	1	m^3/yr					Cs137	7.81E+04		21	0.0099934		21	0
24	***Source Area***	1	m^2					Eu152	0		22	0.0104821		22	0
25								Eu154	0		23	0.0109707		23	0
26								Eu155	0		24	0.0114594		24	0
27								Gd152	0		25	0.0119481		25	0
28								H3	0		26	0.0121882		26	0
29								I129	1.10E-01		27	0.0129156		27	0
30								K40	1.19E-01		28	0.0134043		28	0
											:	:		:	:
											100	0		40	0

Figure 9-38. Partial Listing of Input File for ORWBG.

Table 9-15. Source Parameters in Column B of Input File

Row	Parameter	Description
3	Inventory Year	Year when the waste inventory is placed and material transport begins.
4	Cap Placement Year	Year when engineered cap is placed over the waste and reduced infiltration begins.
5	Cap Time to Failure	Number of years the cap remains intact. Used only for metal caps that, due to corrosion, will suddenly lose integrity.
6	Cap Type	1 – Clay cap 2 – Geosynthetic cap 3 – Concrete cap 4 – Metal cap 0 or > 4 – No cap (natural infiltration)
7	Waste Thickness	Waste layer thickness in feet.
8	Fraction Sand in Waste	The volume fraction of sandy soil in the waste cells. One minus this number is the volume fraction of clayey soil in the cells.
9	Barrier Thickness	Thickness of barrier layer below waste in feet.
10	Volume Fraction Soil in Barrier	The volume fraction of soil in the barrier cells. One minus this number is the volume fraction of concrete in the cells.
11	Vadose Zone Thickness	Total 20 cell vadose zone thickness in feet.
12	Aquifer Total Path Length	Total 200 cell aquifer path length in feet.
13	Length of First Aquifer Zone	Length of first region with 100 cells of saturated sandy soil within aquifer.
14	Length of Aquitard Zone	Length of 40 cell region in the second set of 100 aquifer cells where the clay fractions specified in column O are entered.
15	Aquifer Pore Velocity	Pore velocity in ft/yr in the aquifer cells.
16	Aquitard Pore Velocity	Pore velocity in ft/yr in the aquitard cells.
17	Stream ID	1 – UTR 2 – LTR 3 – FMB 4 – SC/PB 5 – SR
18	Release Mechanism	Identification number from Table 9-6.
19	Source Release Rate	Constant fractional release rate, used with release mechanism 2.
20	CDP Activation Switch	Enter 1 to apply CDP factors to the Kds else enter 0 to not use CDP factors.
21	Use PA Flux	An entry other than 0 specifies use of a time series table of fluxes at the water table. The number entered indicates the spreadsheet read in Excel workbook PA_Flux.xls where the first worksheet ("ZeroFlux") is number 0.
22	Stream Bed Water Volume	Water volume in stream bed in m^3.
23	Steam Bed Flow	Water flow from stream bed into stream in m^3/yr.
24	Source Area	Source area in square meters. The actual source area is only required for solubility controlled releases (mechanism 0), otherwise a default value of 1 m^2 is used.

See Figure 9-14 for model geometry.

9.3.3.1 Output Collection

Output from a GoldSim™ simulation for a single source is written to four Excel workbooks:

1. WT_Flux.xls stores isotopic fluxes at the water table in pCi/yr.
2. Stream_Flux.xls stores isotopic fluxes at the stream in pCi/yr.
3. Dose_Species.xls stores the total dose from each CA isotope in mrem/yr.
4. Dose_Pathway.xls stores doses from individual pathways, and summed residential, recreational and total doses in mrem/yr.

A separate worksheet is created in the workbooks for each disposal unit run during the simulation. The frequency with which data is written to the output files is controlled by the set of functions in top-level container Simulations_Settings shown in Figure 9-39. As the text in the GoldSim™ model block states, the simulation time is divided into three segments and different print frequencies are specified for each segment. The CA model is currently hardwired to output results every year for the first 100 years of simulation, every five years for the next 1,000 years and every 50 years thereafter. The intent was to capture peaks from mobile species that occur early in the simulation with more detail and to print less frequently as the simulation progresses to save space and to reduce plot file sizes. All of the results and plots presented in this section are based on this reporting frequency.

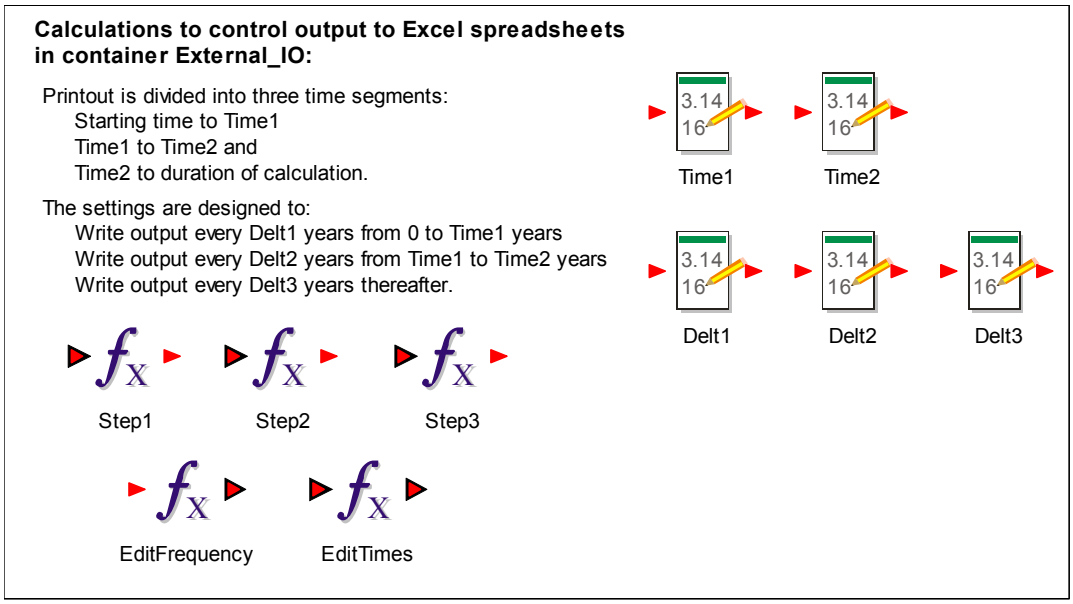


Figure 9-39. Output Control Functions Embedded within GoldSim™ CA Model

9.3.3.2 Simulation Automation

Methods using VBA macros in Excel workbooks were developed to automatically run through a series of GoldSim™ simulations for any number of sources. The macros are primarily used to automatically set up the required input and output files from a list of sources that will be run. Another special macro automatically runs the requested sequence of GoldSim™ simulations. Details of the automation methods are provided in Appendix A of *Savannah River Site Composite Analysis: Base Case Deterministic Calculations* (Smith et al. 2009a).

9.3.4 Input Data

As indicated by Figure 9-38, Table 9-15, and Section 9.3.2, a significant amount of data must be assembled to perform a simulation for each CA source. The following subsections describe the input data used for the CA calculations.

9.3.4.1 Inventory

CA source inventories were based on those provided in *Inventory of Residual Radioactive Material at the Projected Savannah River Site End State* (Hiergesell et al. 2008). The inventory values used in the CA modeling are provided in Appendix A. Some modifications to the Hiergesell et al. (2008) inventory were made, most of which are documented in the CA Section 8.0 and Appendix A. The following are the modifications made that were not documented in Section 8.0 or Appendix A:

- Tritium inventory for the following units was accounted for separately by the groundwater plumes associated with these facilities and was not redundantly added as a source:
    1. E-Area MWMF/Low Level Radioactive Waste Disposal Facility (LLRWDF) flow to Fourmile Branch
    2. F-Area Inactive Process Sewer Line flow to FMB
    3. H-Area Inactive Process Sewer Line flow to FMB
    4. E-Area ORWBG
    5. F-Area Seepage Basin (FSB)
    6. H-Area Seepage Basin (HSB)
- Inventory dates provided in CA Section 8.0 and Appendix A which are associated with operating facilities were set to the presumed site end-state date of 2025. This was applied to the following 12 sources: 773-A, 776-A, 778-A, 772-F, 772-1F/4F, H-Canyon, H-Area Sand Filter, HANM, HAOM (including Vault 217-H), 234-7H, 236-H and 237-H.
- Inventory dates associated with the ORWBG, MWMF and LLRWDF were set to the mid points of the operating times to better represent the actual disposal time.

- In some cases, the Hiergesell et al. (2008) report provided estimated inventories in 2025 by decaying the inventory from an earlier date. In these cases, the CA base case calculations always used the earliest inventory provided and performed decay chain calculations internally.

- The tritium inventory associated with the MWMF GOU noted in Table 9-12 (also Table A-77) was not modeled as an established plume. Rather, as discussed in Section 9.3.1.6.7.2, the tritium inventory listed in Table A-23 of Appendix A for the MWMF (and LLRWDF) was modeled as a surface release.

Table C-1 of Appendix C provides a cross-reference between the 152 identified CA sources (see Section 9.2) and the tables in Appendix A where the radionuclide inventory data were obtained. The abbreviated name assigned to each source is also listed in Table C-1. The inventory for some sources was split between source release mechanisms, aquifer flow paths, or streams of influence. The manner in which the inventory was partitioned for these cases is documented in Table C-2. Table C-2 also lists the manner in which the inventories for some sources (e.g., individual tanks in FTF and HTF facilities) were grouped for modeling purposes. The inventory data are placed in columns H (species) and I (curies) and the inventory year is placed in cell B3 of the input file as shown in Figure 9-38.

9.3.4.2 Waste Zone

The total thickness of the waste zone for sources requiring specific source release modeling (Section 9.3.1), given in Table C-3, is entered in cell B7 of the Disposal_Unit.xls spreadsheets. The total thickness is evenly distributed among the five mixing cells used to represent the source material. The composition of the waste material is calculated as a mixture of sandy soil and clayey soil properties with the volume fraction of sandy soil entered in cell B8. This gives the user some limited flexibility in specifying waste properties. The waste zone only impacts K_d-, solubility-, or rate-controlled source release modeling. For most of the CA calculations transport through the waste zone was neglected and the zone properties set to those for one foot of sandy soil.

9.3.4.3 Vadose Zone and Barrier

Vadose zone data was extracted from *Savannah River Site Vadose Zone Data for Composite Analysis Calculations* (Noffsinger et al. 2009, Table 1), and is listed in Table C-4. The vadose zone lithology was obtained by looking at cross sections, field geologic logs, or geophysical logs in SGCP reports. The source layers and distinct lithological layers used for contaminant migration analysis calculations performed for Resource Conservation and Recovery Act (RCRA) Facility Investigation/Remedial Investigation (RFI/RI) documents were retained for the CA analysis. The lithology descriptors were limited to either 'sand' or 'clay', to be consistent with the report, *Geochemical Data Package for Performance Assessment Calculations Related to the Savannah River Site* (Kaplan 2006). In this report, the 'Sandy Sediment Layer', or sand, is <25 wt-% silt + clay, and the 'Clay Sediment Layer', or clay, is 25 to 45 wt-% silt + clay (Noffsinger et al. 2009).

The total length of the vadose zone is entered in cell B11 of the Disposal_Units.xls spreadsheets (see Figure 9-38). The length of the clay layer in the vadose zone was used to calculate the portion of the 20 cells used to represent the vadose zone (unsaturated zone) occupied by clay. The composition of the 20 unsaturated zone cells is set by the entries in column F of the input file as shown in Figure 9-38 where an entry of one identifies the cell as a clay cell. In practice, these values were entered into a spreadsheet that was in turn read by the input automation code described in Appendix A of Smith et al. (2009a) to create the Disposal_Units.xls worksheets.

In some cases, where source release modeling was required, the vadose zone data report also provided the thickness of the concrete base of the facility. This transport zone was modeled using the three Barrier cells shown in Figure 9-15. As shown in Figure 9-38, barrier input includes the thickness (cell B9) and volume fraction soil (cell B10). Flow through the concrete base was determined by the infiltration rate so, in most cases, the impact of the concrete barrier was an increase in K_d in the concrete and a longer transport time for the radionuclides.

9.3.4.4 Aquifer

A significant amount of input data was obtained from the aquifer flow analysis (Hamm et al. 2009). This analysis involved using existing models (3-dimensional, or 3D) of groundwater flow within the SRS boundary to determine the path contaminants would follow from each of the CA sources to stream outcrops. Section 9.2.1 provides information on each of the existing models used in the aquifer flow analysis. The method used involved running multiple streamlines originating over the surface area occupied by each source and following the path through the groundwater to the stream. This analysis provided the following data:

- **Average one-dimensional (1D) flow path length through the aquifer** (input in cell B12 as shown in Figure 9-38). For deterministic CA calculations, the average aquifer path length was used as the total length of the 200 aquifer cells shown in Figure 9-15. In some cases, the flow path distribution showed a distinct bimodal distribution of both long and short path lengths. In this event, the source was divided between the two average path lengths to more accurately represent the actual transport. The total inventory for the source was divided between the flow paths by estimating the fraction of the source draining to each path.

- **Average 1D flow path length through clay within an aquitard.** The flow path length through aquitard clay was used to determine the cell length in the 40 cell aquitard region shown in Figure 9-15. When no clay was present in the flow path, the 40 "aquitard" cells were given the properties of saturated sandy soil. When aquitard clay was present, the model did not require that all 40 cells in the "aquitard" region be assigned clay properties. This provided some flexibility in the model to keep aquifer and aquitard cell lengths as uniform as possible within the constraints of the fixed aquifer cell configuration illustrated in Figure 9-15. The composition of the 40 "aquitard" cells is set by the entries in Column O of the input file as shown in Figure 9-38, where an entry of 1 identifies the cell as a clay cell. The total length of the 40 "aquitard" cells is input in cell B14 as shown in Figure 9-38.
- **Average 1D pore velocities through the aquifer and aquitard regions**. The aquifer and aquitard pore velocities are entered in cells B15 and B16, respectively as shown in Figure 9-38.
- **Identification of the stream or streams that the contaminant is transported to**. In some cases, flow divides were identified by the aquifer analysis and the source inventory was transported to outfalls for two different streams. In this event, the source was divided between the two flow paths. The total inventory for the source was divided between the flow paths by estimating the fraction of the source draining to each path. Table 9-15 lists stream identification numbers which are input as appropriate into cell B17 as shown in Figure 9-38.
- **A distribution of the flow from the vadose zone over the first 100-cell aquifer region for each source**. The source area, which may have an irregular shape, was collapsed into a one-dimensional distribution of area over the first 100 aquifer cells. The inventory is assumed to be uniformly distributed throughout the source area. Therefore, flow from the vadose zone is distributed into the aquifer by the fraction of the source area over each aquifer cell. This process is illustrated in Figure 9-40, where the flow distribution is represented by the blue histogram. The footprint distribution is entered in Column L of the Disposal_Unit.xls data sheet as shown in Figure 9-38. Not all 100 cells must receive flow from the vadose zone (i.e., the fraction may be zero).

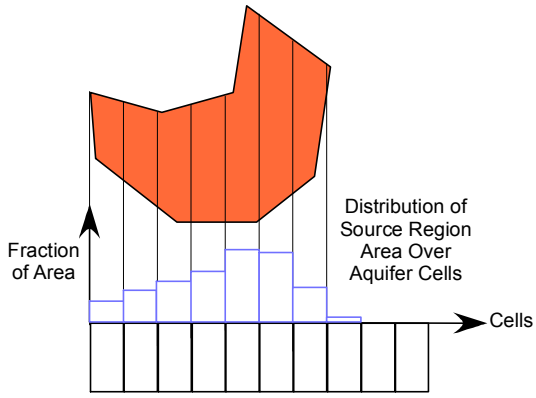
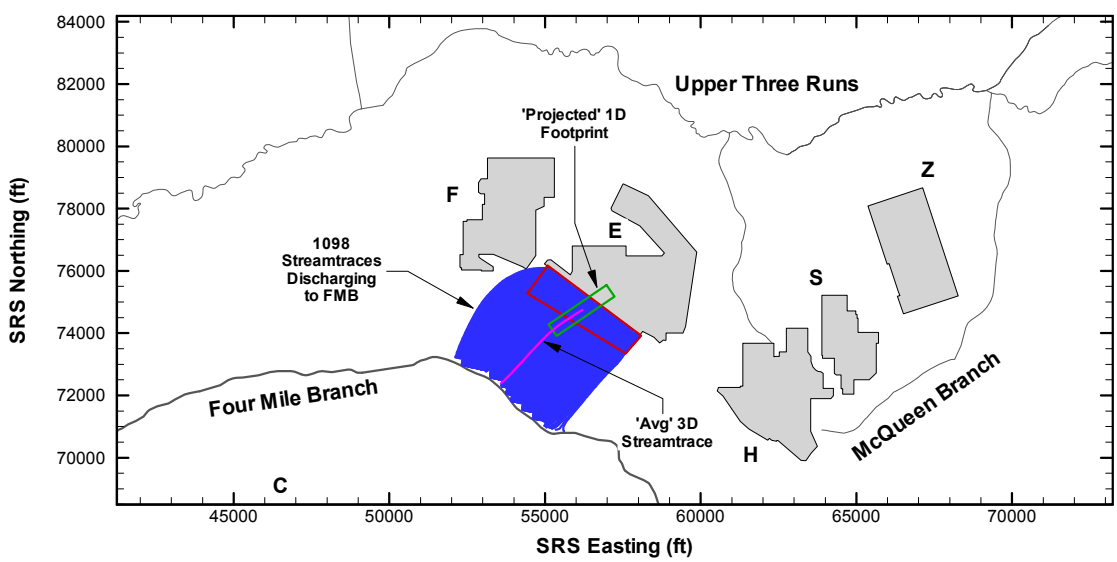


Figure 9-40. Illustration of Source Area Footprint Distribution

The computational approach taken to estimate the required 1D aquifer modeling parameters (described in detail in *Savannah River Site Composite Analysis: Aquifer Flow Path Parameters*, Hamm et al. 2009) was to track a large number of 3D streamtraces emanating from each CA source unit. Approximately 1,000 uniformly-distributed streamtraces were tracked over the aerial footprint of each source unit to its appropriate discharge point(s) (i.e., seepage faces). During this tracking (integration) process, streamtrace dependent values for variables such as travel distance, travel time, and Sandy/Clayey pore velocities are obtained. The average streamtrace Sandy and Clayey travel lengths, unretarded times, and unretarded pore velocities for each CA source unit are listed in tables in Section 4.2 of Hamm et al. (2009). From this information average 1D aquifer parameters are computed for each CA source unit of interest, as well as distributions of these variables.

To illustrate the process of computing the required 1D aquifer modeling parameters, selected results from the E-Area ORWBG source unit are shown below in Figure 9-41. The 1098 3D streamtraces shown in blue begin at the water table just beneath the ORWBG footprint (uniformly placed) and then travel to their outcrop locations along the seepage faces associated with FMB. These streamtraces were computed using the GSA Flow Model under nominal ("best estimate") conditions. An 'Avg' 3D streamtrace (shown in magenta) is computed from these 1098 streamtraces that is used to determine the 1D aquifer segment lengths. The "Projected" 1D footprint (shown in green) is used to determine the inventory profile in the direction of the 'Avg' streamtrace.



An 'Avg' 3D streamtrace is shown in magenta and the 'Projected' 1D footprint of this unit is shown in green.

Figure 9-41. 3D Streamtraces (in blue) Emanating from the E-Area ORWBG and Discharging into FMB

Tracking a large number of streamtraces emanating from a CA source provides average aquifer values as well as statistical information. For the ORWBG example, the variation in overall travel distance of the 3D streamtraces can be viewed in the histogram plot shown in Figure 9-42. Several statistical values are provided in Figure 9-42 where an approximately 25% spread in travel distance is seen in the population of 3D streamtraces considered. Given the relative size of the E-Area ORWBG footprint versus the average travel distance, a 25% spread in travel distance is to be expected. For units with a small footprint versus its average travel distance, a much smaller spread is expected.

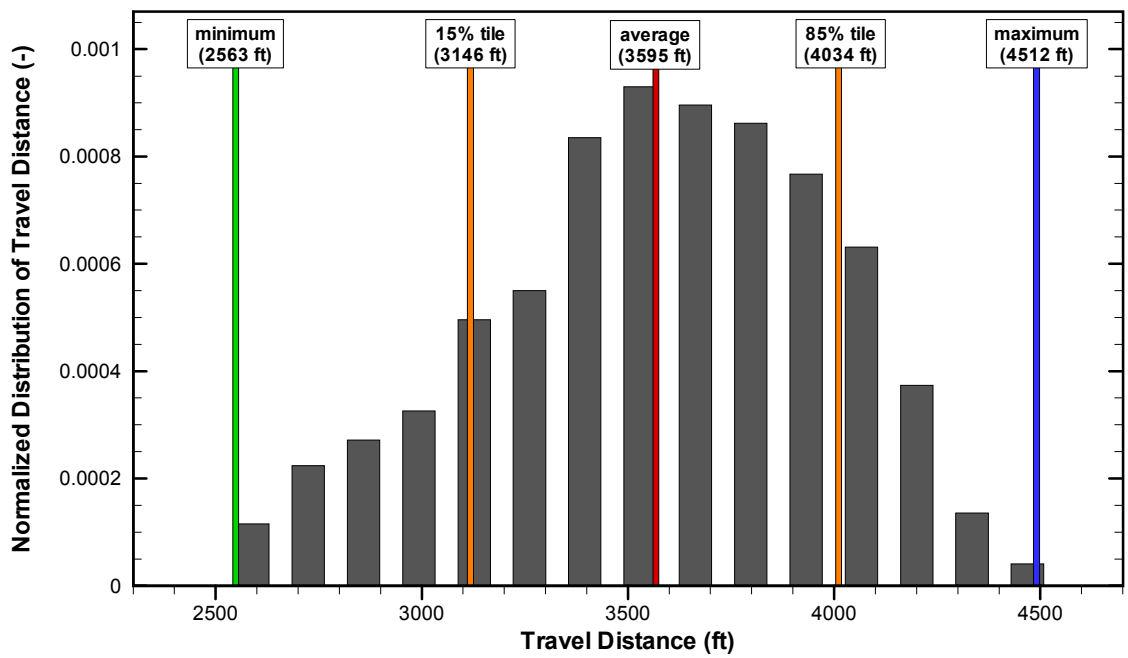


Figure 9-42. Histogram Plot of 3D Stream Trace Overall (Sandy plus Clayey) Travel Distance for the E-Area ORWBG CA Unit.

In Figure 9-43 through Figure 9-46, the 'Avg' 3D streamtrace (in magenta) and 'Projected' footprint (in green) for all CA source units of interest within the AM, CKLP, GSA, and R Flow Model domains are shown, respectively. There are 6 source units in the AM model domain, 24 units in the CKLP model domain, 89 units in the GSA model domain, and 4 units in the R model domain for a total of 123 CA source units. The correspondence between these 123 units, and the 137 surface sources identified in Sections 9.2 and 9.3.1.6, is provided in Table C-1 of Appendix C. Close-up views of the source units' footprints and streamtraces within the four Flow Model domains are provided in Appendix D (and Section 4.2 of Hamm et al. [2009]).

A very broad range in travel lengths and travel times for these 'Avg' 3D streamtraces exists. For each flow model domain and overall domains the mean, minimum, and maximum values for the travel distance (ft) and travel time (yr) are provided in Table 9-16. These travel times are unretarded times based on the local pore velocities along a streamtrace. The range in travel distance is a minimum of 437 ft for the T-Area "Outfall Delta and Swamp" to a maximum of 50,117 ft for the M-Area "Inactive Process Sewer Lines (081-M)". For all 123 CA source units, a mean travel length of 6,012 feet and mean travel time of 157 years is observed.

Table 9-16. The 'Avg' 3D Stream Trace's Mean, Minimum and Maximum Travel Distance (ft) and Travel Time (yr) for all CA Source Units within Each Flow Model Domain and for All Domains

Streamtrace Parameter	Statistical Measure	AM	CKLP	GSA	R	Overall
Travel Distance (ft)	min	18858.2	436.7	576.7	3248.0	436.7
	max	50117.4	30479.0	17399.6	5679.4	50117.4
	mean	38391.7	3780.5	4524.6	3929.2	6012.1
Travel Time (yr)	min	74.0	1.4	1.4	25.3	1.4
	max	2674.0	294.4	687.7	210.8	2674.0
	mean	1143.0	42.1	125.2	78.3	157.1

In the AM Flow Model domain (refer to Section 9.2.1 for additional details regarding the AM Flow Model) shown in Figure 9-43 there are 6 CA source units located within the following process areas:

- 4 units in A-Area
- 2 units in M-Area

CA source units within the AM Flow Model domain discharge to the UTR and SR POAs (see Figure 4-2 in Section 4.2.

In the CKLP Flow Model domain (refer to Section 9.2.1 for additional details regarding the CKLP Flow Model) shown in Figure 9-44 there are 24 CA source units located within the following process areas:

- 3 units in C-Area
- 5 units in D-Area
- 4 units in K-Area
- 3 units in L-Area
- 1 unit in N-Area
- 3 units in P-Area
- 5 units in T-Area

CA source units within the CKLP Flow Model domain discharge to the UTR, FMB, SC/PB, and Savannah River POAs.

In the GSA Flow Model domain (refer to Section 9.2.1 for additional details regarding the GSA Flow Model) shown in Figure 9-45 there are 89 CA source units located within the following process areas:

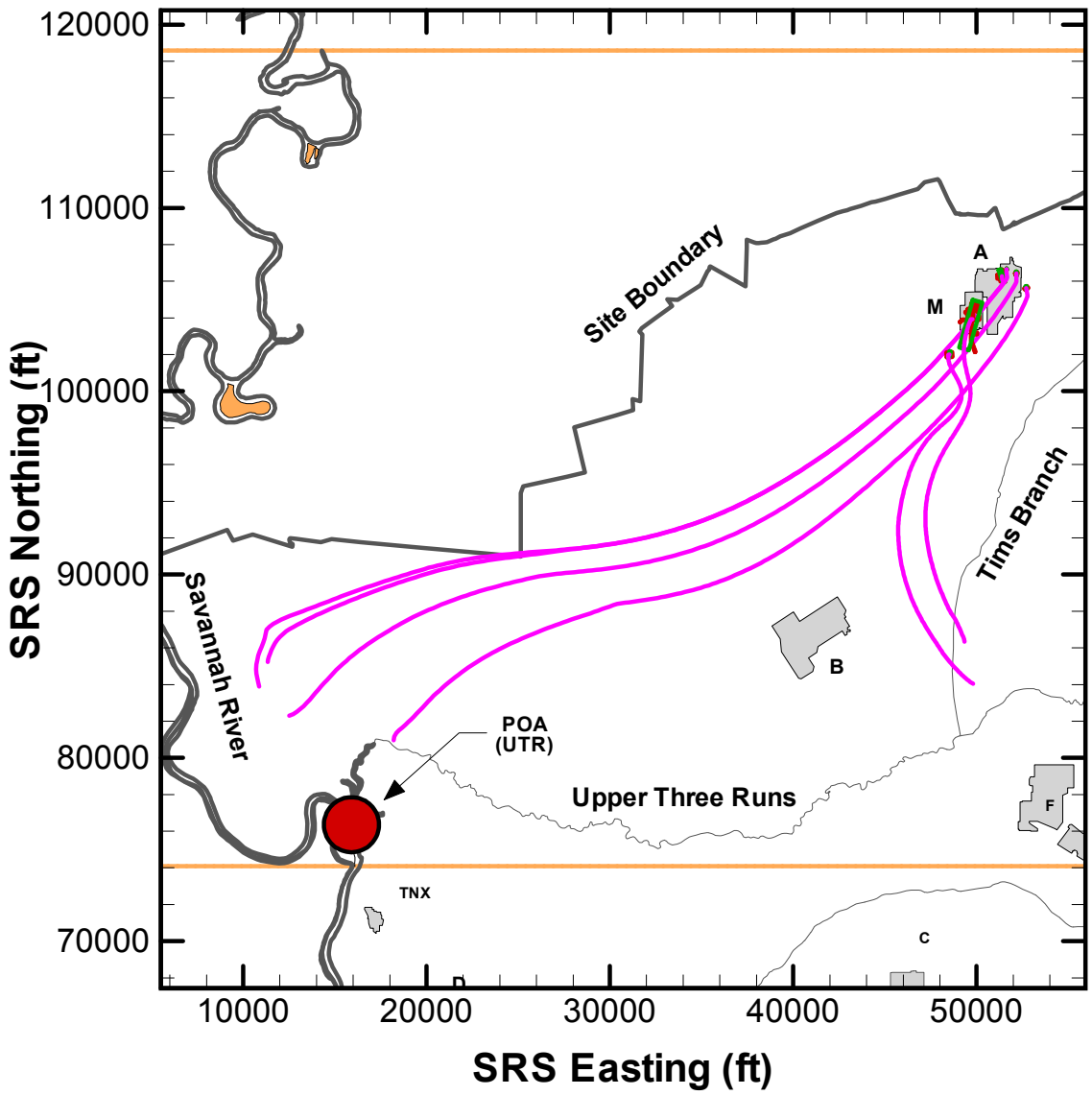
- 24 units in E-Area (11 of these units are for a proposed ELLWF Completion Project, or ECP)
- 19 units in F-Area
- 36 units in H-Area
- 1 unit in J-Area
- 4 units in S-Area
- 5 units in Z-Area

CA source units within the GSA Flow Model domain discharge to the UTR and FMB POAs.

In the R Flow Model domain (refer to Section 9.2.1 for additional details regarding the R Flow Model) shown in Figure 9-46 there are 4 CA source units located within the following process areas:

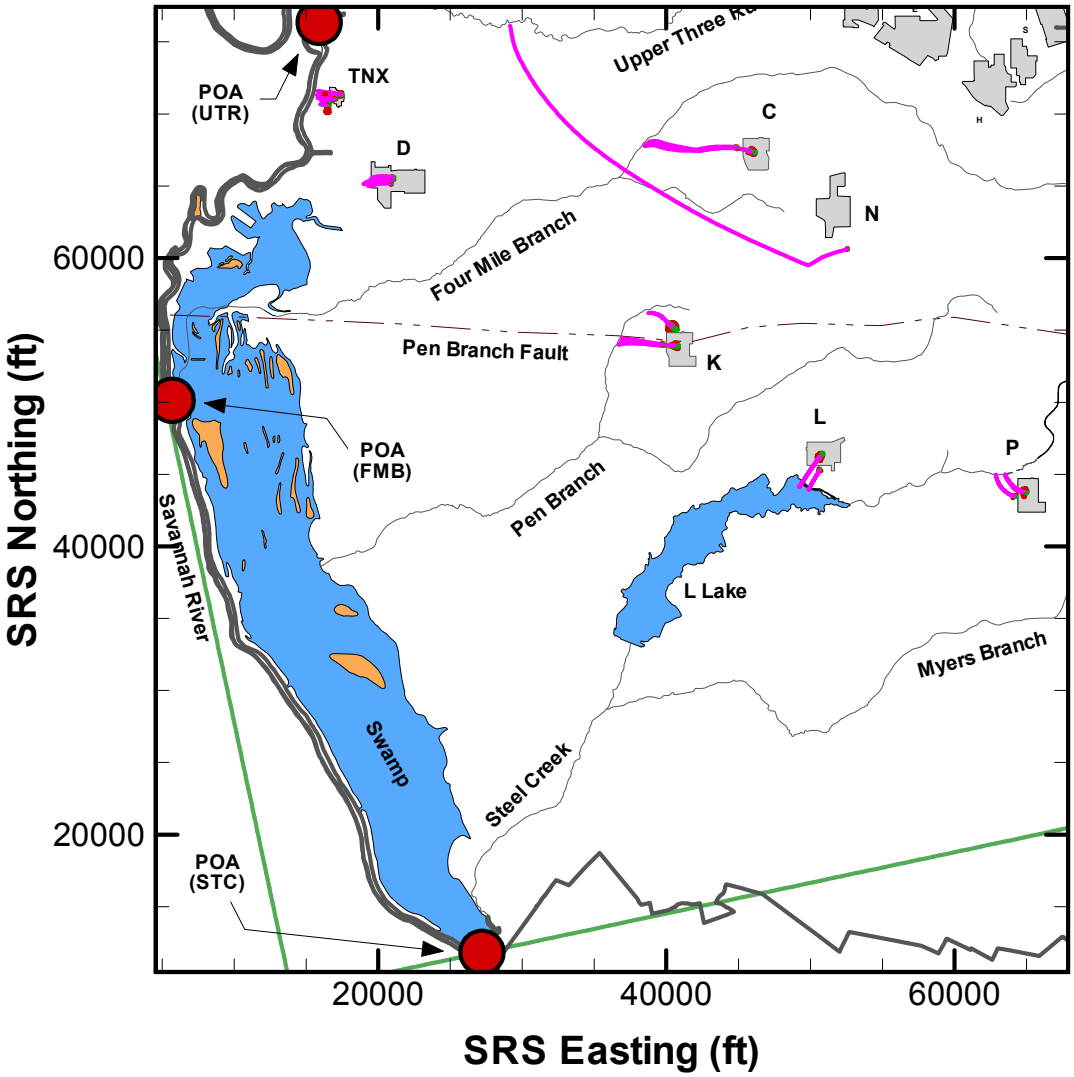
- 4 units in R-Area

CA source units within the R Flow Model domain discharge to the LTR POA.



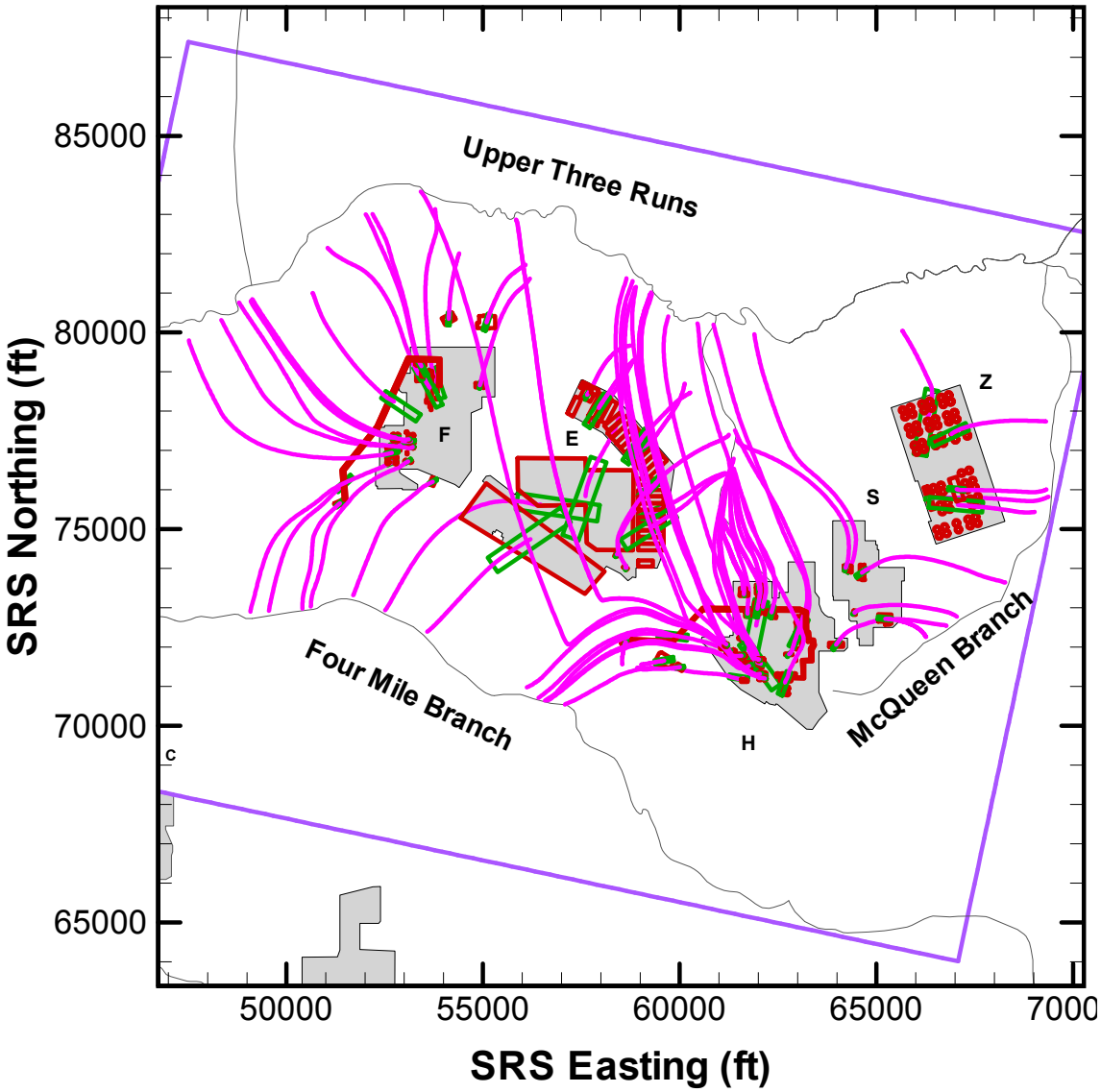
Each unit's average streamtrace from its footprint to its corresponding discharge point is shown (from Hamm et al. 2009).

Figure 9-43. View of CA Source Units within the AM Flow Model Domain



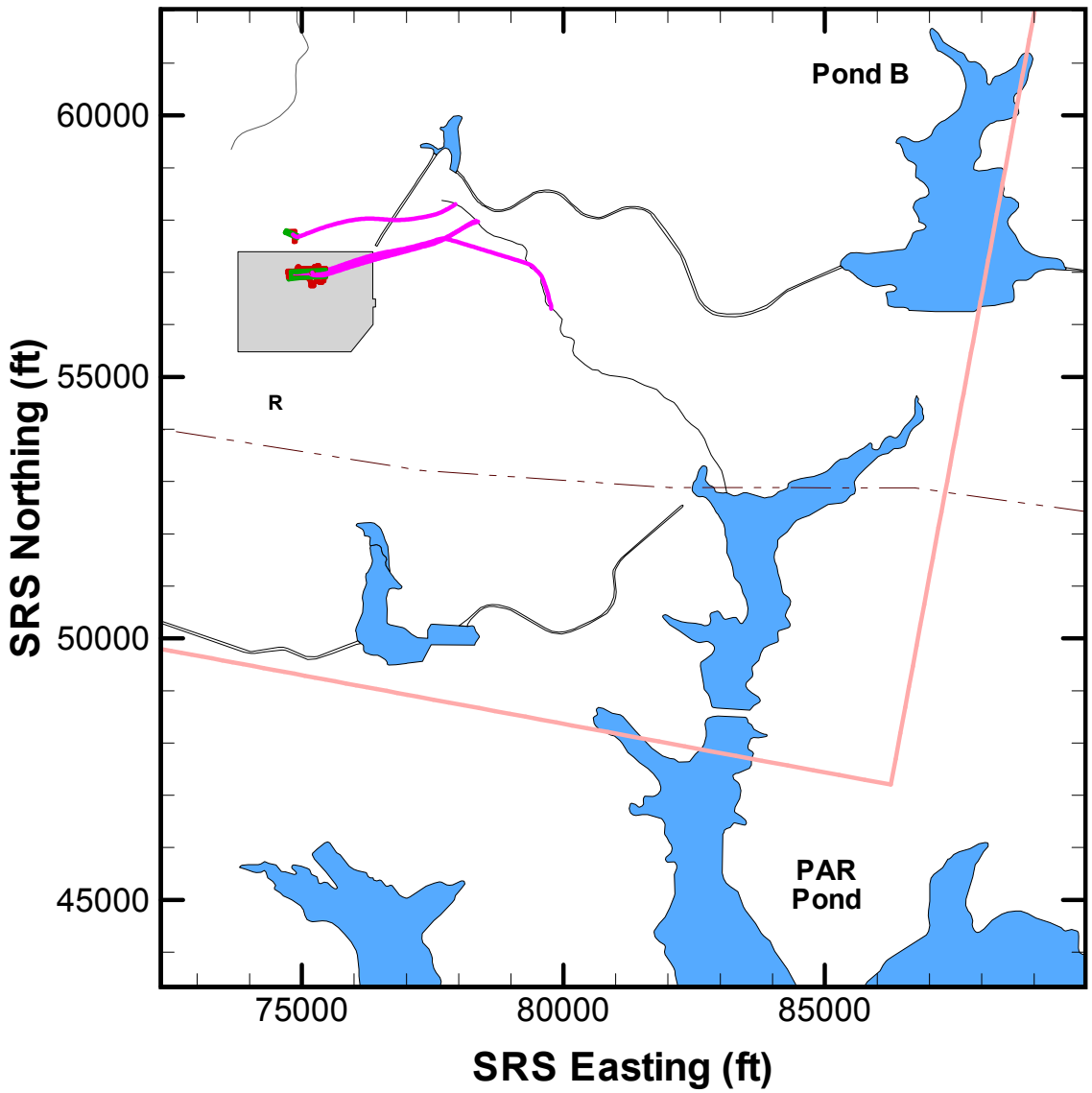
Each unit's average streamtrace from its footprint to its corresponding discharge point is shown (from Hamm et al. 2009).

Figure 9-44. View of CA Source Units within the CKLP Flow Model Domain



Each unit's average streamtrace from its footprint to its corresponding discharge point is shown (Hamm et al. 2009).

Figure 9-45. View of CA Source Units within the GSA Flow Model Domain



Each unit's average streamtrace from its footprint to its corresponding discharge point is shown (Hamm et al. 2009).

Figure 9-46. View of CA Source Units within the R Flow Model Domain

The 1D GoldSim™ aquifer model input is in the following format for each CA source unit of interest:

1. The POA for each CA source unit's discharge seepage face is defined using a numeric number (Model ID number) from 1 to 5, provided in Table 9-7, and shown below:

Stream or River	Model ID
UTR	1
LTR	2
FMB	3
SC/PB	4
SR	5

2. The overall aquifer model length in feet. This total length represents the total arc length of the average 3D streamtrace plus the up front length associated with the projected CA source unit's footprint.
3. The Segment 1 arc length in feet representing the first 100 GoldSim™ cells (see Figure 9-15) that are required to be sandy material only.
4. The Segment 3 arc length in feet representing the next 40 GoldSim™ cells (i.e., directly connected to the downstream end of Segment 1, Figure 9-15). This segment can accommodate both a section of clayey material and a downstream section of sandy material.
5. The material-averaged velocity (i.e., average streamtrace pore velocity) to be used for all sandy sections of the 1D aquifer flow path in ft/yr.
6. The material-averaged velocity (i.e., average streamtrace pore velocity) to be used for all clayey sections of the 1D aquifer flow path in ft/yr.
7. The number of cells in the front end of Segment 3 that contains clayey material (i.e., all of the volume of clayey material within the 1D aquifer model is collected and placed into these specific cells where every cell is either completely sandy or clayey material [no mixture cells allowed]).
8. The inventory fractions (i.e., that sum up to unity) for the 100 cells defined in Segment 1 (Figure 9-15). In the majority of cases, Segment 1 extends beyond the projected CA Unit's footprint where for those cells an inventory fraction of zero is applied.

Items 1 through 7 are provided in Tables D-1 through D-4 in Appendix D, for each of the CA source units identified for the four model domains (AM, CKLP, GSA, and R). Item 8 is provided in Tables D-5 through D-8 of Appendix D. In Appendix C, Table C-1 provides the correlation between the CA Base Case sources modeled and the applicable aquifer flow path input data set.

Key attributes for all of the CA source units' streamtrace studies are depicted in 123 sets of five unit-specific figures in the first four appendices of Hamm et al. (2009). Each set of five figures illustrates: a) a 2D areal view of 3D streamtraces, the average streamtrace emanating from the source unit's footprint and discharging to its predicted seepage face (i.e., POA); b) the inventory distribution along the 'projected' unit's footprint; c) the clay fraction profiles along the average 3D streamtrace; d) the probability density function (pdf) for the transport travel time of a conservative tracer; and e) the pdf for the travel distance (i.e., arclength along the 3D streamtraces) through the aquifer. Two of these sets of five figures are shown as examples in Figure 9-47 and Figure 9-48. These figures were chosen as they illustrate one source unit for which streamtraces discharge to seepage faces associated with more than one POA; in this case, streamtraces from the F-Area Type 3A tanks discharge to both FMB and UTR.

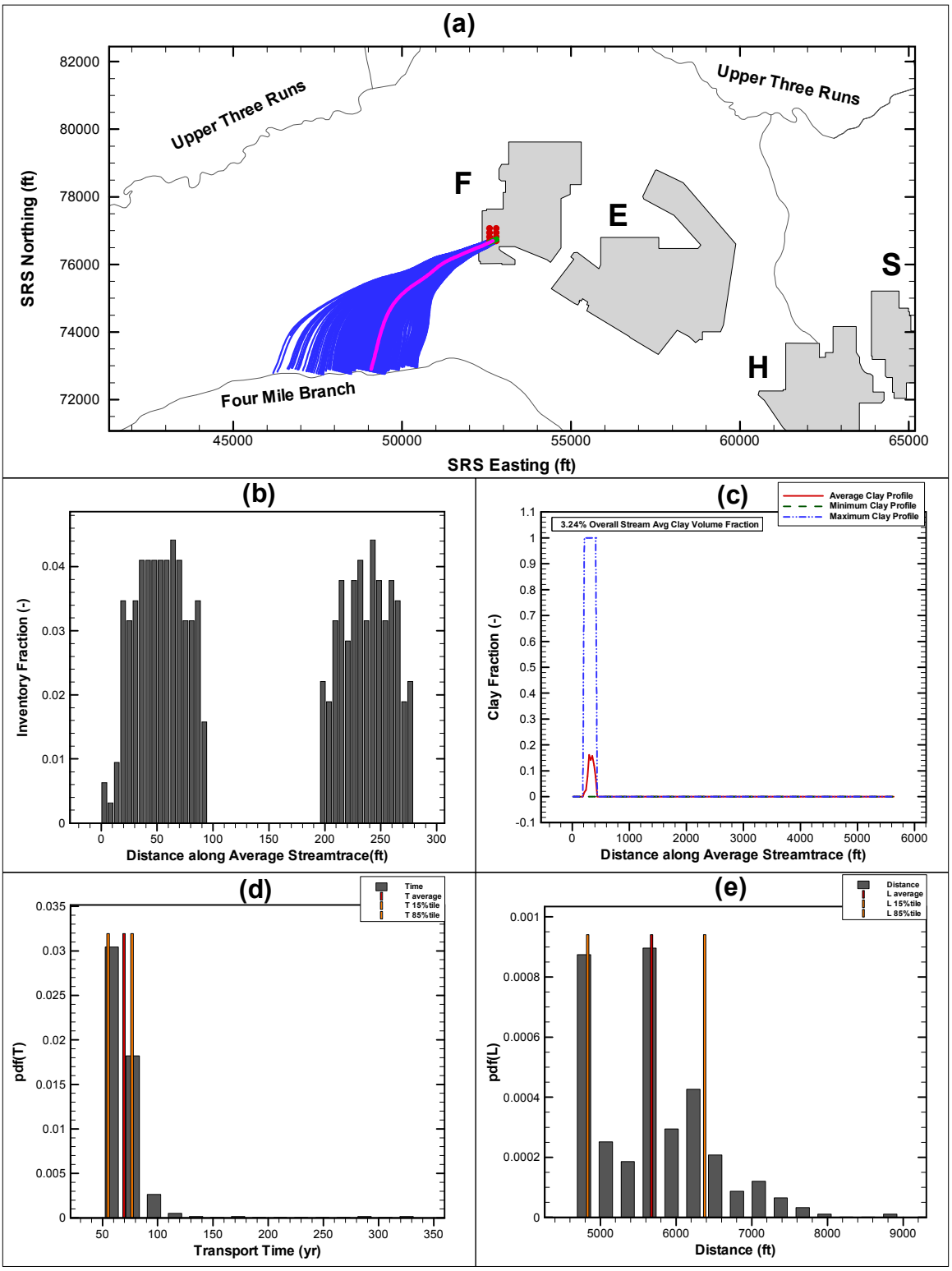


Figure 9-47. F-Area Type 3A Tanks Discharging to FMB POA and Based on the GSA Flow Model (from Hamm et al. 2009).

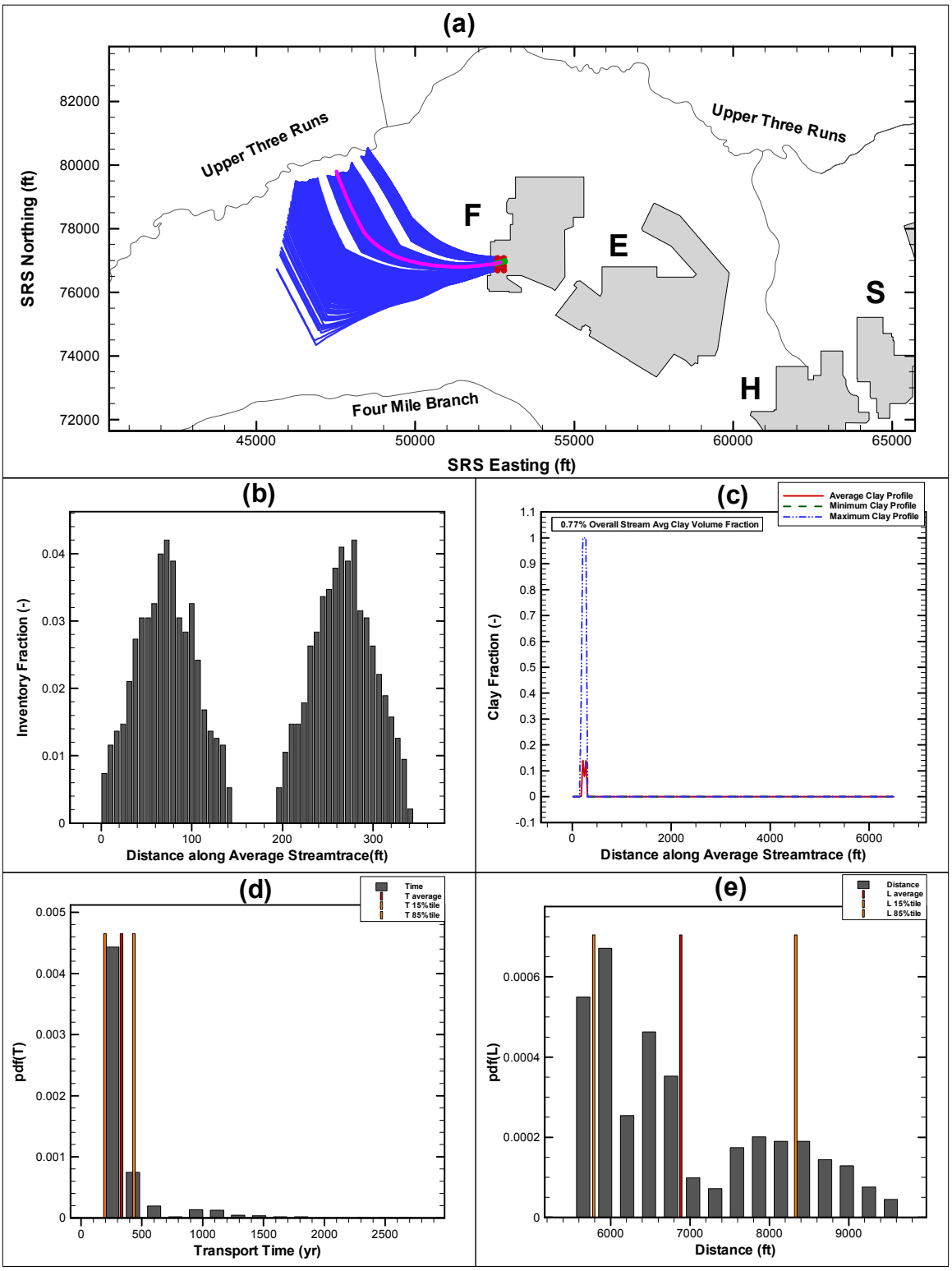


Figure 9-48. F-Area Type 3A Tanks Discharging to UTR POA and Based on the GSA Flow Model (from Hamm et al. 2009).

9.3.4.5 Stream and River Flows

Stream and river flows are of primary importance to the dose calculation because they determine the dilution of the released radionuclides and hence the water concentrations used to calculate doses. Average annual stream and river flows used for deterministic base-case calculations were obtained from Jones (2009). A summary of the flows used for CA base case modeling (extracted from Table 3-1 of Jones, 2009) are given in Table 9-17. Stream flows are input to the model through the Access database. For all base case CA calculations, the river flow was assumed to be the flow measured at Augusta, GA.

Table 9-17. Savannah River and Tributary Stream Flow Summary

Stream Name	Location	Average Annual Flowrate for Base Case CA Modeling (cfs)
Savannah River	Augusta, GA[1]	9,198
Savannah River	U.S. Highway 301 Bridge[2]	10,175
Upper Three Runs	Highway 125	237
Fourmile Branch	Near D-Area	32
Pen Branch plus Steel Creek	Near Savannah River Swamp/Highway 125	89
Lower Three Runs	Martin, SC (modified with Snelling, SC data)	164

[1] Flow to be used for all Savannah River POAs except for U.S. Highway. 301 Bridge POA

[2] Flow to be used for Burtons Ferry (U.S. Highway 301 Bridge) POA. (See Figure 4-2)

9.3.4.6 Infiltration

Infiltration is controlled by the entries in Disposal_Units.xls worksheet cells B4 through B6. Table C-5 in Appendix C provides the infiltration values assumed for the types of caps and for background soils. Cell B6 specifies the type of cap used which determines the infiltration rate as described in Section 9.3.1.6.2.2. Cell B4 specifies the year the cap is placed which is the time when infiltration starts following the designated curve. If the waste inventory is placed before the cap, infiltration will be the background value of 15 in/yr until the cap placement year. Cell B5 is used to enter a cap failure time. This was only used for metal caps that initially block infiltration then suddenly fail due to corrosion.

9.3.4.7 Cellulose Degradation Product and Source Release Parameters

Input entries in cells B18 through B24 of the Disposal_Units.xls worksheets provide specific source release modeling parameters and CDP activation as explained by the descriptions provided in Table 9-15. The entry in cell B18 is used to specify the source release mechanism. This value was set to 4 for the generic release calculations where the source is released directly into the vadose zone. The source release rate in cell B19 specifies the constant fractional release rate of the source inventory into the waste zone when release mechanism 2 is selected.

The CDP activation switch (cell B20) was used for both generic and source release calculations. CDP factors (see Section 9.3.4.9) were applied to all waste materials having a significant amount of cellulosic material. These were the E-Area facilities: Components in Grout (CIG) Trenches, Engineered Trenches (ET), LAWV, Slit Trenches (ST), TRU Pad-1, LLRWDF, MWMF, and ORWBG.

Cell B21 is used to specify the PA flux used for source release calculations. The specified PA flux to the water table will actually be used in the calculation in addition to any inventory released through another mechanism. Setting the value of cell B21 to zero reads in a flux table with zero fluxes so no additional source is included in the calculations. Use of PA fluxes is explained in Section 9.3.1.6.6.1.

The entries in cells B22 (streambed water volume) and B23 (streambed flow) are used to model source releases from sediments as discussed in Section 9.3.1.6.8. The source area in cell B24 will have no impact on the calculations except for solubility controlled releases as discussed in Section 9.3.1.6.5.1.

9.3.4.8 Isotopic Species

The 49 isotopic species identified by the CA Screening study (Taylor et al. 2008) as those radionuclides most likely to produce significant CA doses were used in this study. This list of isotopes was augmented for the CA transport modeling by including the three daughter isotopes with half-lives greater than three years (Ac227, Nb93m and Ra228) to produce the list of 52 species identified as CA species in Appendix C, Table C-6. Inventories for these 52 isotopes were entered into the model and transport and decay rates calculated for these species.

The Neptune and Company, Inc. developed dose module (Phifer et al. 2009) has been developed for use in support of a variety of PA and CA efforts involving multiple facilities at the SRS. Therefore, it was also necessary to include the other 11 isotopes, for a total of 63, identified in Appendix C Table C-6 as Non-CA species in the CA model for compatibility with the species list used by the dose module. However, no inventory was entered for Non-CA species so concentrations for these isotopes will be identically zero in the stream and river water.

The use of full decay chains in the dose module calculations (see Section 9.1.3) further required expanding the number of isotopes from the 63 species (52 CA species plus the 11 Non-CA species) that appear in the transport and decay model to 115 (the 63 plus 50 CA daughters with half-lives less than 3 years and 2 Non-CA daughters with half-lives less than 3 years). For the CA this results in the actual transport modeling of 52 parents plus daughters with half-lives greater than 3 years and the actual dose modeling of these 52 plus the 50 CA daughters with half-lives less than 3 years for a total of 102 radionuclides considered within the dose module. In Appendix C, Table C-6 provides a listing of the 115 radionuclides and denotes which are and which are not CA species and which are considered for transport and which for dose.

Figure 9-49 through Figure 9-54 illustrate the decay chains associated with the 102 radionuclides considered within the CA (Smith et al. 2009a, Phifer et al. 2009). In Appendix C, Table C-7 provides the half-lives, molecular weight, and sandy soil K_ds for the 102 radionuclides considered within the CA. Also in Appendix C, Table C-8 provides the branching fractions for the 102 radionuclides considered within the CA, and Table C-9 provides supplemental branching fraction information (discussed in Section 9.3.4.10) for the 50 CA daughters with half-lives less than 3 years (Phifer et al. 2009).

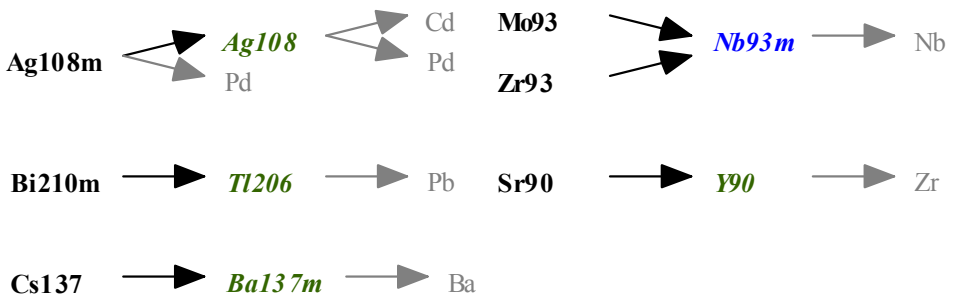


Figure 9-49. CA Radionuclides Associated with Non-Actinide Decay to Dose-Producing Daughters (Progeny)

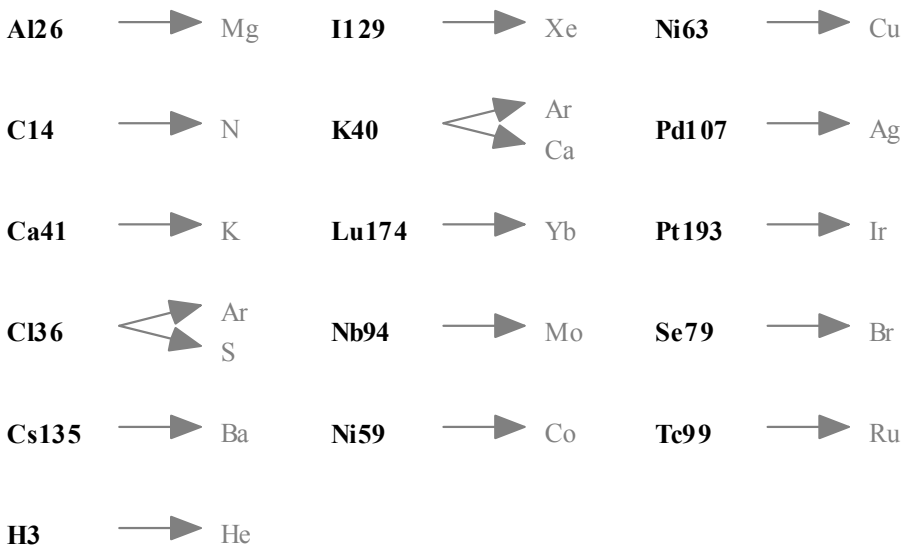


Figure 9-50. CA Radionuclides Associated with Non-Actinide Decay to Stable Daughters (Progeny)

LEGEND:	
Bold Black Lettering	49 CA Parent Radionuclides
Bold Blue Italic Lettering	Three CA radionuclide daughters (progeny) with half-lives >3 years
Bold Green Italic Lettering	50 CA radionuclide daughters (progeny) with half-lives <3 years
Grey Lettering	Stable daughters (progeny)

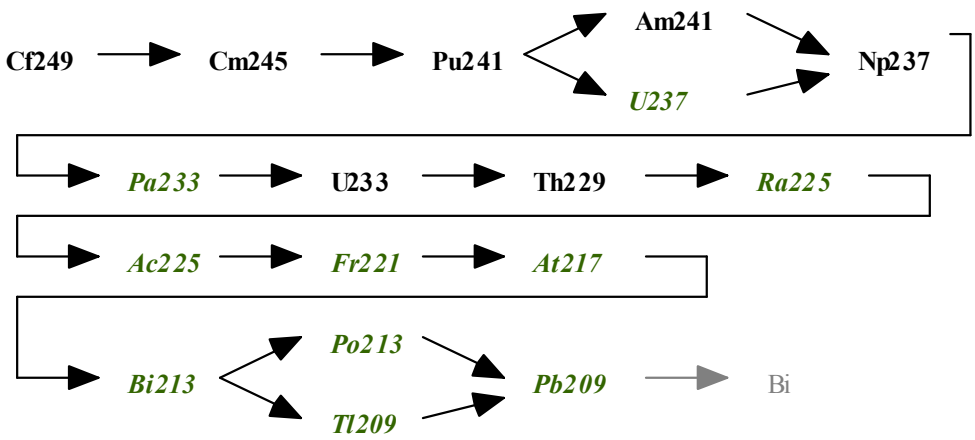


Figure 9-51. CA Radionuclides Associated with the Neptunium Series

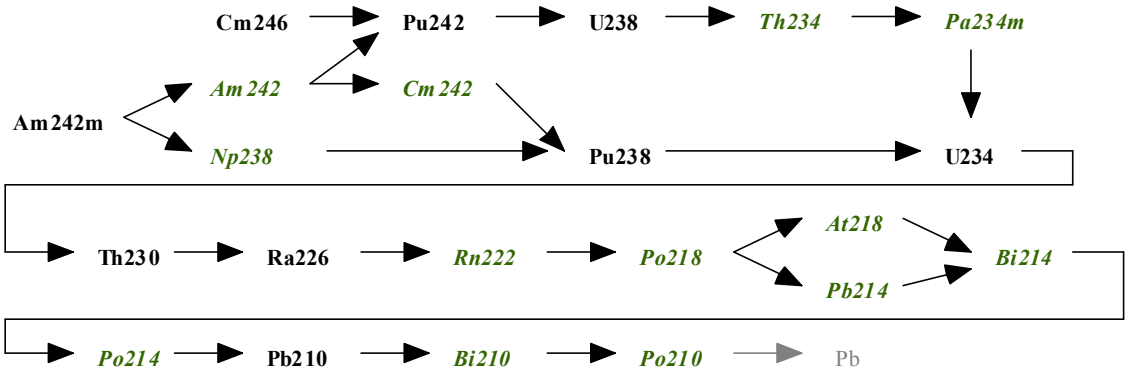


Figure 9-52. CA Radionuclides Associated with the Uranium Series (Plus Am242m)

LEGEND:	
Bold Black Lettering	49 CA Parent Radionuclides
Bold Blue Italic Lettering	Three CA radionuclide daughters (progeny) with half-lives >3 years
Bold Green Italic Lettering	50 CA radionuclide daughters (progeny) with half-lives <3 years
Grey Lettering	Stable daughters (progeny)

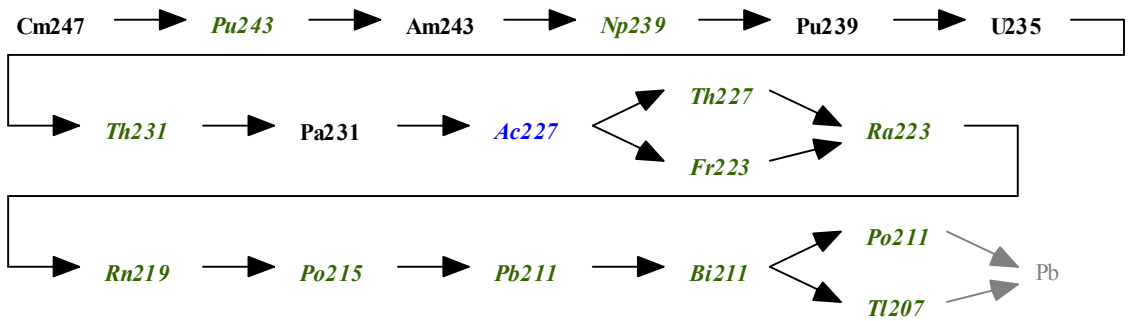


Figure 9-53. CA Radionuclides Associated with the Actinium Series

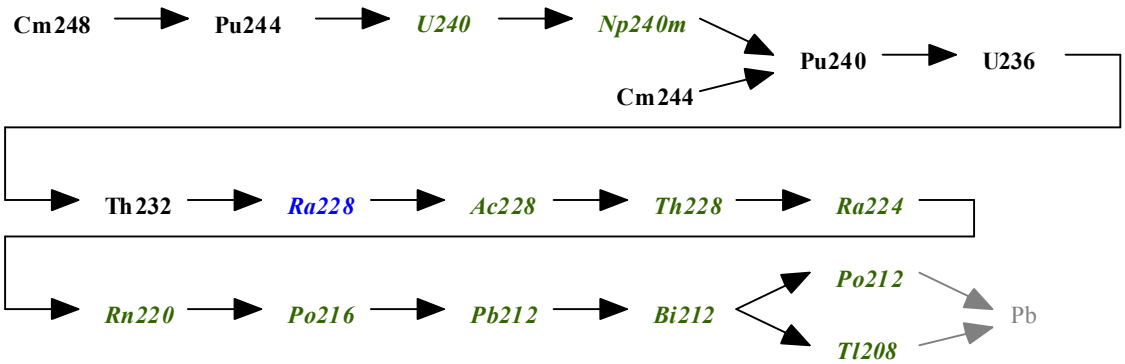


Figure 9-54. CA Radionuclides Associated with the Thorium Series (Plus Cm244)

LEGEND:	
Bold Black Lettering	49 CA Parent Radionuclides
Bold Blue Italic Lettering	Three CA radionuclide daughters (progeny) with half-lives >3 years
Bold Green Italic Lettering	50 CA radionuclide daughters (progeny) with half-lives <3 years
Grey Lettering	Stable daughters (progeny)

9.3.4.9 Materials

The following five materials are used in the CA model: vadose zone sandy soil, vadose zone clayey soil, saturated sandy soil, saturated clayey soil and oxidizing concrete. Physical properties required for CA model calculations are: dry bulk density, porosity, water saturation, K_d, CDP factor, and tortuosity.

Nominal values for the porosity, dry bulk density and tortuosity of the CA materials listed in Table C-10 (from Phifer and Dixon 2009) were used in the base case deterministic calculations. These property values were entered into the CA Access database and read from the database into the GoldSim™ model. Saturation values for sandy soil and clayey soil are listed in Table C-11 (from Phifer and Dixon 2009) as functions of the infiltration rate in tabular form. Nominal saturation values (termed "most likely" values in the reference) were obtained through deterministic PORFLOW modeling as described in the reference. To implement soil saturation as a function of infiltration in the GoldSim™ model, the nominal saturation values were fit to functional forms. The empirical saturation fitting function used for this purpose was:

$$S = a\frac{\left(1 + b\,I + c\,I^2\right)}{\left(1 + d\,I\right)} \tag{9-8}$$

In Equation (9-8), S is the fractional soil saturation, I is the background infiltration (in/yr), and a, b, c and d are fitting coefficients. Table 9-18 gives the fitting coefficients found and Figure 9-55 and Figure 9-56 show the comparison between the fit and tabulated data for sandy soil and clayey soil saturation, respectively. Over the range of infiltrations used, the fitting function gives an excellent approximation to the tabulated values. The saturation functions were hardwired in the model.

Table 9-18. Saturation Function Fitting Coefficients

Fitting Coefficient	Sandy Soil	Clayey Soil
a	0.5371	0.7312
b	0.7262	0.7807
c	0.0026	0.0025
d	0.5882	0.6884

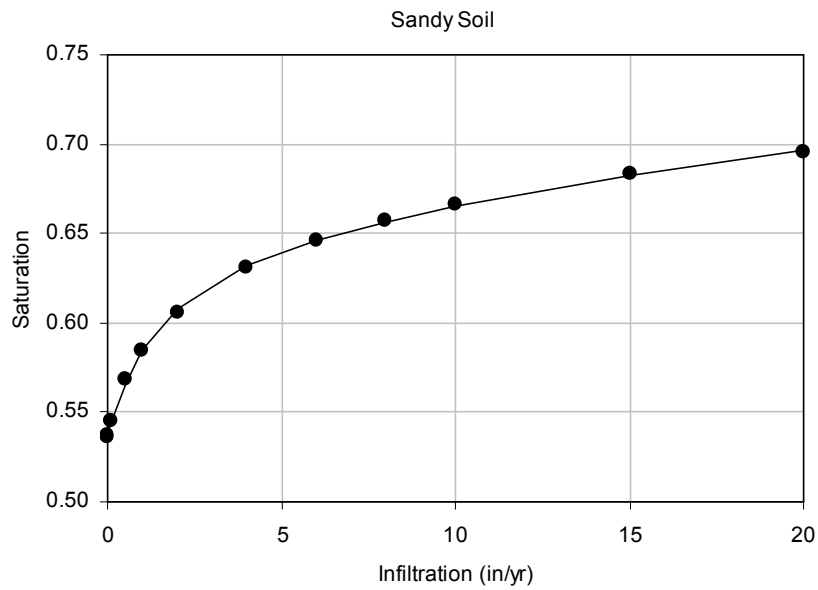


Figure 9-55. Fit of Sandy Soil Saturation as a Function of Infiltration Rate

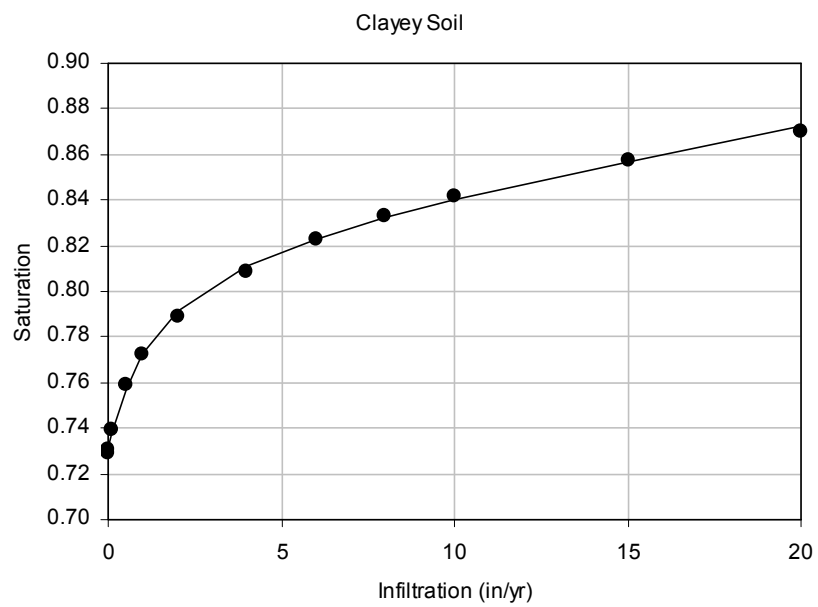


Figure 9-56. Fit of Clayey Soil Saturation as a Function of Infiltration Rate

Nominal values for K_d and CDP factors used in the CA base case deterministic calculations are given in Tables C-12 and C-13 of Appendix C, and were obtained from the Tables 1 and 6 in *Distribution Coefficients (Kds), Kd Distributions, and Cellulose Degradation Product Correction Factors for the Composite Analysis* (McDowell-Boyer and Kaplan (2009)), respectively. These property values were entered into the Parameters.xls workbook in worksheet "Element" and read from the excel spreadsheet into the GoldSim^{TM} model. K_d and CDP factors are functions of the elemental species adsorbing on the material. K_d values for sandy soil, clayey soil, and young, middle and old aged concrete are used in the CA model. Sandy soil K_ds are also used in the dose module to estimate the holdup of radionuclides in irrigated soil and shoreline soil. The user must create a separate table of K_ds for use in the dose module. To be consistent, this list was populated with the same K_d values used in the model transport calculations. However, because the dose module calculations are based on full decay chains instead of the truncated three year half-life chains used elsewhere in the model, the five additional elements Ba, Fr, Po, Tl and Y are introduced. K_d values for these elements are included in the report by McDowell-Boyer and Kaplan (2009). Because these five elements do not appear in the transport calculations, their K_d values were not included in the Parameters.xls file. As an interim measure, K_ds for these elements were entered as fixed values internally to the CA model. In future versions of the CA model, for better QA control, these values will be moved to the Parameters.xls file.

9.3.4.10 Dose Module Input Parameters

As outlined in Sections 9.1.3 and 9.3.4.8, the CA has considered the transport of 52 radionuclides and the dose associated with 102 radionuclides (i.e., the 52 transport radionuclides plus 50 daughters with half-lives less than or equal to three years). The activity concentration of the 52 CA radionuclides in the streams and river from the transport module are imported into the dose module. Within the dose module, secular equilibrium was assumed for these 50 daughters (i.e., those with half-lives less than or equal to three years) with the closest preceding member in their decay chain for which a concentration is calculated in the transport model (Smith et al. 2009a). Then the activity concentration for the shorter-lived daughters was calculated based upon the activity concentration of the closest preceding member in their decay chain and their associated branching fractions as outlined below. Thus, the time history of stream or river concentrations output by the transport model is modified to account for the short-lived daughter equilibrium assumed when input to the dose module.

Decay chain details pertaining to the CA dose module are illustrated in Figure 9-49 through Figure 9-54. The 49 CA parent radionuclides are shown in bold black lettering in these figures, the three CA radionuclide daughters with half-lives greater than three years are shown in bold blue italic lettering, and the 50 shorter-lived daughters with half-lives less than three years are shown in bold green italic lettering. Stable end members are included and shown in grey lettering for completeness.

Each of the 102 radionuclides considered by the dose module is listed in Appendix C, Table C-8, along with its predominant decay mode(s), associated branching fraction(s), and the resulting daughter(s). The radionuclides in the table are color coded the same as in Figure 9-49 through Figure 9-54 for ease of use. The decay chain figures (Figure 9-49 through Figure 9-54) were used in conjunction with the Appendix C, Table C-8, branching fractions to calculate the secular equilibrium concentrations for the 50 shorter-lived daughters. This calculation is explained by the following three examples:

1) From Figure 9-49 it is seen that Ag108 is a short-lived daughter of Ag108m, and that an activity concentration was calculated for Ag108m within the transport module. From Appendix C, Table C-8, it is seen that the branching fraction from Ag108m to Ag108 is 0.087. Therefore the activity concentration of Ag108 is 0.087 times the activity concentration of Ag108m.
2) From Figure 9-51 it is seen for Bi213 that Th229 is the closest preceding decay chain member for which an activity concentration was calculated in the transport module. It is also seen that Ra225, Ac225, Fr221, and At217 are in the decay chain between Bi213 and Th229. From Appendix C, Table C-8, it is seen that the branching fractions from Th229 to Ra225; Ra225 to Ac225; Ac225 to Fr221; Fr221 to At217; and Ac217 to Bi213 are 1; 1; 1; 1; and 0.9999, respectively. Therefore the activity concentration of Bi213 is 0.9999 (i.e., $1\times1\times1\times1\times0.9999$) times the activity concentration of Th229.
3) From Figure 9-52, it is seen for Cm242 that Am242m is the closest preceding decay chain member for which an activity concentration was calculated in the transport module. It is also seen that Am242 is in the decay chain between Cm242 and Am242m. From Appendix C, Table C-8, it is seen that the branching fractions from Am242m to Am242; and Am242 to Cm242 are 0.9955; and 0.827, respectively. Therefore the activity concentration of Cm242 is 0.8233 (i.e., 0.9955×0.827) times the activity concentration of Am242m.

Such use of Figure 9-49 through Figure 9-54 and Appendix C, Table C-8, has been made for each of the 50 shorter-lived daughters in order to provide the supplemental branching fraction vector for use in calculating secular equilibrium concentrations for use within the dose module. Appendix C, Table C-9 lists each of the 50 shorter-lived daughters along with the closest preceding decay chain member for which an activity concentration was calculated in the transport module and the supplemental branching fraction vector used to calculate secular equilibrium with that closest parent.

In the dose module where the doses for radionuclides reaching the POA are evaluated, the shorter-lived daughters are handled explicitly, rather than assumed to be in secular equilibrium. Thus, transfer factors, dose conversion factors, and exposure parameters and consumption rates, are addressed for elements corresponding to the 52 radionuclides as well as for 50 additional radionuclides with half-lives shorter than three years. This then requires the use of half-lives, molecular weights, K_ds, transfer factors, and dose conversion factors for a total of 102 radionuclides.

Table C-7 provides the half-lives, molecular weight, and sandy soil K_ds for the 102 radionuclides considered within the CA. Values of element-specific transfer factors and DCFs used in the CA dose module are listed in Appendix C Tables C-14 through C-17 by GoldSim™ model element ID. Tables C-14 and C-16 list transfer factors and DCFs for the 52 CA nuclides (49 parents plus daughters with half-lives greater than three years), respectively. Tables C-15 and C-17 list transfer factors and DCFs for the additional 50 daughters with half-lives less than three years, respectively. The transfer factors in Tables C-14 and C-15, used in the dose module to quantify movement of radionuclides to plants, beef, milk and fish, where they can be ingested, are consistent with those provided by Lee and Coffield (2008), and those used in the SRS PAs. The internal DCFs in Tables C-16 and C-17, used in calculating ingestion and inhalation dose, are consistent with ICRP Publication 72 values (ICRP 1995), when available. In the absence of ICRP Publication 72 values for several short-lived radionuclides listed in Table C-17, a value of zero was used, in accordance with DCFs provided in NCRP Report No. 123, Volume I (NCRP 1996). The external DCFs in Tables C-16 and C-17, used in calculating dose due to shore shine, submersion, and direct exposure to garden soil, were taken from EPA Federal Guidance Report No. 12 (EPA 1993). Table C-18 lists the exposure parameters and consumption rates used in the dose module calculations described in Section 9.3.2.2. Because the dose module is implemented with GoldSim™, the GoldSim™ Element ID is provided in Table C-18, along with the corresponding parameter symbol used in the dose module equations in Section 9.3.2.2.

9.3.5 Model Validation

Verification and validation testing was performed during the model development process to ensure that the GoldSim™ model was functioning correctly. SRNL QA for computer codes defines software verification as the type of activity performed in what is termed the design checking process (i.e., having an independent reviewer check that a specific part of the model is coded correctly and functions as expected). Design checking of the GoldSim™ CA model was performed and documented in a series of internal memorandums. We define model validation as comparison of code results to data and/or results from other codes. The model validation studies performed as part of the CA involved running PORFLOW and GoldSim™ models for essentially identical one dimensional transport paths. The following model configurations were tested:

1. Transport through 1,000 feet of saturated sandy soil with cellulose degradation product (CDP) factors applied. Plots of radionuclide flux to stream predicted at the exit of the transport path for this case are presented in Section 9.3.5.1.
2. Transport through 1,000 feet of saturated sandy soil with no CDP factors applied. Plots of radionuclide flux to stream predicted at the exit of the transport path for this case are presented in Section 9.3.5.2.
3. Transport through 5,000 feet of saturated sandy soil with no CDP factors applied. Plots of radionuclide flux to stream predicted at the exit of the transport path for this case are presented in Section 9.3.5.3.
4. Transport through 10,000 feet of saturated sandy soil with no CDP factors applied. Plots of radionuclide flux to stream predicted at the exit of the transport path for this case are presented in Section 9.3.5.4.

5. Transport through 900 feet of saturated sandy soil and 100 feet of saturated clayey soil with no CDP factors applied. Plots of radionuclide flux to stream predicted at the exit of the transport path for this case are presented in Section 9.3.5.5.

For the verification calculation in Case 5, having a clay layer in the transport path, the PORFLOW model was composed of a 450 ft section of saturated sandy soil followed by a 100 ft section of saturated clayey soil and a final 450 ft section of saturated sandy soil. The GoldSim™ model was not exactly identical and modeled transport through a one foot vadose zone of unsaturated sandy soil before the aquifer and approximated the last 550 ft of the aquifer as 99 ft of saturated clayey soil followed by 451 ft of saturated sandy soil. This model is illustrated in Figure 9-57. For the other test cases, the only difference between the GoldSim™ and PORFLOW models was that the GoldSim™ model included a one foot vadose zone while the PORFLOW model did not. These minor differences between the two models would not impact the results to any noticeable degree.

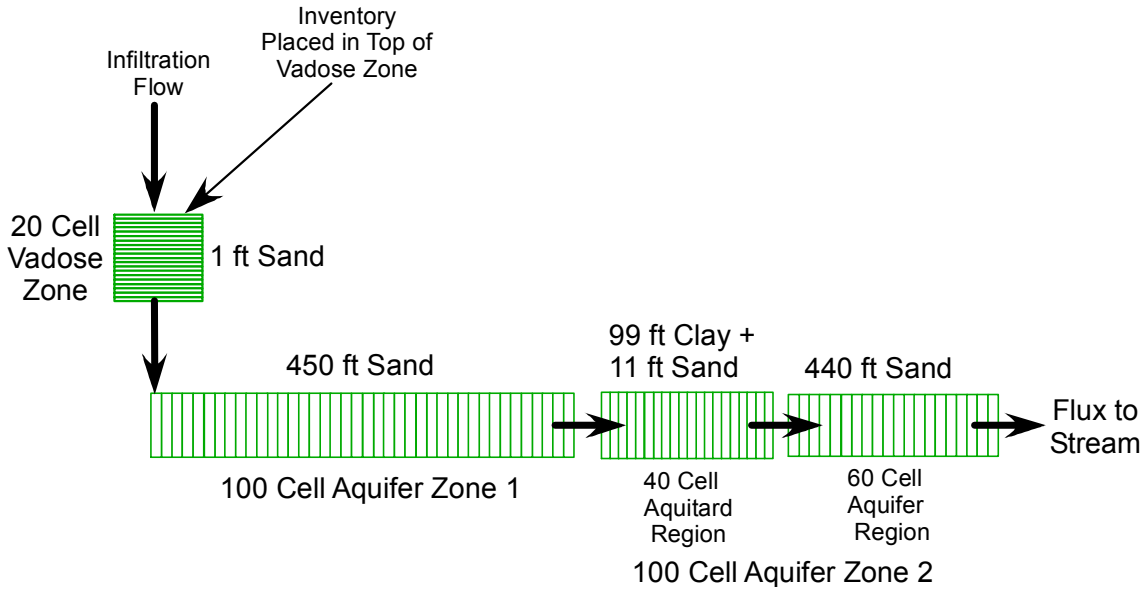


Figure 9-57. Model Used for GoldSim™ CA Model Validation

Calculations were performed starting with 1.0 Ci of each of the following radionuclides: Cl36, Cm248, Cs135, H3, I129, Pu239, Pu241, Tc99, and Th230. These parent isotopes have the following decay chains using a cut off of a three year half-life:

Cm248 → Pu244 → Pu240 → U236 → Th232 → Ra228

Pu239 → U235 → Pa231 → Ac227

Pu241 → Am241 → Np237 → U233 → Th229

Th230 → Ra226 → Pb210

The intent was to cover a wide range of K_d and half life values and to test both individual radionuclides and decay chains. Complete results from the model validation testing are presented in the subsections below where radionuclide fluxes in pCi/yr at the outlet of the transport path are plotted for both the GoldSim™ and corresponding PORFLOW calculations. In all of the plots, the solid curves show results from the PORFLOW calculation while the dashed curves are from the corresponding GoldSim™ calculation.

9.3.5.1 *Model validation results for 1,000 feet of saturated sandy soil with CDP factors*

Flux to stream results from the validation case testing radionuclide transport through 1,000 ft of saturated sandy soil with CDP factors applied are presented in Figure 9-58 through Figure 9-66. For all of the test calculations, the GoldSim™ results agree very closely with the corresponding PORFLOW results. In particular, peak fluxes from both calculations are nearly identical and occur at essentially the same time. Peak fluxes occur at approximately 10 years for the mobile species (K_d = 0), Cl36, H3 and I129 and as late as 4,500 years for Pu239 which has a K_d of 270 ml/g in sandy soil (CDP = 0.51). Tc99 has a small K_d value in sandy soil (K_d = 0.6 ml/g, CDP = 1) and, as expected, behaves similar to the mobile species. Cs135, with a K_d of 50 ml/g and CDP value of 1.66, peaks at about 2,900 years, significantly later than the more mobile species.

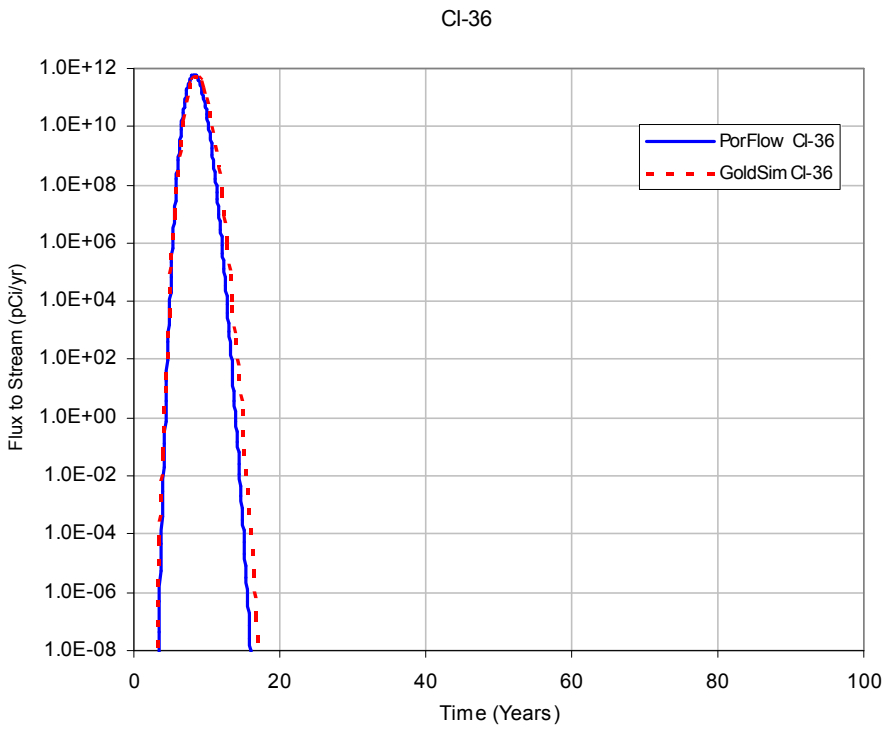


Figure 9-58. Transport of Cl36 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil with CDP Factors

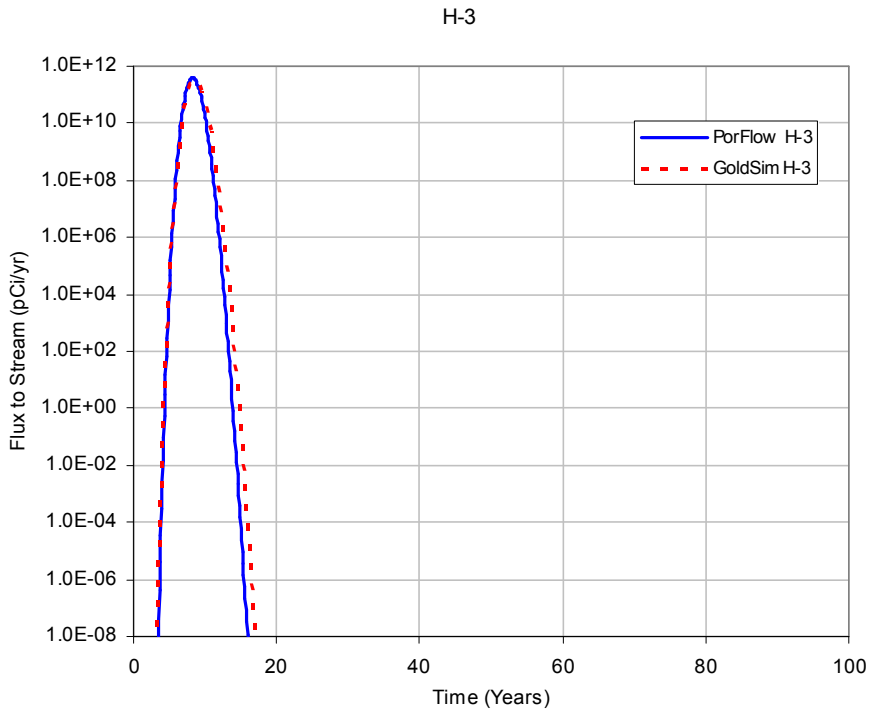


Figure 9-59. Transport of H3 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil with CDP Factors

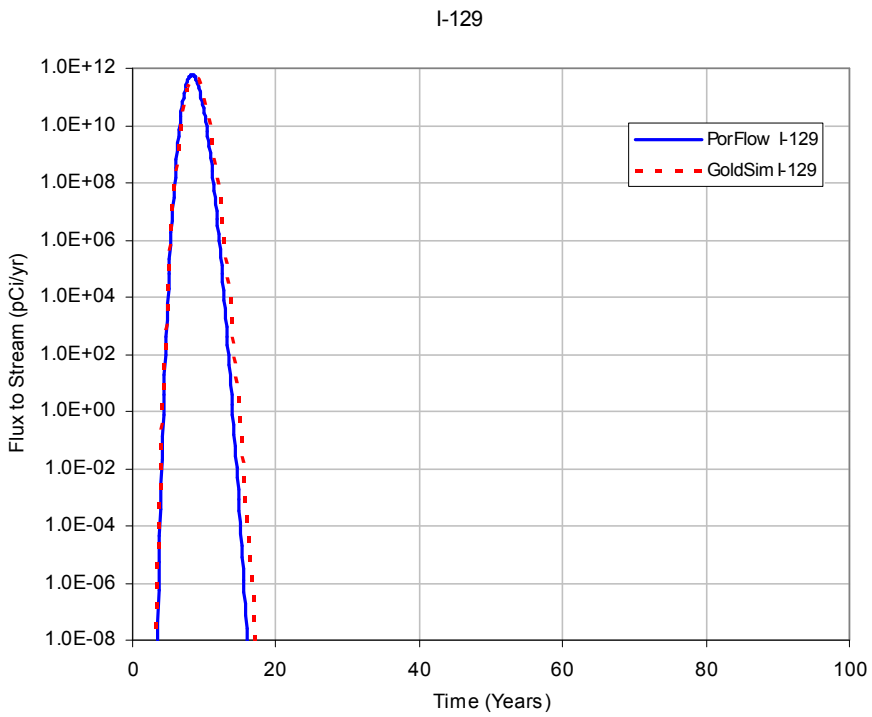


Figure 9-60. Transport of I129 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil with CDP Factors

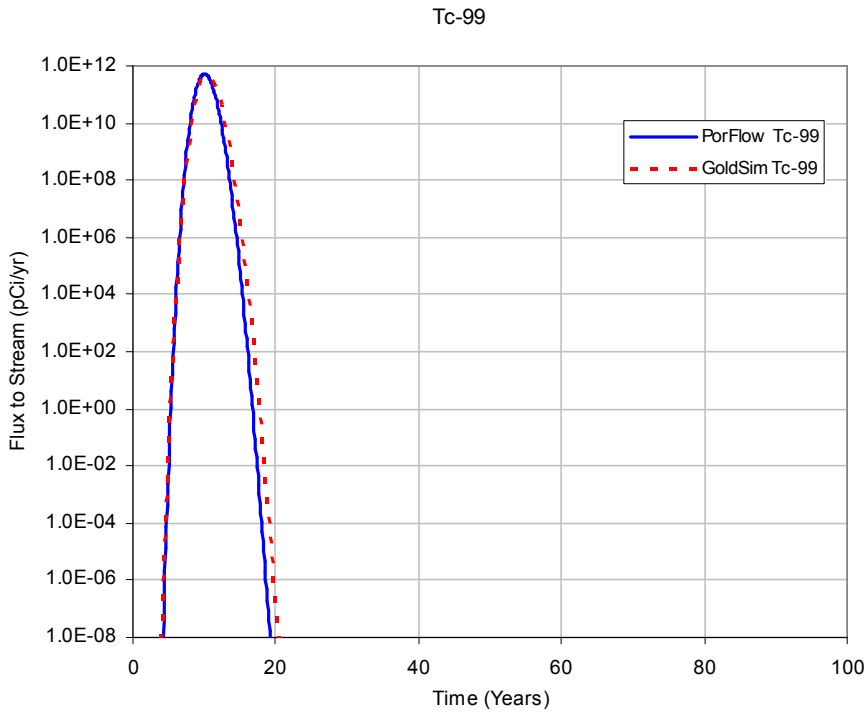


Figure 9-61. Transport of Tc99 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil with CDP Factors

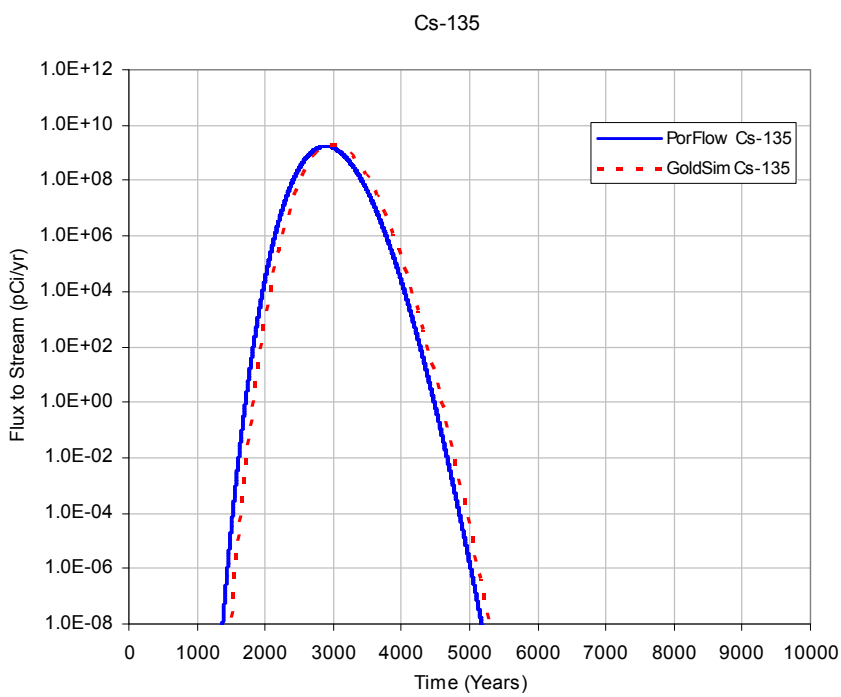


Figure 9-62. Transport of Cs135 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil with CDP Factors

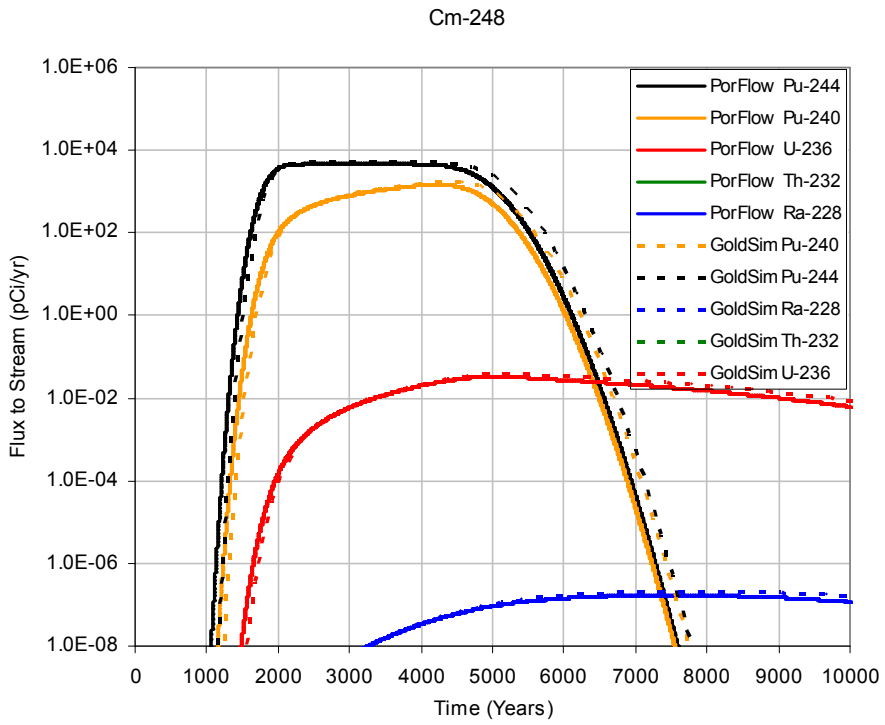


Figure 9-63. Transport of Cm248 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil with CDP Factors

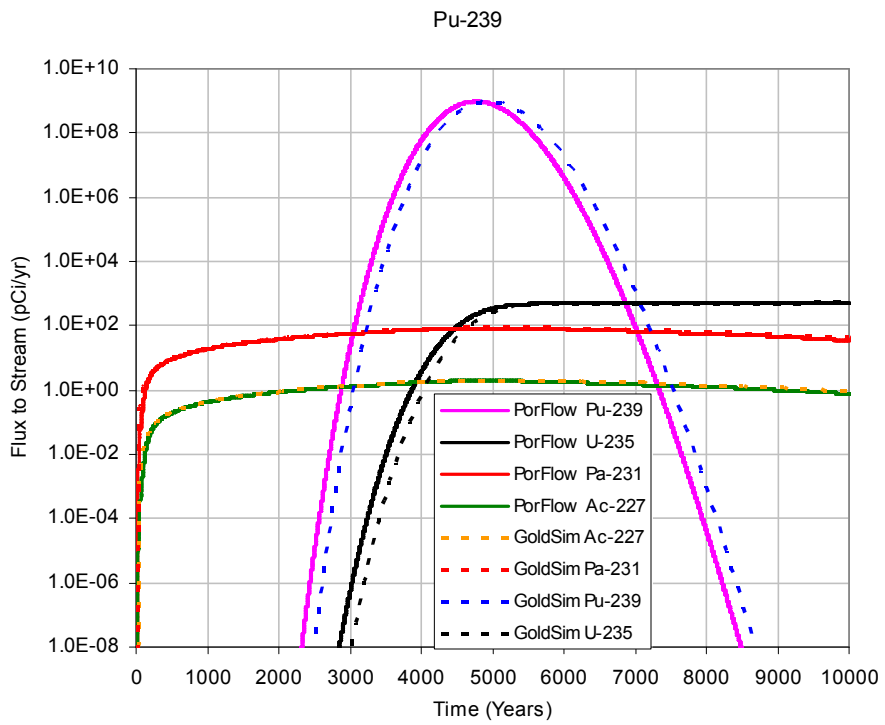


Figure 9-64. Transport of Pu239 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil with CDP Factors

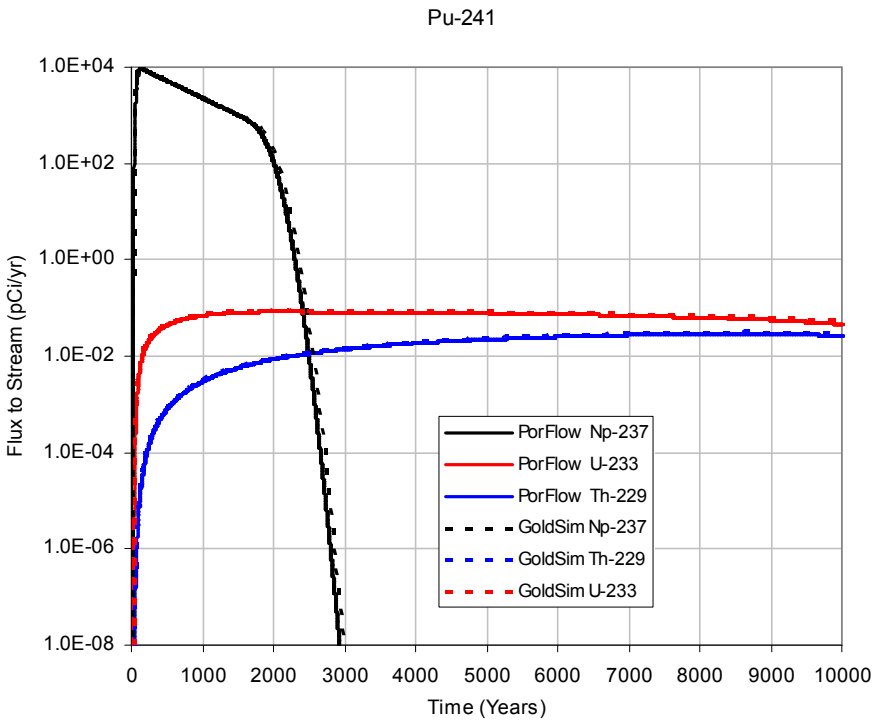


Figure 9-65. Transport of Pu241 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil with CDP Factors

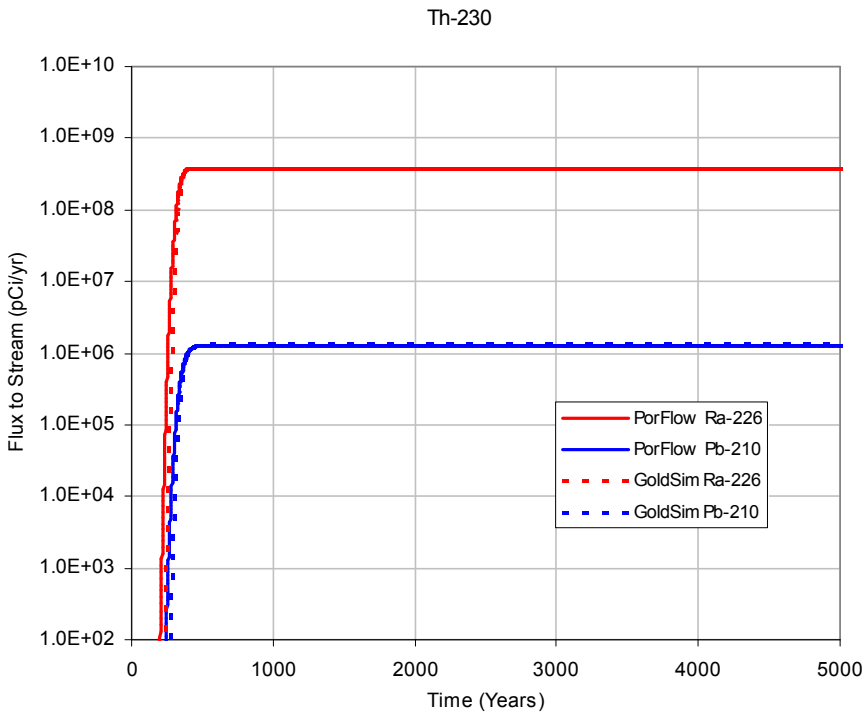


Figure 9-66. Transport of Th230 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil with CDP Factors

9.3.5.2 Model validation results for 1000 feet of saturated sandy soil without CDP factors

A comparison of flux to stream results from the PORFLOW and GoldSim™ calculations for the second test case with an aquifer path of 1,000 ft in saturated sandy soil, are shown in Figure 9-67 through Figure 9-75. These calculations are the same as those in Section 9.3.5.1 except that CDP factors were not applied. As before, there is good agreement between the GoldSim™ and PORFLOW calculations. For the mobile species Cl36, H3 and I129, CDP factors are 1.0 so there should be no difference in the transport behavior of these radionuclides with and without CDP factors applied. This is confirmed by comparing Figure 9-58 through Figure 9-60 with Figure 9-67 through Figure 9-69. Cs135 has a CDP factor of 1.66 which retards the transport rate as seen by comparing Figure 9-62 with Figure 9-71. At the other extreme, Cm248 has a CDP factor of 0.049 which greatly enhances the transport rate as seen by comparing Figure 9-63 with Figure 9-72.

Figure 9-72 and Figure 9-73 plotting fluxes from Cm248 and Pu239 decay chains, respectively, indicate a tendency for the GoldSim™ calculations to initially lag the PORFLOW results. However, peak values and peak times compare well. Figure 9-74 for the Pu241 decay chain, shows that the GoldSim™ fluxes for Pu241 decay products U233 and Th229 are uniformly higher than the PORFLOW results. Np237 is a precursor to U233 and Th229 in the Pu241 decay chain. As can be seen in Figure 9-73, the GoldSim™ calculation shows a much higher peak in Np237 flux at the start of the transient. This higher Np237 flux produces higher fluxes of the daughter radionuclides throughout the transient. In this case, it appears that the GoldSim™ simulation is capturing the expected behavior of the transport model better than the PORFLOW simulation. That is, there is no apparent physical explanation for the rapid initial increase in Np237 flux followed by a relatively slow increase to a maximum at about 800 years as shown by the PORFLOW simulation. The PORFLOW simulation probably needed to take smaller time steps early in the transient to capture the sharp peak shown by the GoldSim™ model.

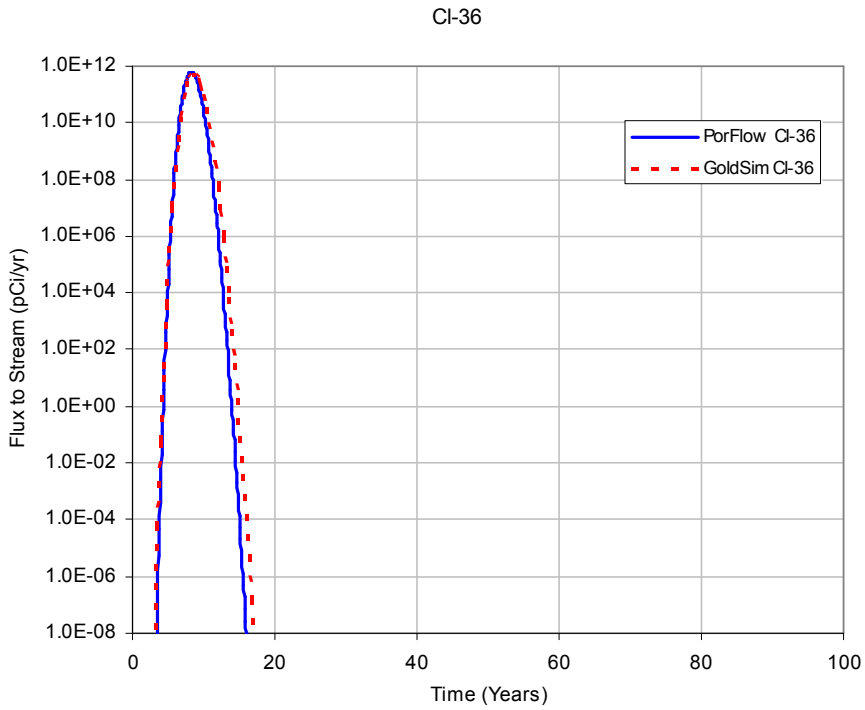


Figure 9-67. Transport of Cl36 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil without CDP Factors

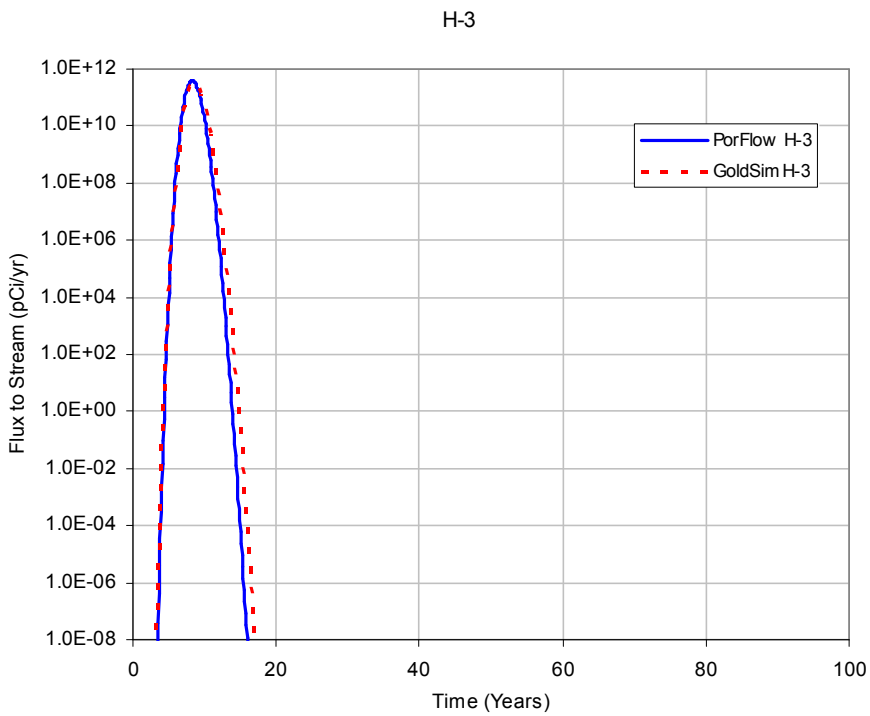


Figure 9-68. Transport of H3 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil without CDP Factors

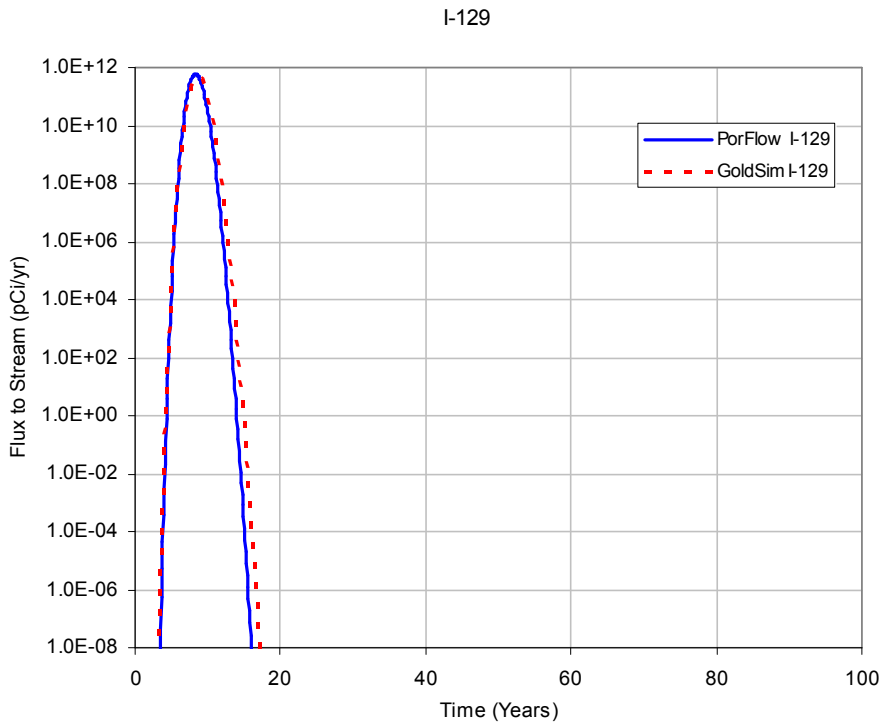


Figure 9-69. Transport of I129 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil without CDP Factors

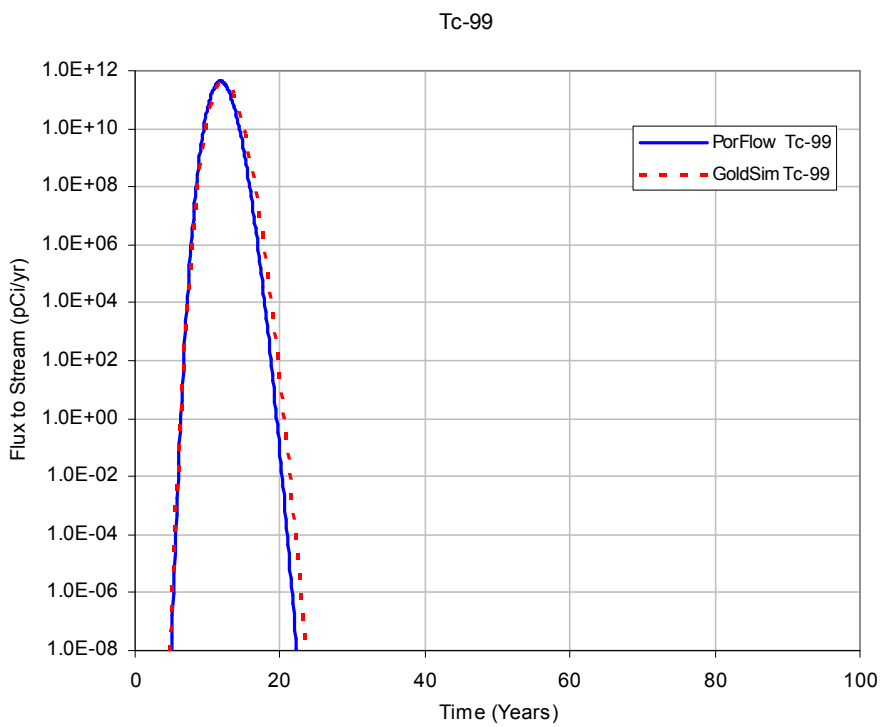


Figure 9-70. Transport of Tc99 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil without CDP Factors

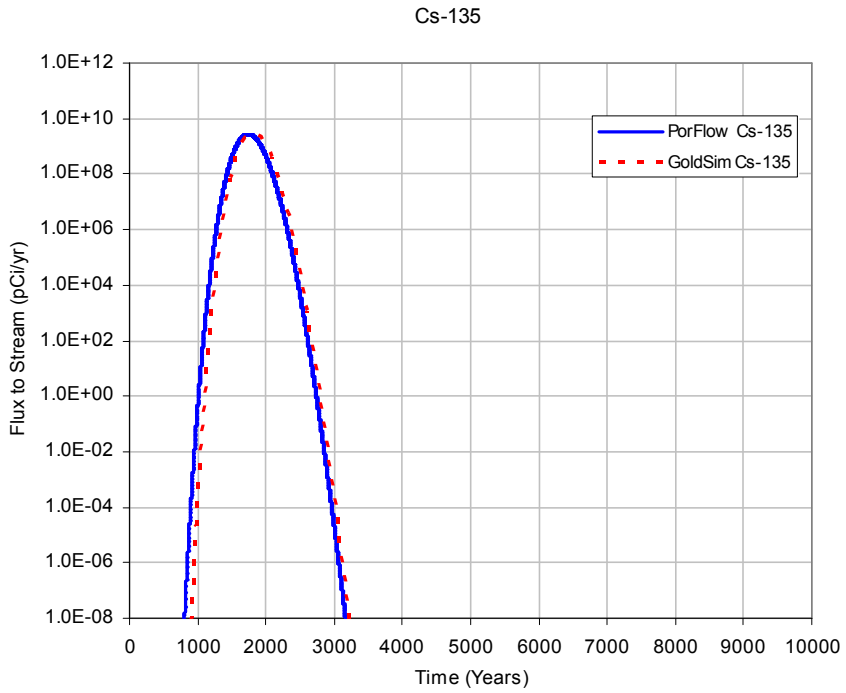


Figure 9-71. Transport of Cs135 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil without CDP Factors

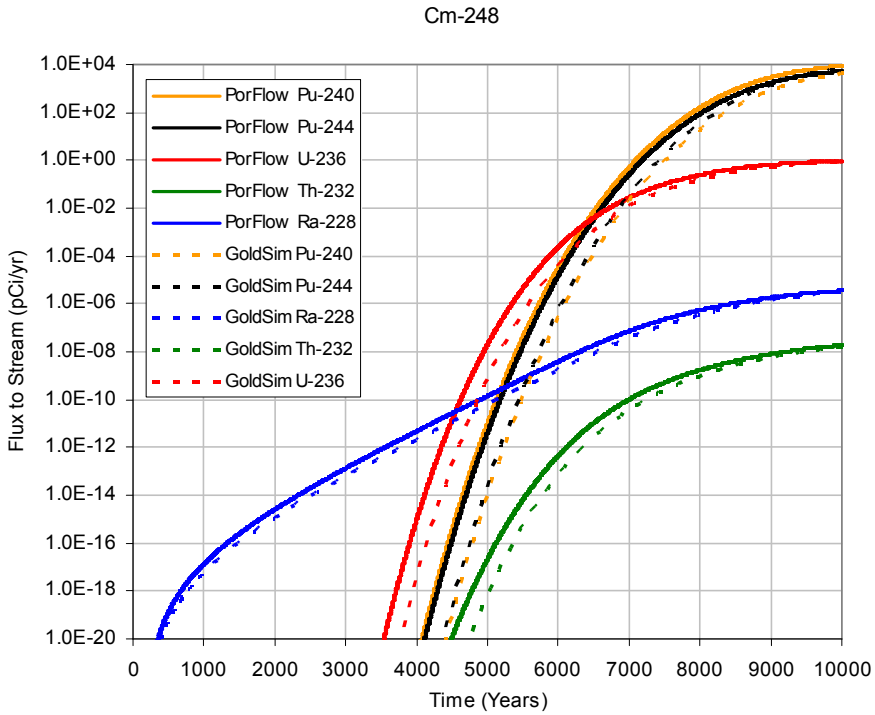


Figure 9-72. Transport of Cm248 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil without CDP Factors

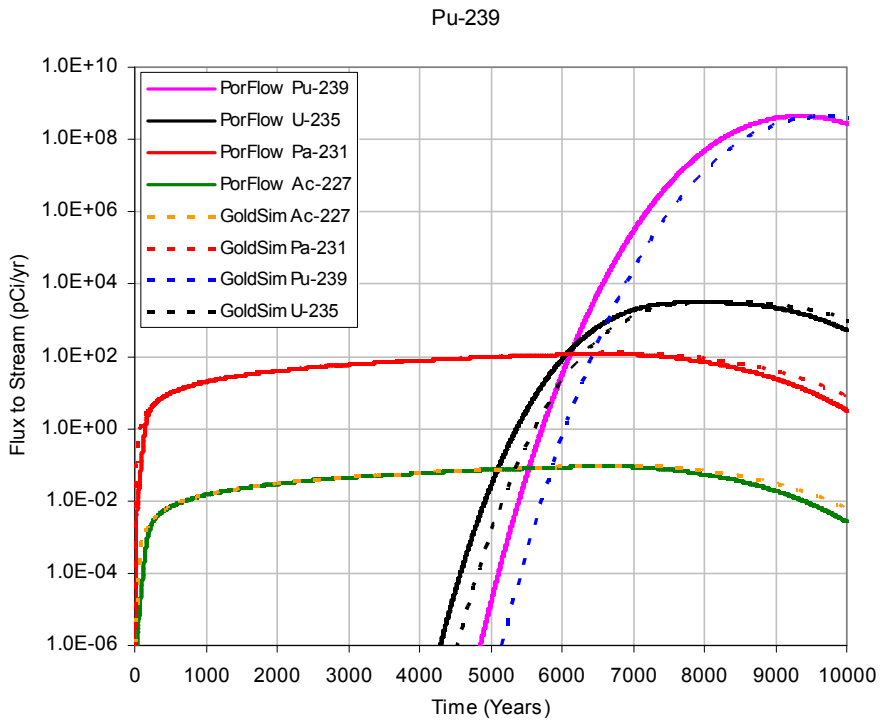


Figure 9-73. Transport of Pu239 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil without CDP Factors

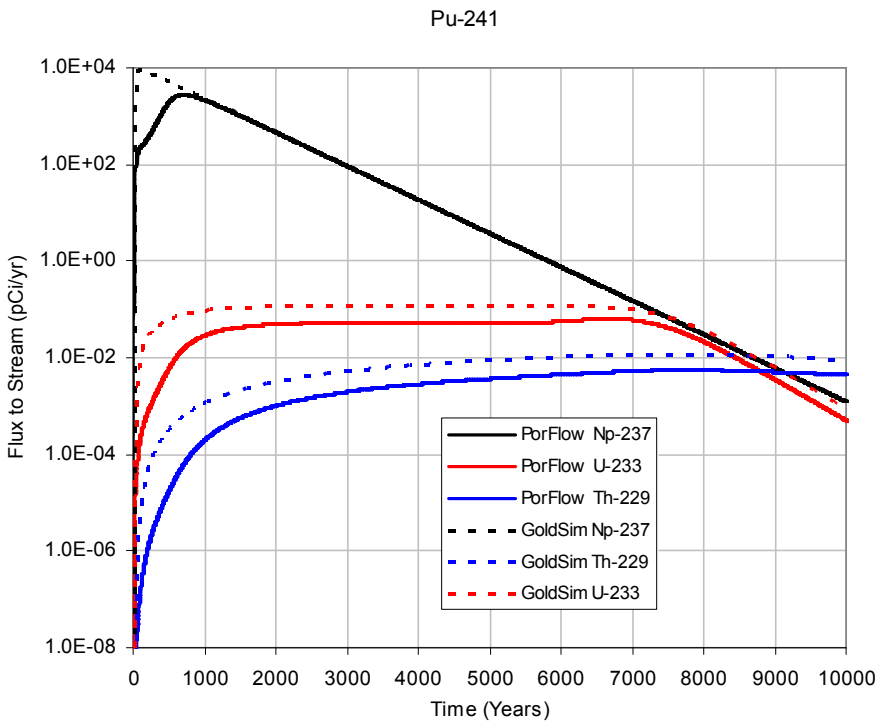


Figure 9-74. Transport of Pu241 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil without CDP Factors

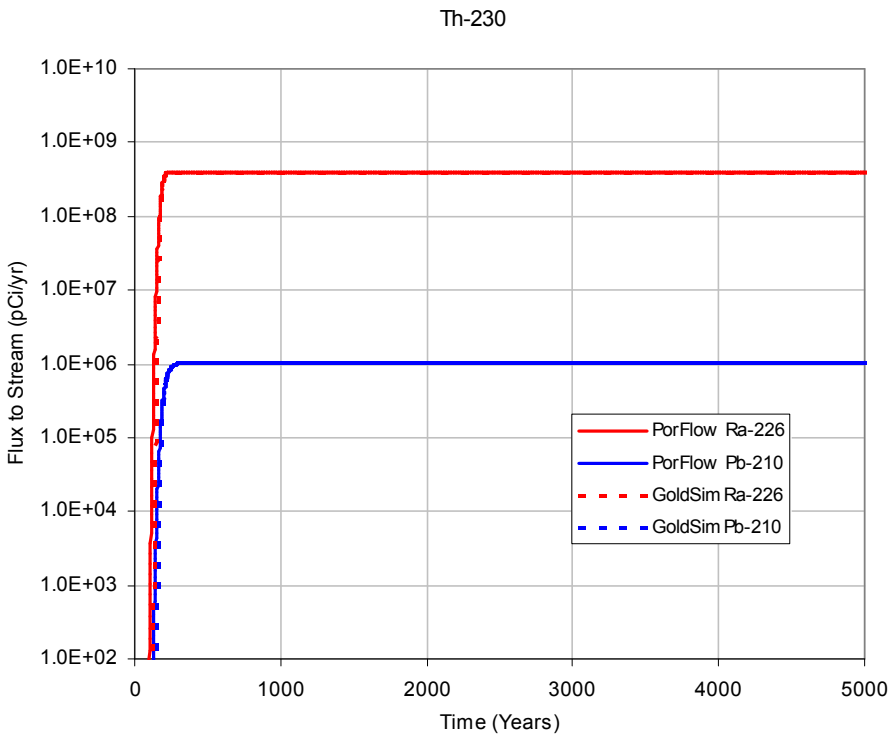


Figure 9-75. Transport of Th230 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy Soil without CDP Factors

9.3.5.3 Model validation results for 5,000 feet of saturated sandy soil without CDP factors

Flux to stream results from the validation case testing radionuclide transport through 5,000 ft of saturated sandy soil without CDP factors applied are presented in Figure 9-76 through Figure 9-83. In all cases, the comparison of the GoldSim™ results to those obtained using PORFLOW is very good. The timing and magnitude of peak radionuclide fluxes are in close agreement. In general, the results are in better agreement than those obtained for Case 2 that used a 1,000 ft transport path without the CDP factors. For this 10,000 year simulation with a 5,000 ft transport path no radionuclides from the Cm248 decay chain reached the path exit; therefore, a figure for Cm248 is not provided.

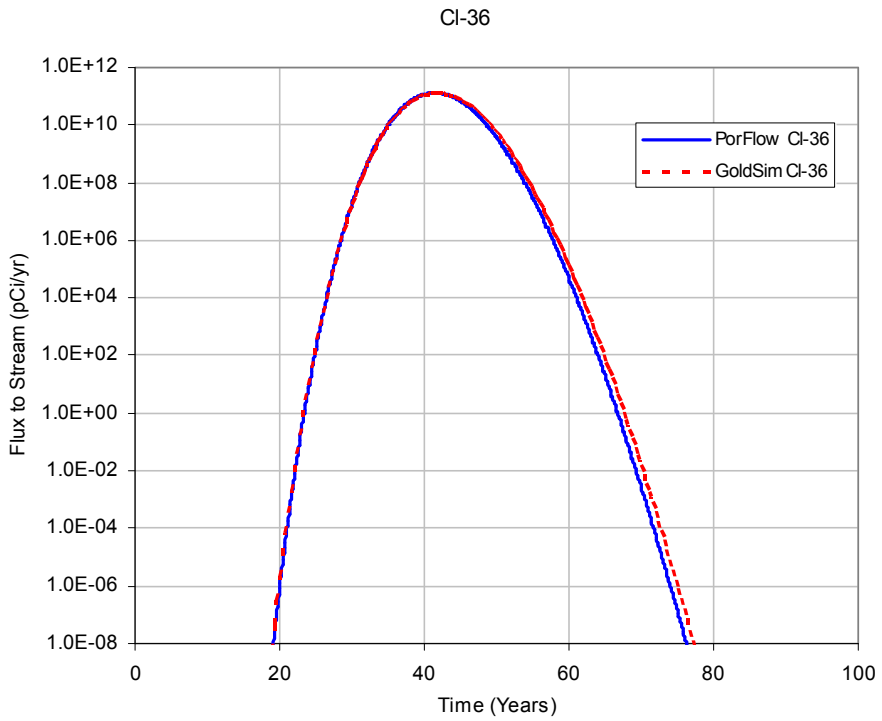


Figure 9-76. Transport of Cl36 in PORFLOW and GoldSim™ Models Through 5000 ft of Saturated Sandy Soil without CDP Factors

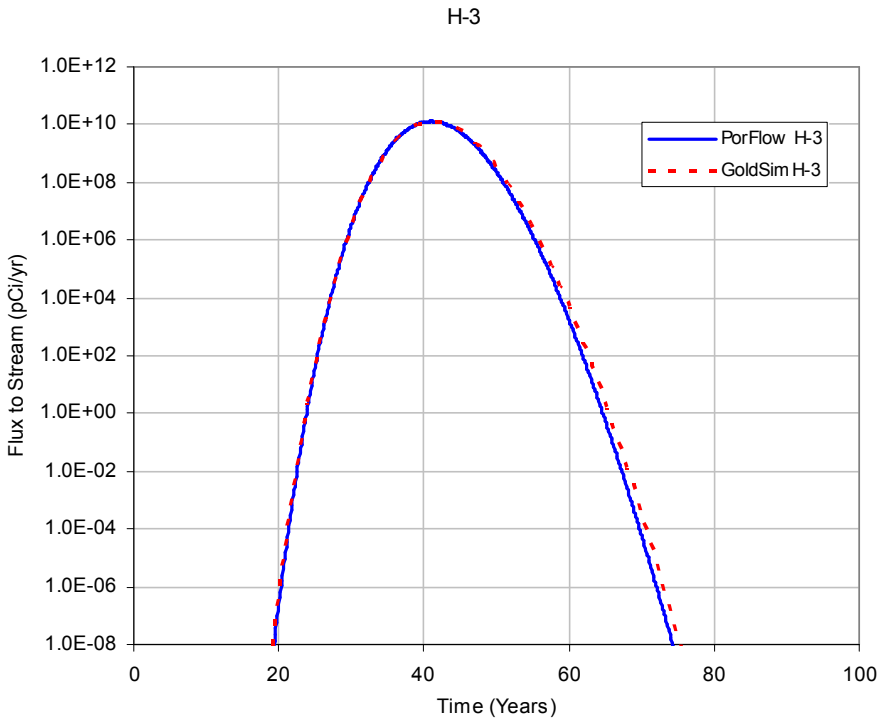


Figure 9-77. Transport of H3 in PORFLOW and GoldSim™ Models Through 5000 ft of Saturated Sandy Soil without CDP Factors

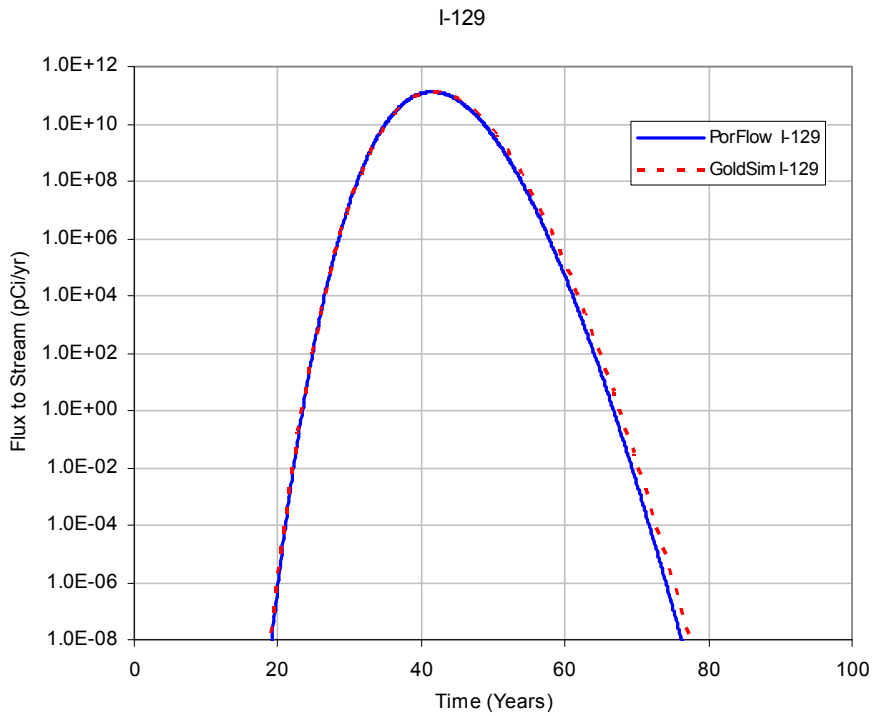


Figure 9-78. Transport of I129 in PORFLOW and GoldSim™ Models Through 5000 ft of Saturated Sandy Soil without CDP Factors

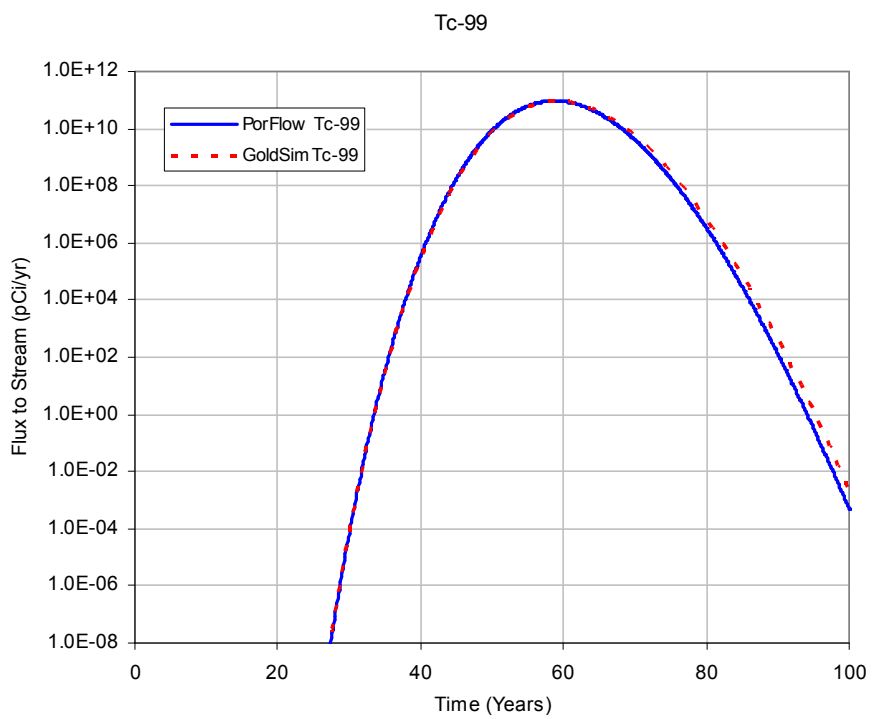


Figure 9-79. Transport of Tc99 in PORFLOW and GoldSim™ Models Through 5000 ft of Saturated Sandy Soil without CDP Factors

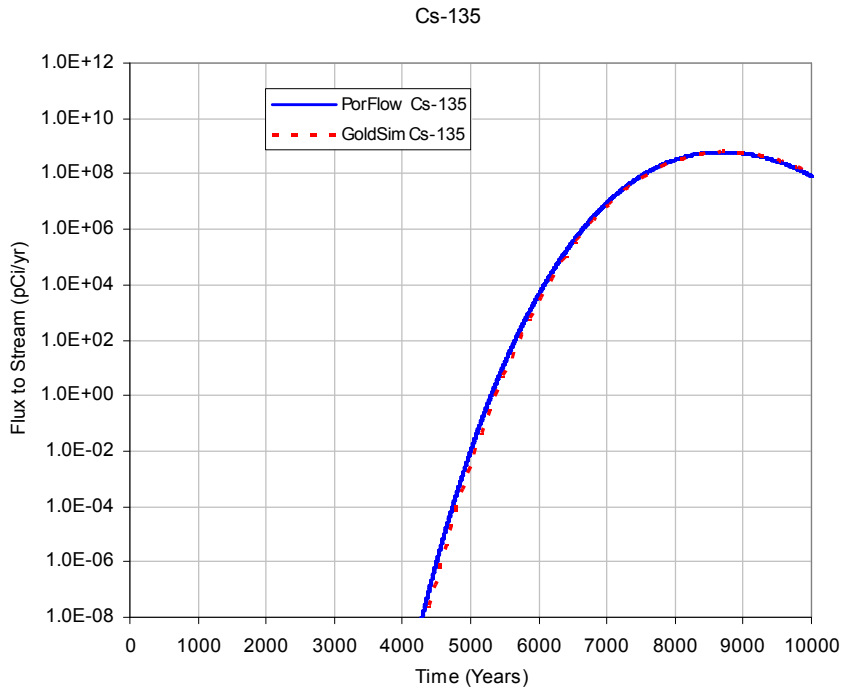


Figure 9-80. Transport of Cs135 in PORFLOW and GoldSim™ Models Through 5000 ft of Saturated Sandy Soil without CDP Factors

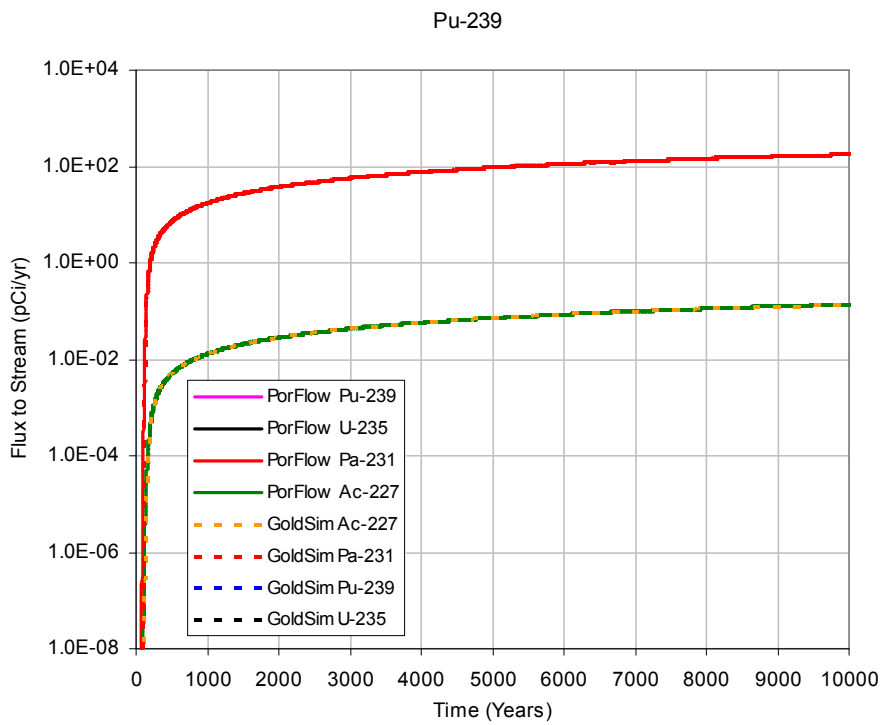


Figure 9-81. Transport of Pu239 and daughters in PORFLOW and GoldSim™ Models Through 5000 ft of Saturated Sandy Soil without CDP Factors

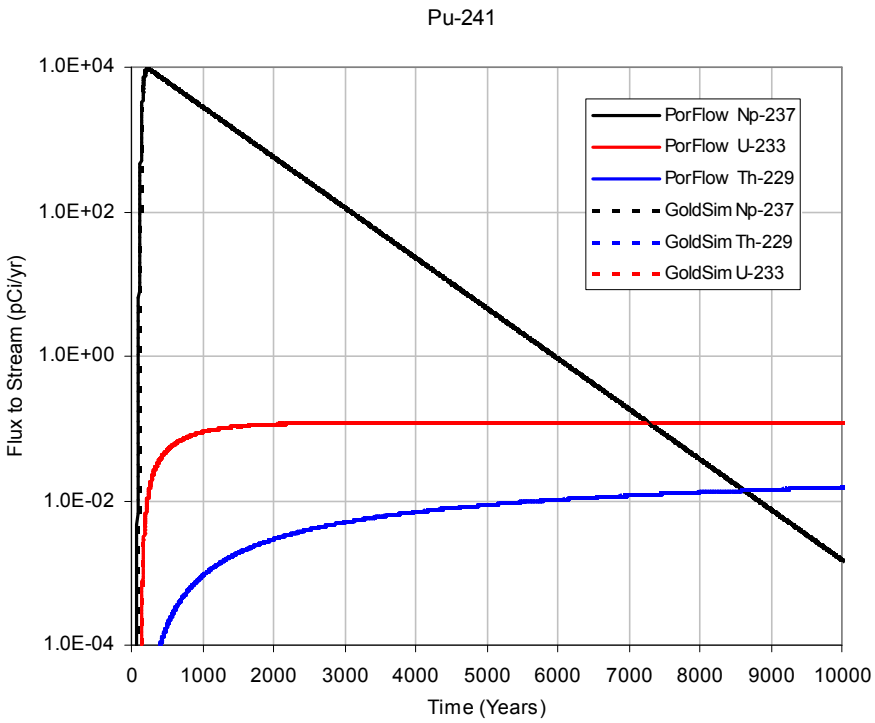


Figure 9-82. Transport of Pu241 and daughters in PORFLOW and GoldSim™ Models Through 5000 ft of Saturated Sandy Soil without CDP Factors

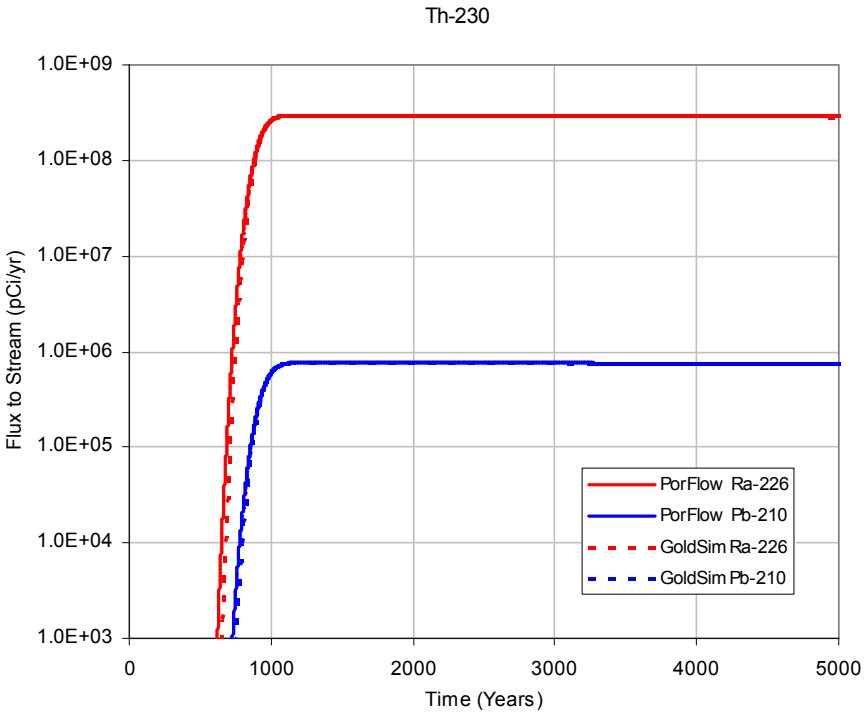


Figure 9-83. Transport of Th230 and daughters in PORFLOW and GoldSim™ Models Through 5000 ft of Saturated Sandy Soil without CDP Factors

9.3.5.4 Model validation results for 10,000 feet of saturated sandy soil without CDP factors

Flux to stream results from the validation case testing radionuclide transport through 10,000 ft of saturated sandy soil with CDF factors not applied are presented in Figure 9-84 through Figure 9-91. The results are similar to those obtained in Case 3 with the 5,000 ft transport path and, in all cases, comparisons of the GoldSim™ calculated fluxes to those obtained using PORFLOW are extremely good. The timing and magnitude of peak radionuclide fluxes are again in very close agreement. Again, a figure for Cm248 is not provided because no radionuclides from the Cm248 decay chain reached the exit path within 10,000 years.

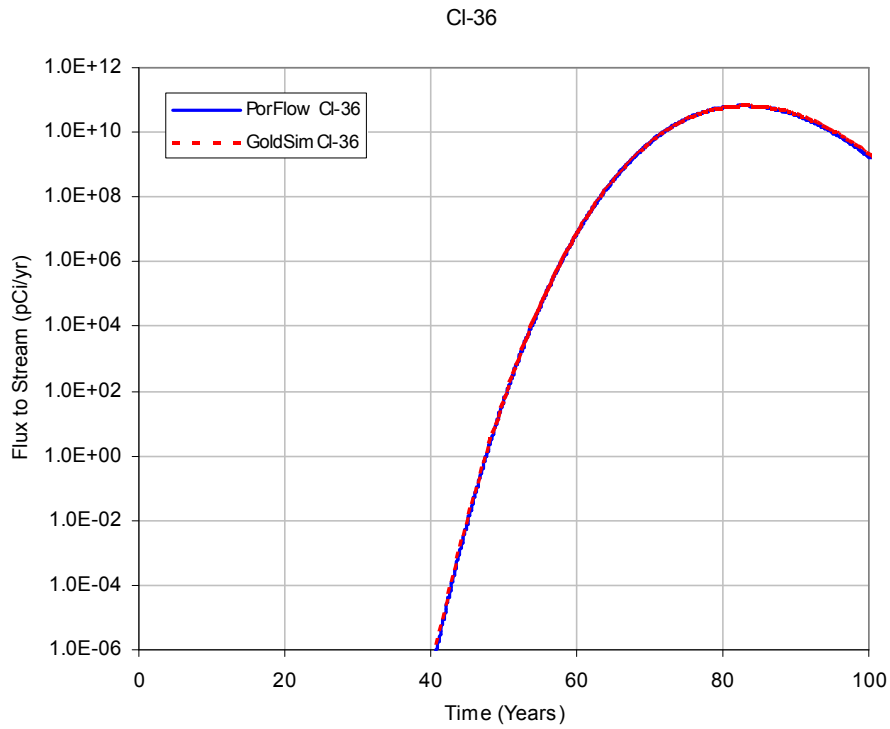


Figure 9-84. Transport of Cl36 in PORFLOW and GoldSim™ Models Through 10000 ft of Saturated Sandy Soil without CDP Factors

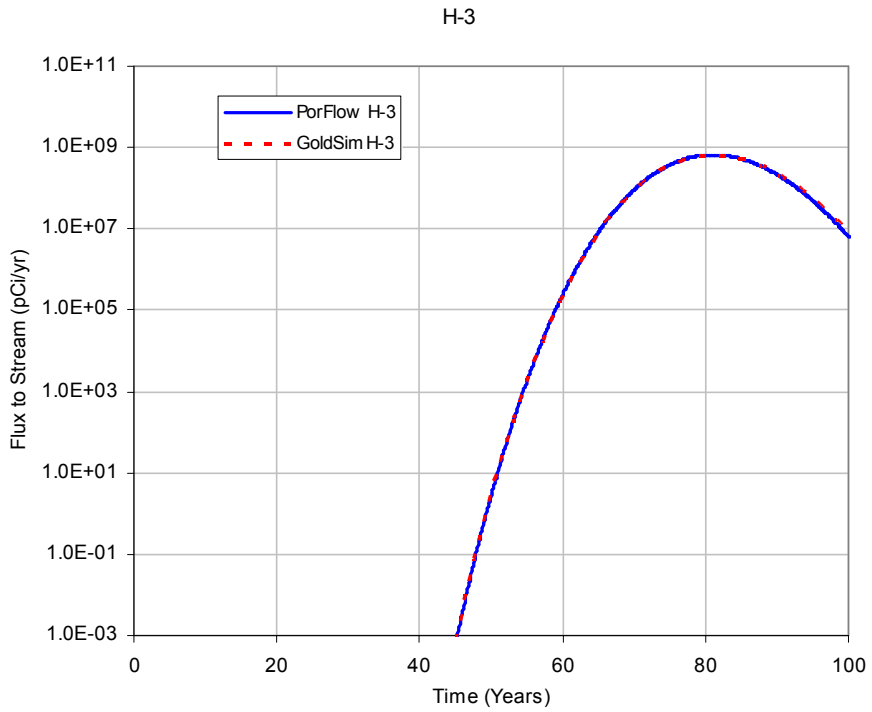


Figure 9-85. Transport of H3 in PORFLOW and GoldSim™ Models Through 10000 ft of Saturated Sandy Soil without CDP Factors

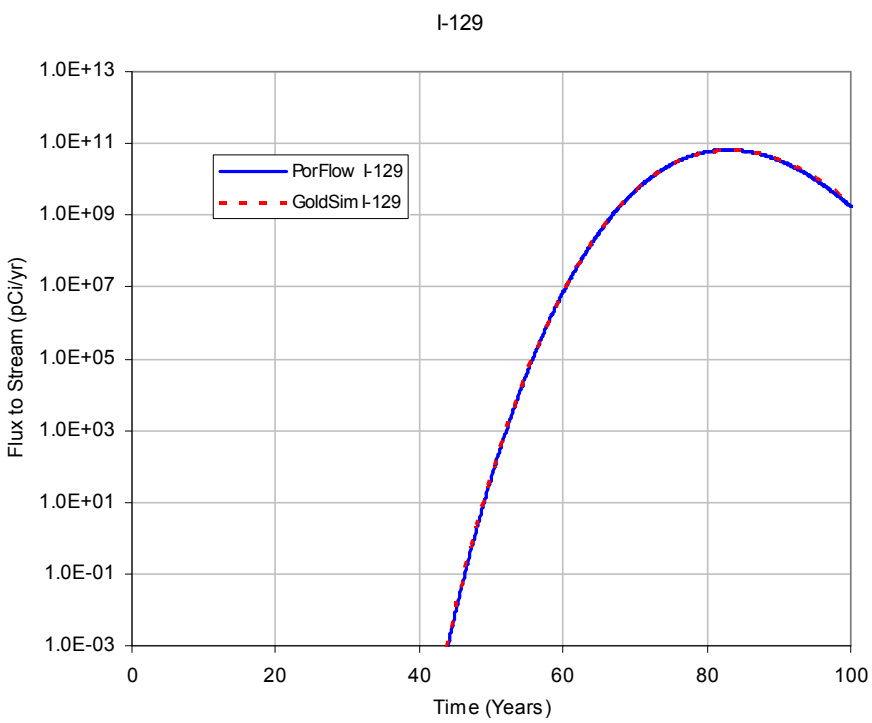


Figure 9-86. Transport of I129 in PORFLOW and GoldSim™ Models Through 10000 ft of Saturated Sandy Soil without CDP Factors

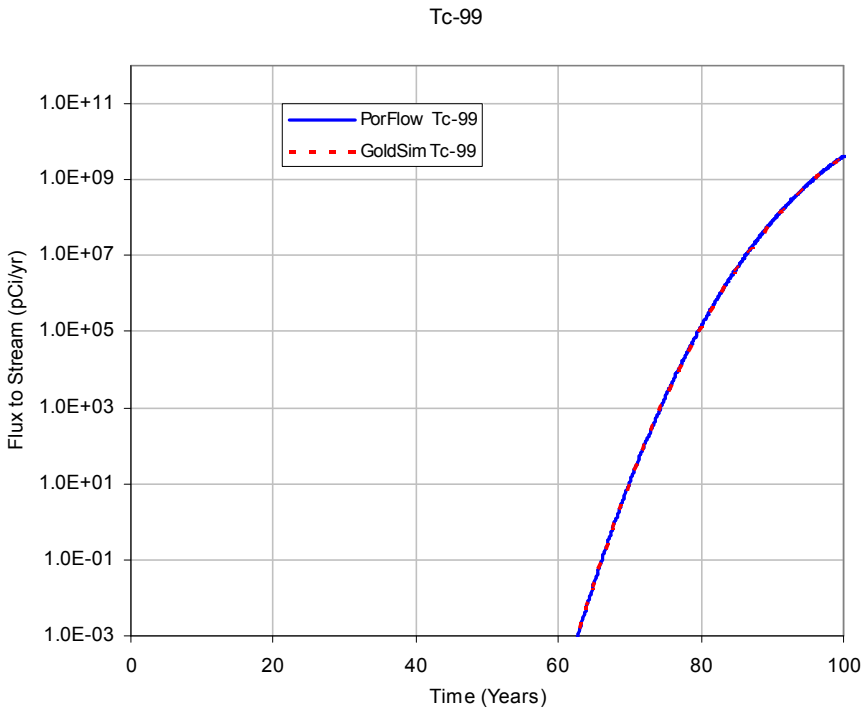


Figure 9-87. Transport of Tc99 in PORFLOW and GoldSim™ Models Through 10000 ft of Saturated Sandy Soil without CDP Factors

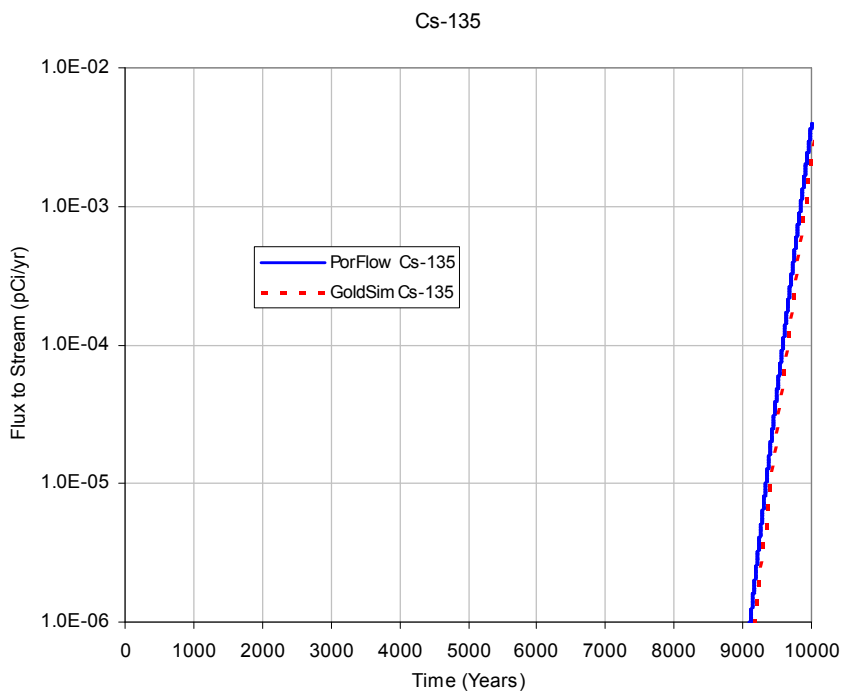


Figure 9-88. Transport of Cs135 in PORFLOW and GoldSim™ Models Through 10000 ft of Saturated Sandy Soil without CDP Factors

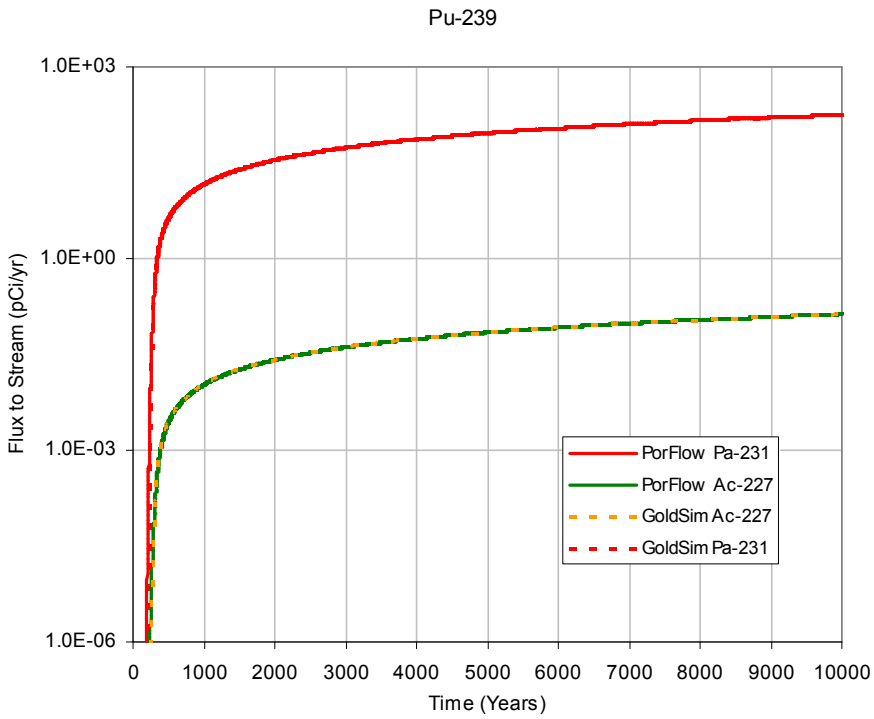


Figure 9-89. Transport of Pu239 and daughters in PORFLOW and GoldSim™ Models Through 10000 ft of Saturated Sandy Soil without CDP Factors

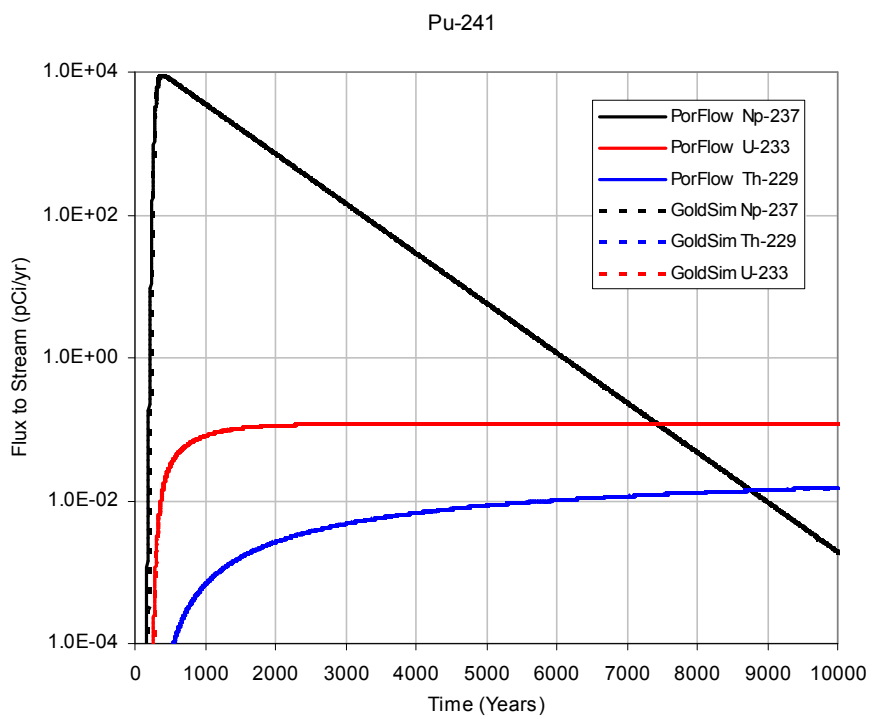


Figure 9-90 Transport of Pu241 and daughters in PORFLOW and GoldSim™ Models Through 10000 ft of Saturated Sandy Soil without CDP Factors

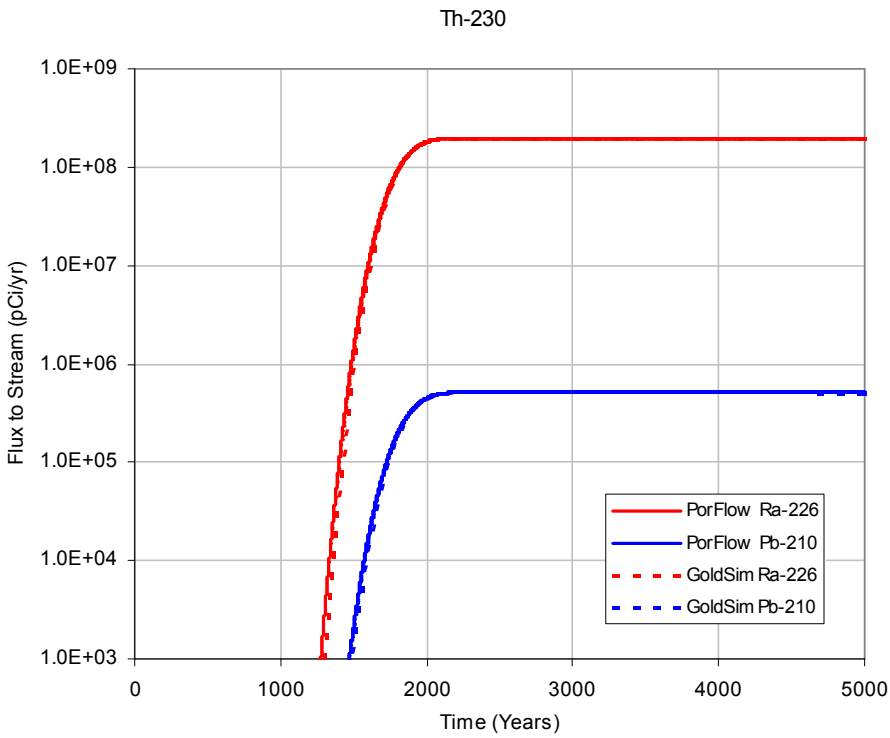


Figure 9-91. Transport of Th230 and daughters in PORFLOW and GoldSim™ Models Through 10000 ft of Saturated Sandy Soil without CDP Factors

9.3.5.5 Model validation results for 900 feet of saturated sandy soil and 100 feet of saturated clayey soil without CDP factors

Comparisons of flux to stream results from the PORFLOW and GoldSim™ calculations for the case of radionuclide transport through 900 ft of saturated sandy soil and 100 ft of saturated clayey soil without CDP factors applied are shown in Figure 9-92 through Figure 9-100. Fluxes for the mobile species Cl36 and H3, which have K_d values of zero in both sand and clay, are identical to those obtained previously for transport through a 1,000 ft aquifer (e.g., Figure 9-67 and Figure 9-68). I129 and Tc99, which have small K_d values in clay, show the expected small deviation from the previous results. Figure 9-96 shows the close agreement obtained between the two calculations for Cs135 transport.

Of the radionuclides tested, plutonium has the greatest relative difference in K_d between sandy soil (K_d = 270 ml/g) and clayey soil (K_d = 5900 ml/g). The enhanced plutonium retardation caused by replacing 100 ft of sandy soil with 100 ft of clayey soil can be seen by comparing the Pu240 and Pu244 curves in Figure 9-97 with those in Figure 9-72 and the Pu239 curve in Figure 9-98 with that in Figure 9-73.

As in Case 2, Figure 9-97 and Figure 9-98, which plot fluxes to the stream from Cm248 and Pu239 decay chains, respectively, show a tendency for the GoldSim™ calculations to initially lag the PORFLOW results. However, peak values and times when the peak fluxes are reached compare well between the corresponding calculations. Figure 9-99 shows less agreement for the transport calculation of Pu241 decay products. GoldSim™ calculated fluxes for Pu241 decay products U233 and Th229 are uniformly higher than the PORFLOW results. As explained in the discussion of Case 2, it is likely that the GoldSim™ simulation is capturing the transport behavior more accurately for this case.

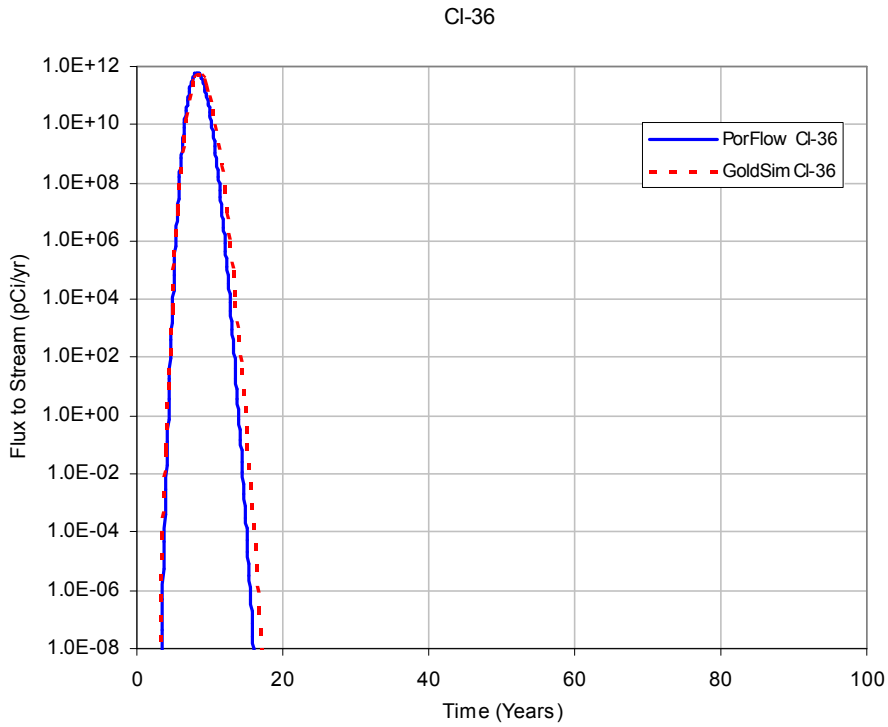


Figure 9-92. Transport of Cl36 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy and Clayey Soil without CDP Factors

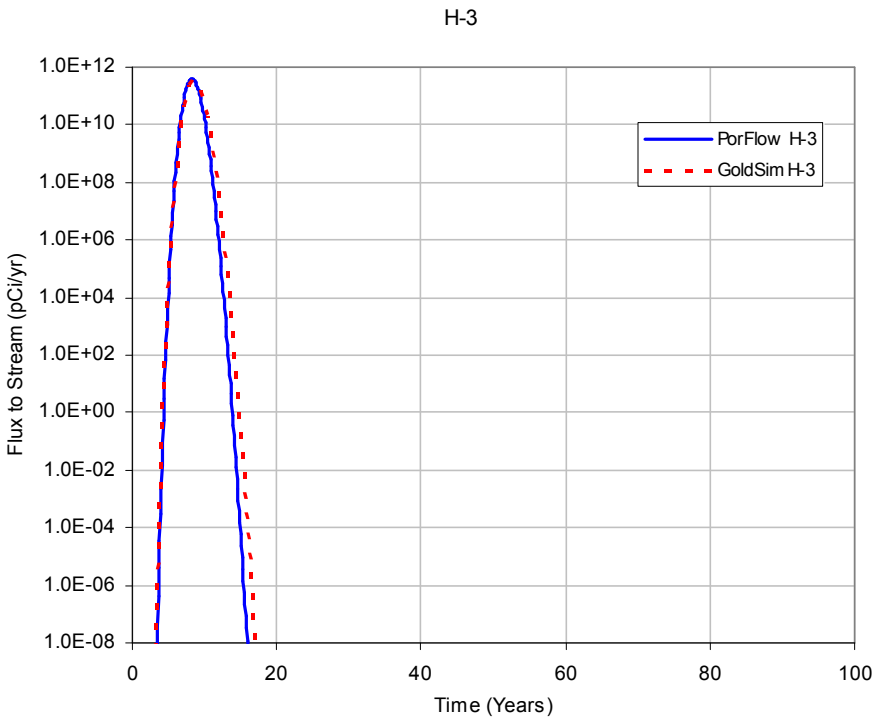


Figure 9-93. Transport of H3 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy and Clayey Soil without CDP Factors

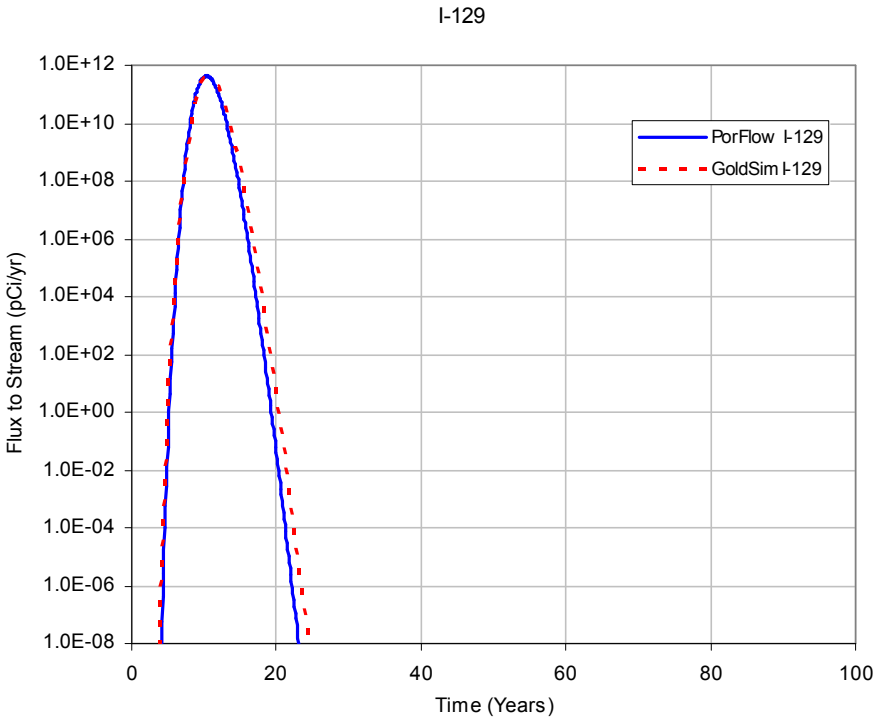


Figure 9-94. Transport of I129 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy and Clayey Soil without CDP Factors

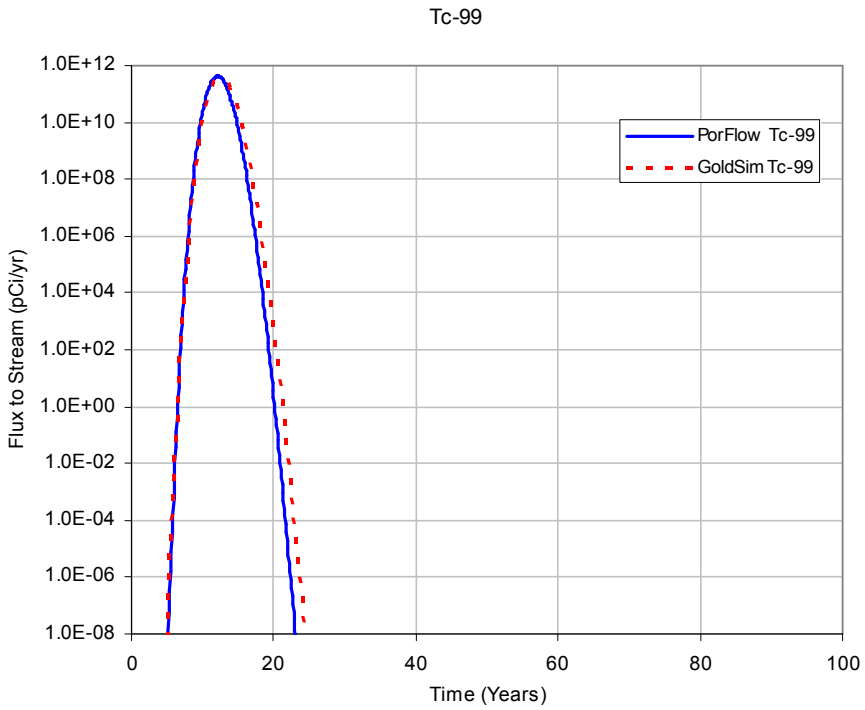


Figure 9-95. Transport of Tc99 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy and Clayey Soil without CDP Factors

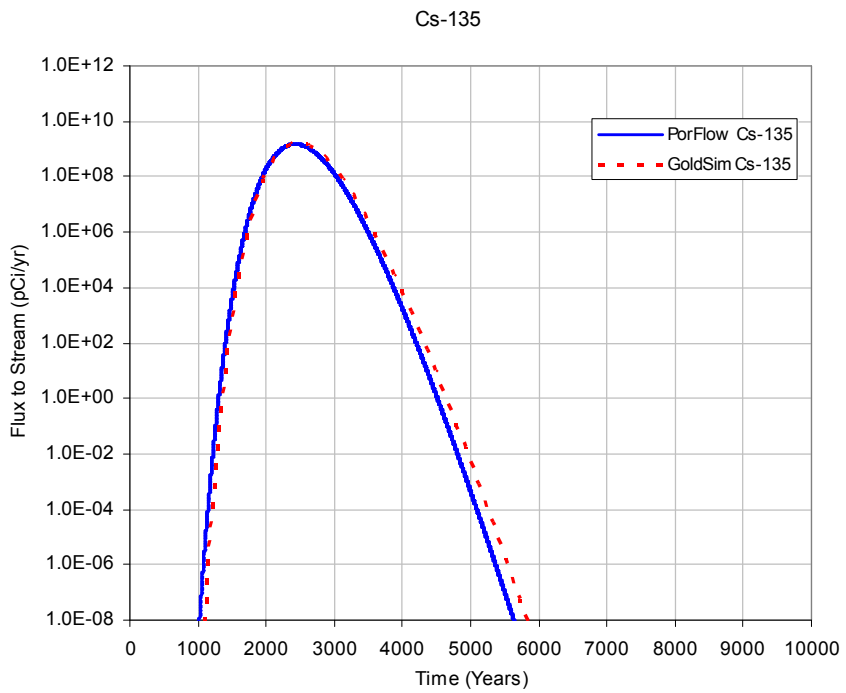


Figure 9-96. Transport of Cs135 in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy and Clayey Soil without CDP Factors

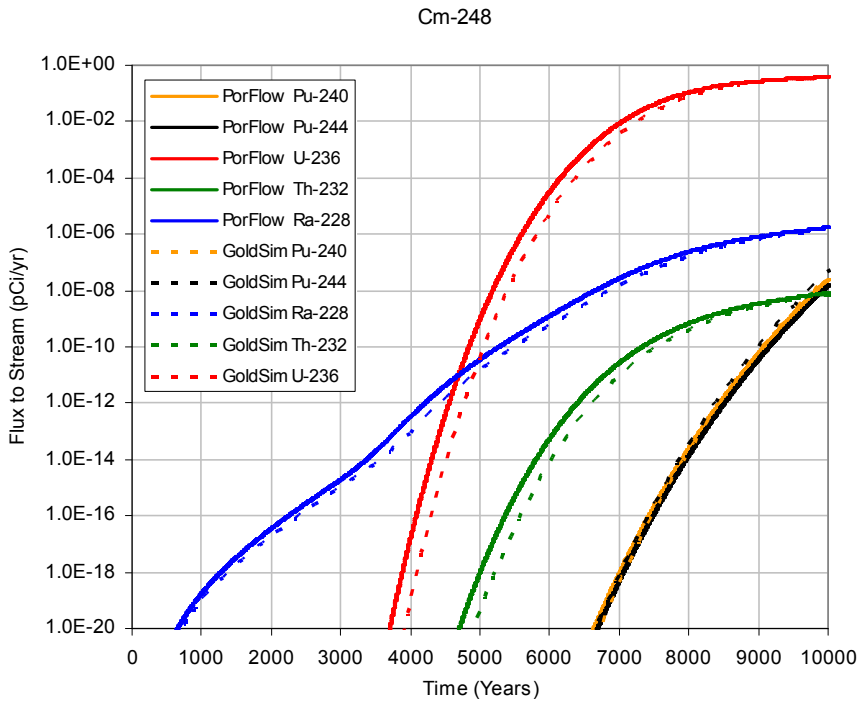


Figure 9-97. Transport of Cm248 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy and Clayey Soil without CDP Factors

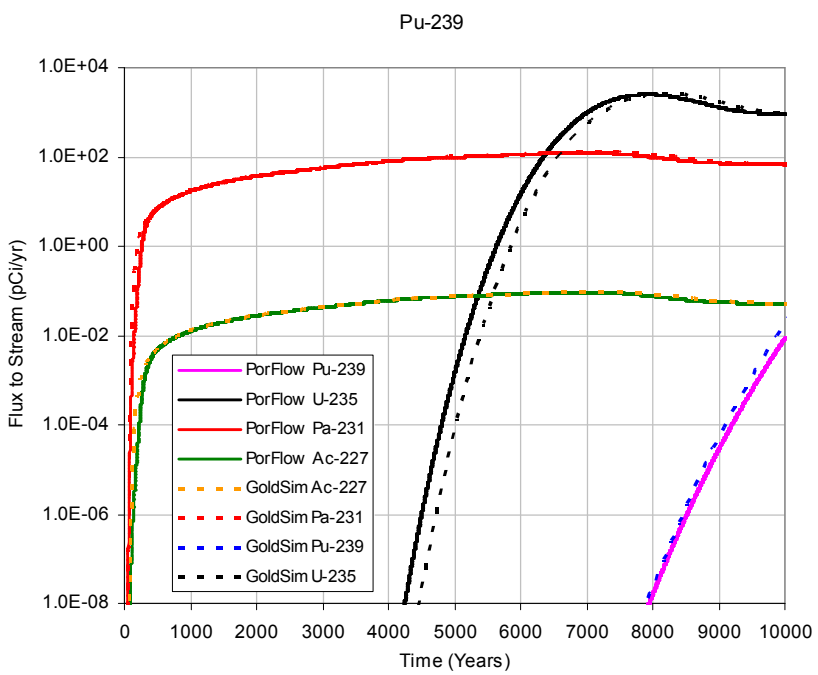


Figure 9-98. Transport of Pu239 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy and Clayey Soil without CDP Factors

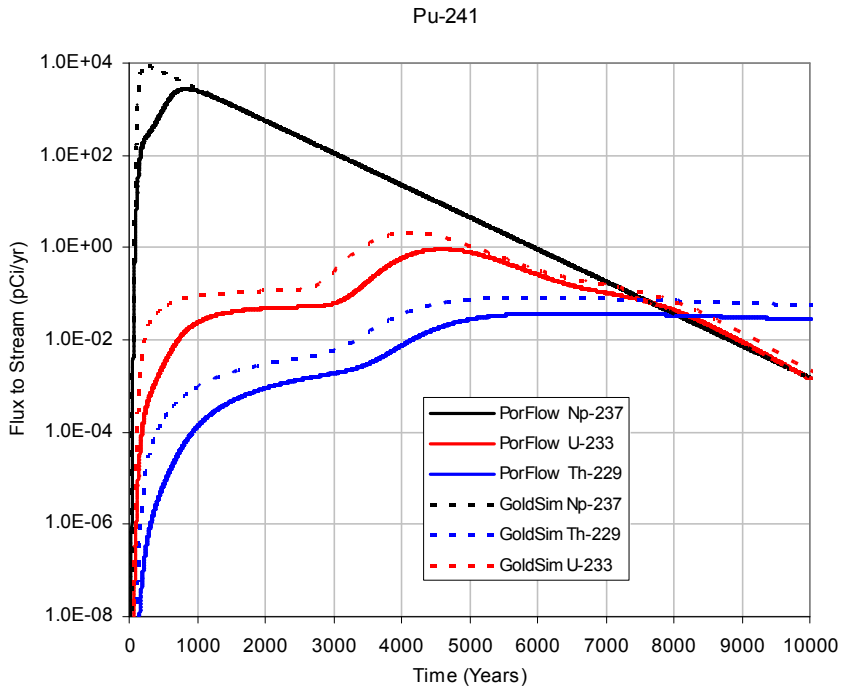


Figure 9-99. Transport of Pu241 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy and Clayey Soil without CDP Factors

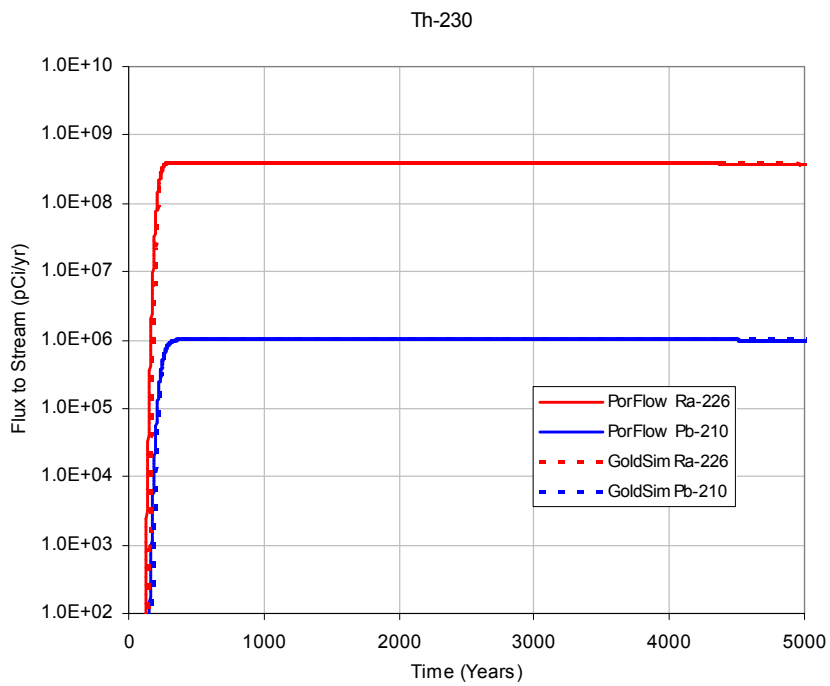


Figure 9-100. Transport of Th230 and daughters in PORFLOW and GoldSim™ Models Through 1000 ft of Saturated Sandy and Clayey Soil without CDP Factors

9.3.6 Generic Release Results

Results from the generic release modeling are summarized in Table 9-19 and Table 9-20. In this scenario, the source inventory is released at the top of the vadose zone, transported through the vadose zone by natural infiltration (15 in/yr), and transported through the aquifer to the stream and river where doses are calculated. Table 9-19 and Table 9-20 show the maximum total dose calculated during the 1,000 year CA assessment period starting at the site end-state date of 2025 for each of the 137 surface sources (i.e., does not include the 15 stream bed and tritium plume subsurface sources [see Figure 9-2]). Maximum doses greater than 1.0 mrem/yr are highlighted in orange and maximum doses greater than 0.1 mrem/yr are highlighted in yellow. Seven maximum doses from the generic release analysis were greater than 1.0 mrem/yr and another 17 maximum doses were greater than 0.1 mrem/yr. The sum of all of the maximum doses is 37.7 mrem/yr. This a conservative estimate of total maximum dose that does not account for the fact that not all of the maximum doses occur simultaneously and does not account for the distribution of dose to different points of assessment. Nonetheless, the total maximum dose is still below the Department of Energy (DOE) TEDE limit of 100 mrem/yr and only 25% greater than the dose constraint of 30 mrem/yr. The POA identification numbers in Table 9-19 and Table 9-20 are the same as those used in the CA model shown in Table 9-7 (Section 9.3.1.5). A lower limit of 1.0E-99 mrem/yr was placed on the dose values to avoid problems plotting the data on a logarithmic scale which cannot accept values of zero.

The 24 surface sources (highlighted in yellow and orange in Table 9-19 and Table 9-20) contributing doses greater than 0.1 mrem/yr during the CA assessment period were selected for more detailed source release modeling that would account more accurately for actual radionuclide release mechanisms. An additional six surface sources (highlighted in cyan in Table 9-19 and Table 9-20) were included in the source release modeling for consistency. Looking at the broader time period from 1950 to 12,050, which covers the time from the start of site operations out to 10,025 years beyond the site end-state date, the only additional maximum dose greater than 0.1 mrem/yr was from the HSB which gave a relatively high maximum dose of 0.96 mrem/yr prior to 2025. Therefore, the HSB was also included in the source release modeling. The FSB was then included so both similar sources were treated alike. The 105-R reactor vessel activated stainless steel was also included in the source release analysis because all of the other reactor vessel stainless steel sources were modeled. The ORWBG was included along with the special waste form I129 in the ORWBG that is in the form of silver iodide. The two paths from the LLRWDF were included in the source release modeling because the various parts of the LLRWDF are positioned in close proximity to the MWMF.

The 105-P surface contamination in Table 9-20 was removed from source release modeling because the contamination is spread throughout the building and may be released by different mechanisms in different parts of the building. Addressing surface contamination in the reactor buildings through modeling was deemed not warranted because only one of the five sources was just slightly above the cutoff criterion of 0.1 mrem/yr. Starting with the original

24 surface sources considered for source release modeling, adding the six other sources, and deleting one, left 29 surface sources that received additional modeling (see Figure 9-2). Table 9-21 lists the units analyzed using source release modeling, not including the 15 streambed and tritium plume subsurface sources, which also require source release modeling.

The grouping and numbering of sources in Table 9-20 and Table 9-21 follow the order:

A-Area	1 - 4	Tritium Area	74 - 82	P-Area	99 - 104
C-Area	5 - 10	J-Area	83	R-Area	105 - 113
D-Area	11 - 15	K-Area	84 - 89	S-Area	114 - 117
E-Area	16 - 36	L-Area	90 - 95	T-Area	118 - 122
F-Area	37 - 52	M-Area	96 - 97	Z-Area	123 - 127
H-Area	53 - 73	N-Area	98	MOXFF	128
				Spills	129 - 137

Table 9-19. Maximum Doses Calculated by Generic Release Modeling for Surface Sources 1 – 76

Number	POA	Unit Name	Max Dose mrem/yr	Number	POA	Unit Name	Max Dose mrem/yr
1	1	A-Area Seep	3.00E-99	39	3	F-Area IPSL FMB	1.94E-05
2	5	773-A	1.26E-12	40	1	221-F	2.75E-03
3	1	776-A	3.00E-67	41	1	211-F	6.15E-05
4	5	778-A	5.37E-82	42	1	281-3F	8.34E-03
5	3	105-C SS	1.55E-01	43	3	F-Area Seep	1.01E-02
6	3	105-C Al	9.25E-14	44	1	FTF Type I	2.16E-06
7	3	105-C Con	1.91E-04	45	1	FTF Type IV	1.27E-07
8	3	105-C Surf	5.75E-02	46	1	FTF Type III	1.43E-03
9	3	C-Area Seep	1.39E-20	47	1	FTF Type IIIA UTR	3.07E-06
10	3	C-Area Dirt	1.71E-23	48	3	FTF Type IIIA FMB	1.03E-05
11	5	420-D	6.10E-15	49	1	FTF Equip	1.08E-05
12	5	420-2D	2.07E-08	50	1	772-F	1.82E-06
13	5	421-D	4.89E-24	51	1	772-1F,-4F	1.87E-06
14	5	421-2D	6.80E-14	52	1	F-Area Old Seep	1.16E-02
15	5	772-D	4.15E-09	53	1	HTF Type I UTR E	1.07E-04
16	1	E-Area CIG	2.09E-02	54	1	HTF Type I UTR W	8.44E-06
17	1	E-Area ET	2.70E-04	55	1	HTF Type II	9.81E-05
18	1	E-Area ILV	1.03E+01	56	1	HTF Type IV UTR	1.49E-06
19	1	E-Area LAWV	1.30E+01	57	3	HTF Type IV FMB	1.53E-04
20	1	NR Pad 1 Part 1	1.93E+00	58	1	HTF Type III UTR E	1.29E-05
21	1	NR Pad 1 Part 2	4.83E-01	59	1	HTF Type III UTR W	4.36E-07
22	1	NR Pad 1 Part 3	4.65E-01	60	3	HTF Type III FMB	2.49E-03
23	1	NR Pad 1 Part 4	1.07E-01	61	1	HTF Type IIIA UTR E	2.67E-04
24	1	NR Pad 1 Part 5	3.88E-01	62	1	HTF Type IIIA UTR W	4.43E-06
25	1	NR Pad 1 Part 6	4.75E-01	63	3	HTF Type IIIA FMB	3.89E-03
26	1	NR Pad 1 Crud	5.90E-04	64	1	HTF Equip UTR	4.55E-06
27	1	NR Pad 2	3.65E-02	65	3	HTF Equip FMB	5.88E-04
28	1	E-Area ST East	3.23E-04	66	1	H-Area IPSL UTR	3.69E-03
29	1	E-Area ST Cent	1.33E-02	67	3	H-Area IPSL FMB	4.36E-03
30	1	E-Area ST West	1.36E-04	68	1	281-1H, 2H	8.17E-07
31	1	TRU Pad	1.19E-12	69	1	281-8H	6.07E-04
32	3	LLRWDF FMB	3.90E-02	70	3	H-Area Seep	7.21E-02
33	1	LLRWDF UTR	3.94E-02	71	1	H-Canyon	1.25E+00
34	1	MWMF	1.42E-01	72	1	RBOF	5.72E-08
35	3	ORWBG	1.10E-02	73	1	H-Area Sand	1.20E-08
36	3	ORWBG AgI	1.17E-01	74	1	232-H	1.17E-01
37	1	F-Area MS	1.67E-06	75	1	HANM	2.90E+00
38	1	F-Area IPSL UTR	4.89E-04	76	1	HAOM	1.15E-01

Table 9-20. Maximum Doses Calculated by Generic Release Modeling for Surface Sources 77 – 137

Number	POA	Unit Name	Max Dose mrem/yr
77	1	234-7H	4.21E-03
78	1	236-H	7.13E-04
79	1	237-H	5.33E-05
80	1	TEF	3.63E-01
81	1	264-2H	1.77E+00
82	1	TPBAR	2.44E-02
83	1	J-Area SWPF	1.34E-05
84	4	KAC SS	1.52E-01
85	4	KAC Al	4.46E-11
86	4	KAC Con	2.17E-03
87	4	KAC Surf	5.30E-02
88	4	K-Area Seep	7.11E-05
89	4	K-Area Dirt	5.02E-17
90	4	LAC SS	2.84E-01
91	4	LAC Al	3.61E-07
92	4	LAC Con	8.24E-03
93	4	LAC Surf	7.57E-02
94	4	L-Area Seep	2.17E-04
95	4	L-Area Dirt	2.04E-06
96	1	M-Area HWMF	5.25E-10
97	1	M-Area MIPSL	3.99E-12
98	1	N-Area Seep	1.00E-99
99	4	105-P SS	3.03E-01
100	4	105-P Al	6.46E-07
101	4	105-P Con	1.54E-02
102	4	105-P Surf	1.14E-01
103	4	P-Area Seep	2.80E-06
104	4	P-Area Dirt	6.61E-04
105	2	105-R SS	4.38E-02
106	2	105-R Al	9.33E-10
107	2	105-R Con L	4.15E-07
108	2	105-R Con S	3.26E-03
109	2	105-R Surf L	4.10E-04
110	2	105-R Surf S	4.68E-02
111	2	R-Area Seep	3.84E-05
112	2	R-Area Dirt L	1.46E-48
113	2	R-Area Dirt S	1.31E-06
114	1	221-S	7.11E-04
115	1	511-S LPPP	3.76E-04
116	1	294-S Sand L	9.83E-07
117	1	294-S Sand S	3.43E-06
118	5	TNX BG	7.78E-06
119	5	T-Area IPSL	3.90E-06
120	5	TNX Gully	4.43E-06
121	5	TNX Outfall	4.61E-04
122	5	677-T	2.76E-07
123	1	Z-Area Vault 1	7.71E-03
124	1	Z-Area Vault 4	4.93E-02
125	1	Z-Area Vault 2 NL	6.98E-01
126	1	Z-Area Vault 2 NS	2.89E-01
127	1	Z-Area Vault 2 S	1.08E+00
128	1	MOXFF	3.62E-05
129	1	T-13 Spill	1.89E-05
130	1	T-9 Spill	6.00E-06
131	1	T-16 Spill	4.73E-05
132	1	T-37 Spill UTR	2.57E-05
133	3	T-37 Spill FMB	8.20E-04
134	1	281-3F Spill	1.83E-06
135	1	T-3 Spill	1.61E-06
136	1	T-8 Spill	1.78E-04
137	1	281-3H Spill	8.97E-06

Table 9-21. Sources Selected for Source Release Modeling

Sources Giving Maximum Doses > 0.1 mrem/yr with Generic Release Modeling	Other Sources Included	Sources Removed
E-Area ILV		
E-Area LAWV		
E-Area MWMF		
	LLRWDF FMB	
	LLRWDF UTR	
E-Area ORWBG I129		
	ORWBG	
Z-Area Vault 2 North Long Path		
Z-Area Vault 2 North Short Path		
Z-Area Vault 2 South Path		
H-Canyon		
NRCDA 643-7E Metal Part 1		
NRCDA 643-7E Metal Part 2		
NRCDA 643-7E Metal Part 3		
NRCDA 643-7E Metal Part 4		
NRCDA 643-7E Metal Part 5		
NRCDA 643-7E Metal Part 6		
C Reactor Stainless Steel		
K Reactor Stainless Steel		
L Reactor Stainless Steel		
P Reactor Stainless Steel		
	R-Reactor Stainless Steel	
105-P Surface Contamination		105-P Surface Contamination
232-H		
H-Area Old Manufacturing Facility (HAOM)		
H-Area New Manufacturing Facility (HANM)		
264-2H		
Tritium Extraction Facility (TEF)		
	H-Area Seepage Basin	
	F-Area Seepage Basin	
Total Sources = 24	+ 6 = 30	– 1 = 29

9.3.7 Source Release Results

Based on the results of the generic modeling presented in Section 9.3.6, the 44 sources (i.e., 29 surface sources and 15 streambed and tritium plume subsurface sources) indicated in Table 9-22 were analyzed using more detailed source release modeling (see Figure 9-2). Table 9-22 lists these sources and indicates the source release mechanisms (described in Section 9.3.1.6) applied to each source. Table 9-23 gives a comparison of the maximum doses obtained from the generic release calculation and the corresponding source release calculation for the 29 surface sources. In some cases, Table 9-23 sums maximum doses from individual sources to arrive at an estimate of the total dose for a facility. This approach is conservative because maximum doses do not necessarily overlap. Table 9-23 also provides the maximum source release dose for the 15 subsurface sources (i.e., streambed and tritium plume sources) for which no generic release calculation is available.

Table 9-22. Use of Source Release Modeling in CA Base Case Analysis

		Source Release Mechanism										
Source Unit	**Number of Sources**	**Clay Cap (3.2.6.2)**	**Concrete Cap (3.2.6.2)**	**Geo-synthetic Cap (3.2.6.2)**	**Metal Cask (3.2.6.2.2)**	**AgI Solubility (3.2.6.5)**	**Diffusion (3.2.6.3)**	**Metal Corrosion (3.2.6.4)**	**Concrete Barrier Kd (3.5.3 and 3.6.2)**	**PA Flux (3.2.6.6)**	**Tritium Plumes (3.2.6.7)**	**Streambed Sediment (3.2.6.8)**
E-Area ILV	1									✓		
E-Area LAWV	1									✓		
E-Area LLRWDF	2			✓								
E-Area MWMF	1	✓										
E-Area ORWBG	2			✓		✓						
Z-Area Vault 2	3									✓		
F-Area Seepage Basin	1	✓										
H-Area Seepage Basin	1	✓										
NRCDA 7E Metal	6				✓			✓				
Reactor Vessel Stainless Steel (C, K, L, P, R)	5		✓					✓	✓			
H-Area Canyon	1		✓									
Tritium Area Buildings: 232-H, HANM, HAOM, 264-2H, TEF	5						✓					
GOUs: FMB, D-Area, K-Area, L-Area, P-Area, R-Area, T-Area	7										✓	
IOUs: FMB, LTR, PB, SR-A, SR-B, SC, UTR, SpdTbr	8											✓

Table 9-23. Comparison of Generic Release and Source Release Doses

	Source	Predominant Isotopes	Maximum Generic Dose 2025 to 3025 mrem/yr	Maximum Source Release Dose 2025 to 3025 mrem/yr
1	E-Area ILV	H3, Nb94	10.30	<0.001
2	E-Area LAWV	H3	12.98	<0.001
3	E-Area LLRWDF/MWMF[1]	I129	0.22	0.171
4	E-Area ORWBG[2]	I129	0.13	0.170
5	Z-Area Vault 2[3]	Tc99, I129	2.07	<0.001
6	F-Area Seepage Basin	- -	0.01	0.002
7	H-Area Seepage Basin	- -	0.07	0.103
8	NRCDA 643-7E[4]	Nb93m	3.85	0.006
9	C-Reactor Building (105-C)[5]	Nb94	0.21	0.058
10	K-Area Complex (KAC)[5]	Nb94	0.21	0.055
11	L-Area Complex (LAC)[5]	Nb94	0.37	0.084
12	P-Reactor Building (105-P)[5]	Nb94	0.43	0.130
13	R-Reactor Building (105-R)[5]	Nb94	0.09	0.050
14	H-Area Canyon/HB Line	Np237	1.25	1.044
15	Building 232-H	H3	0.12	0.011
16	H-Area New Manufacturing Facility (HANM)	H3	2.90	0.061
17	H-Area Old Manufacturing Facility (HAOM)	H3	0.11	0.020
18	TEF	H3	0.36	0.003
19	Building 264-2H	H3	1.77	0.015
20	Fourmile Branch GOU	- -	- -	0.007
21	D-Area GOU	- -	- -	< 0.001
22	K-Area GOU	- -	- -	0.002
23	L-Area GOU	- -	- -	< 0.001
24	P-Area GOU	- -	- -	< 0.001
25	R-Area GOU	- -	- -	< 0.001
26	T-Area GOU	- -	- -	< 0.001
27	Fourmile Branch IOU	Cs137	- -	1.970
28	Lower Three Runs IOU	Cs137	- -	2.940
29	Pen Branch IOU	- -	- -	0.060
30	Savannah River A IOU	- -	- -	0.018
31	Savannah River B IOU	- -	- -	0.014
32	Steel Creek IOU	Cs137	- -	0.243
33	Upper Three Runs IOU	- -	- -	0.072
34	Steed Pond/Tims Branch IOU	- -	- -	0.002
	Total		**37.45**	**7.31**

[1] Sum of maximum doses from MWMF and both segments of LLRWDF irrespective of POA
[2] Sum of maximum doses from ORWBG including I129 on berl saddles, for both generic and source release models, the dose contribution from sources other than I129 was about 0.01 mrem/yr
[3] Sum of maximum doses from all three segments of Z-Area Vault 2
[4] Sum of maximum doses from all six NRCDA 643-7E metal parts and "crud"
[5] Sum of maximum doses from all parts of reactor building (stainless steel, aluminum, concrete, surface contamination and dirt)

For the ORWBG and HSB, the maximum dose after source release modeling was slightly greater than the generic release dose. In both cases, the maximum dose was from I129 which peaked prior to 2025 when the CA period of assessment began. Placing a cap on these sources delayed the peak time and caused a higher dose to occur within the CA assessment period.

9.3.7.1 Surface Sources

The MWMF/LLRWDF was closed in two segments. A 58-acre portion, designated the MWMF, received waste from 1972 to 1986 and was closed with a clay cap in 1990 (WSRC 1991). An additional 18 acres designated as the LLRWDF were used from 1986 to 1995 and closed with a geosynthetic cap in 1998 (WSRC 1999). Therefore infiltration associated with a clay cap and geosynthetic cap were implemented for the 58-acre MWMF and 18-acre LLRWDF, respectively, as outlined in Section 9.3.1.6.7.2, with the Cap Placement Years set to 1990 and 1998 for the MWMF and LLRWDF, respectively. Inventory Years for the MWMF and LLRWDF were set to average placement dates of 1979 and 1990, respectively. The MWMF, closed using a clay cap, was predicted to go to UTR. However, because of the presence of the ground water divide under the facility, a 1.9-acre portion of the LLRWDF was predicted to go to FMB. Because the inventory distribution in the MWMF and the LLRWDF is unknown, the inventory was assumed to be distributed uniformly over the entire facility. As with the ORWBG, discussed below, there was some indication that part of the I129 in the MWMF/LLRWDF is in the form of silver nitrate coated berl saddles. However, no information on the distribution of the I129 between berl saddles and other waste forms was found. Therefore, all I129 was conservatively assumed to be in other waste forms.

For source release modeling of the ORWBG, the Cap Placement Year was set to 2007, when the geosynthetic closure cap (GCC) was completed (WSRC 2002), and infiltration associated with a geosynthetic cap as outlined in Section 9.3.1.6.7.2 was implemented. Waste in the ORWBG was emplaced between 1952 and 1972 (WSRC 1997b). So the Inventory Year was set to the average date of 1962 to best reflect the true waste emplacement time. The primary contributor (by about two orders of magnitude) to the ORWBG dose with generic source release was I129. The vast majority of this I129 was captured on silver nitrate impregnated berl saddles (WSRC 1997b) and would slowly be released by dissolving in water flowing past the waste. It was estimated (WSRC 1997b) that I129 not associated with berl saddles made up 0.25% of the total I129 inventory in the ORWBG. For purposes of the CA, a conservative value of 1% was used for I129 not associated with berl saddles. Because 99% of the I129 is handled as solubility controlled release while everything else is K_d controlled, the inventory for the ORWBG was broken into two parts. One of these parts (ORWBG AgI) contained only the 99% of the I129 assumed to be on the berl saddles and used solubility controlled release in the model (see Section 9.3.1.6.5). The other portion of the ORWBG, named ORWBG, contained the remaining inventory, and was set to use K_d-controlled release in the model.

The FSB and HSB were closed with clay caps in 1990 (WSRC 1993a and WSRC 1993b). However, the year 2002 was set as the Cap Placement Year, and infiltration associated with a clay cap, as outlined in Section 9.3.1.6.2, was implemented at that time for purposes of the CA calculation. Because of the short time lapse, the difference between the actual and model closure dates was deemed insignificant but should be corrected in future calculations.

Source release mechanisms used for the NRCDA 643-7E included the metal cask and metal corrosion source release mechanisms as described in Sections 9.3.1.6.2.2 and 9.3.1.6.4, respectively. The metal cask was assumed to remain intact and preclude infiltration through the waste for 750 years (Section 9.3.1.6.2.2), after that the radionuclides contained within the activated metals were assumed to be released based upon the corrosion rates of the activated Inconel and Zircaloy (Section 9.3.1.6.4, WSRC 2008a).

Source release mechanisms used for the Reactor Vessel stainless steel included the concrete cap, metal corrosion, and concrete barrier K_d source release mechanisms as described in Sections 9.3.1.6.2, 9.3.1.6.4, and 9.3.4.3/9.3.4.9, respectively. Infiltration through the waste was based upon that through a concrete cap (Section 9.3.1.6.2). The radionuclides contained within the activated stainless steel were assumed to be released based upon the stainless steel corrosion rate (Section 9.3.1.6.4, Council 2009 and WSRC 2008a). The radionuclides migrate through the reactor concrete base mat with concrete K_ds prior to being released to the vadose or aquifer zones (Sections 9.3.4.3 and 9.3.4.9).

The only source release mechanism used for the H-Area Canyon was reduced infiltration through the concrete cap described in Section 9.3.1.6.2. See WSRC (2007c) for a description of the H-Area Canyon.

The diffusion source release mechanism through concrete, as described in Section 9.3.1.6.3, is the only source release mechanism used for the tritium area buildings (232-H, HANM, HAOM, 264-2H, TEF). See WSRC (2007d) and WSRC (2007e) for a description of the tritium and TEF facilities, respectively.

The PA flux source release mechanism, as described in Section 9.3.1.6.6, is the only source release mechanism used for E-Area ILV, E-area LAWV, and Z-Area Vault 2. As shown in Section 9.3.1.6.6, using isotopic fluxes at the water table from PA calculations (references WSRC 2008a and LWO 2009 are the E-Area and Z-Area PAs, respectively) in place of the ground release for the ILV, LAWV and Z-Area Vault 2 sources produced significant decreases in the maximum dose.

Accounting for diffusion controlled release of tritium from building slabs led to significant decreases in the maximum doses. Accounting for metal corrosion to the NRCDA 643-7E components also led to significant decreases in maximum doses. The metal corrosion release mechanism also significantly reduced the dose contribution from activated stainless steel in the reactor buildings. Total dose from the reactor buildings was reduced by a factor of two to four. With the generic release mechanism, surface contamination in the reactor buildings typically gave doses of about 0.05 mrem/yr. This dose dominated the total dose calculated for the reactor buildings after the contribution from activated stainless steel was reduced.

The streambed sediment source release mechanism, as described in Section 9.3.1.6.8, is the only source release mechanism used for the eight IOUs: FMB, LTR, PB, SR-A, SR-B, SC, UTR and SpdTbr (see Table C-1 for source identification by abbreviated name).

Source release modeling accounting for reduced infiltration due to caps over the waste typically did not produce a significant decrease in the maximum dose. In particular, the dose from H-Canyon/HB-Line remained above 1.0 mrem/yr and a significant source of CA dose even when the reduced infiltration through a concrete cap (canyon roof) was accounted for. This mechanism increased the transport time but did not significantly decrease the magnitude of the maximum dose because Np237 has a half-life of over 2.0E+06 years. Figure 9-101 shows a graph of the H-Canyon dose obtained using generic release modeling while Figure 9-102 shows the results from source release modeling. Source release modeling causes the peak dose (almost entirely from Np237) to occur later and to be slightly attenuated and more diffuse. Similar results were found for the ORWBG and MWMF/LLRWDF. Delaying the isotope transport actually increased the dose from I129 complexed with silver in the ORWBG. Without the cap, the I129 peak in the ORWBG occurred prior to 2025 while with the cap in place the peak was delayed which caused a larger dose to appear during the CA assessment period.

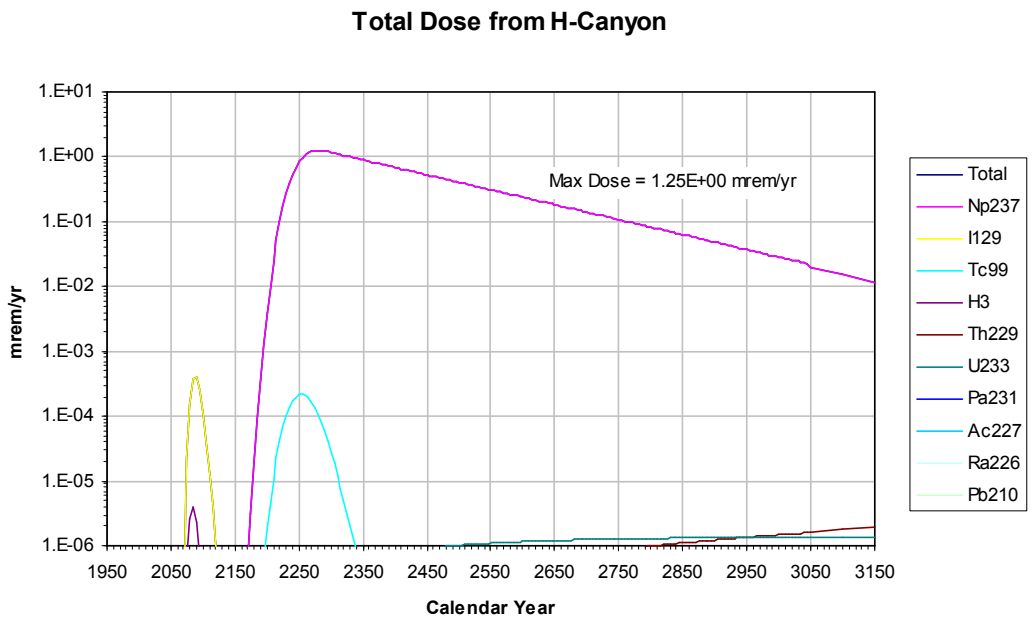


Figure 9-101. H-Canyon Dose with Generic Surface Release Modeling

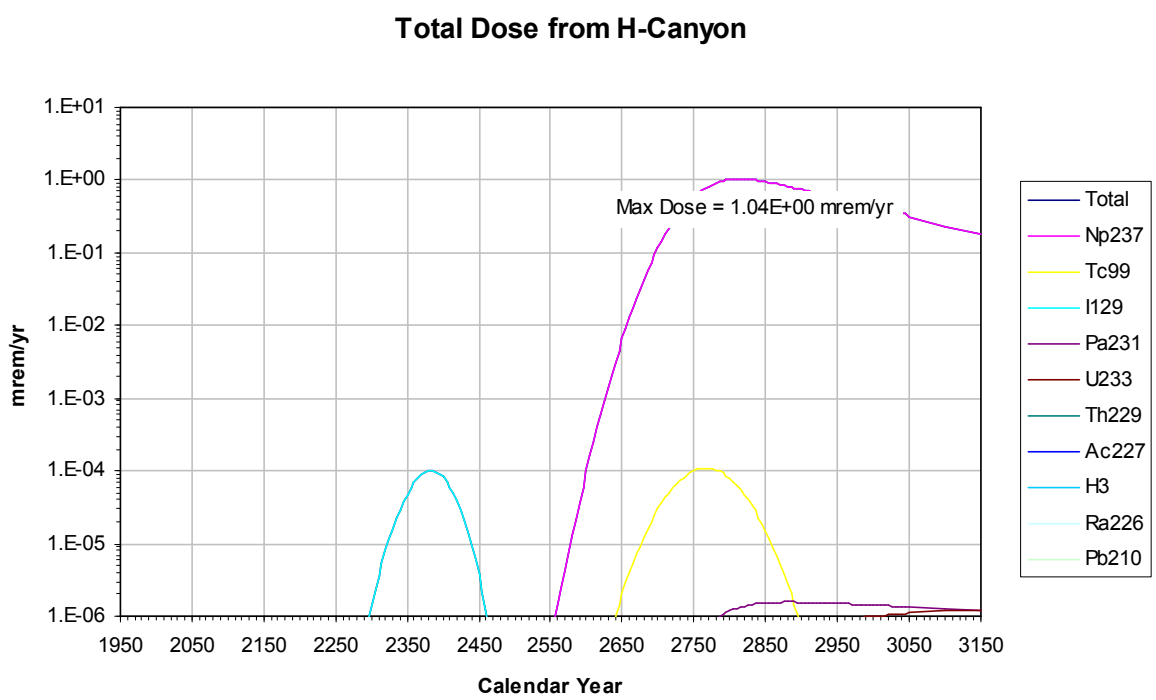


Figure 9-102. H-Canyon Dose with Source Release Modeling

9.3.7.2 Tritium Plumes

The tritium plume source release mechanism, as described in Section 9.3.1.6.7, is the only source release mechanism used for the seven Groundwater Operable Units (GOUs) (FMB, D-Area, K-Area, L-Area, P-Area, R-Area and TNX). Stream concentrations resulting from the discharge of tritium plumes into surface streams emanating from T-, R-, P-, L-, K- and D-Areas are shown in Figure 9-103. The concentrations are plotted on a logarithmic scale and initially appear as a straight line sloping downward where the downward slope is proportional to the half-life of tritium. Each of the straight-line segments eventually begins to decline sharply which reflects the complete flushing out of all of the tritium that was initially placed in the 200 Mixing Cells of the GoldSim™ model. It is clear that all of the tritium currently in the aquifer plumes will flush out into surface streams in less than 25 years. If any tritium still exists within the surface disposal facilities located within T-, R-, P-, L-, K- and D-Areas, that tritium would reach the stream at approximately the same time that the sharp declines are observed. Tritium stream concentrations for FMB are shown in Figure 9-104. This curve appears different than the curves in Figure 9-103 because it is based on direct measurement of tritium concentrations from stream samples. The decline in tritium concentration in FMB is projected into the future based on the assumption that the tritium flux to FMB will decline in proportion to the half-life of tritium, hence the straight line that extends out further in time. The projected concentrations for all SRS streams, as indicated in these graphs, were used to calculate doses to the member of the public at the CA Points of Assessment.

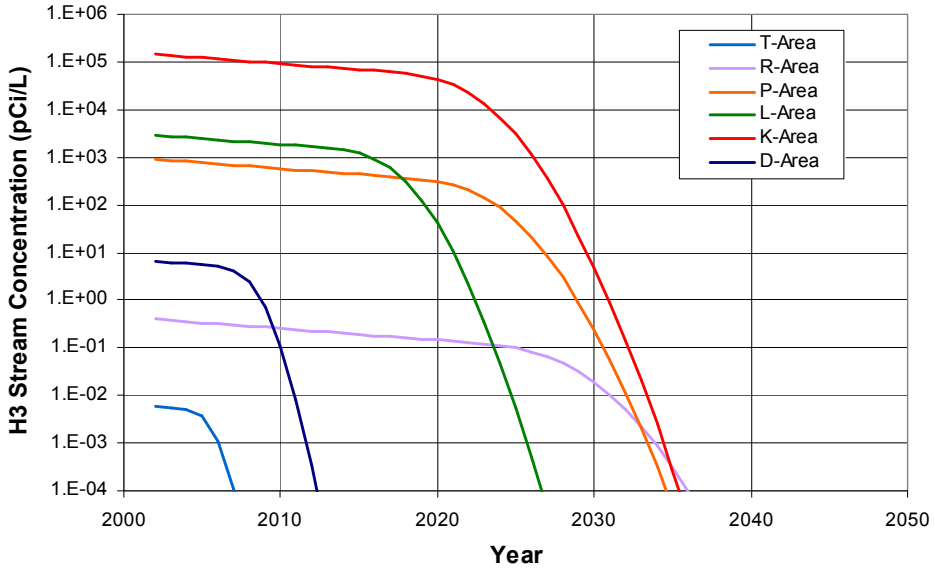


Figure 9-103. Projected Stream Tritium Concentrations Resulting from Plume Discharges

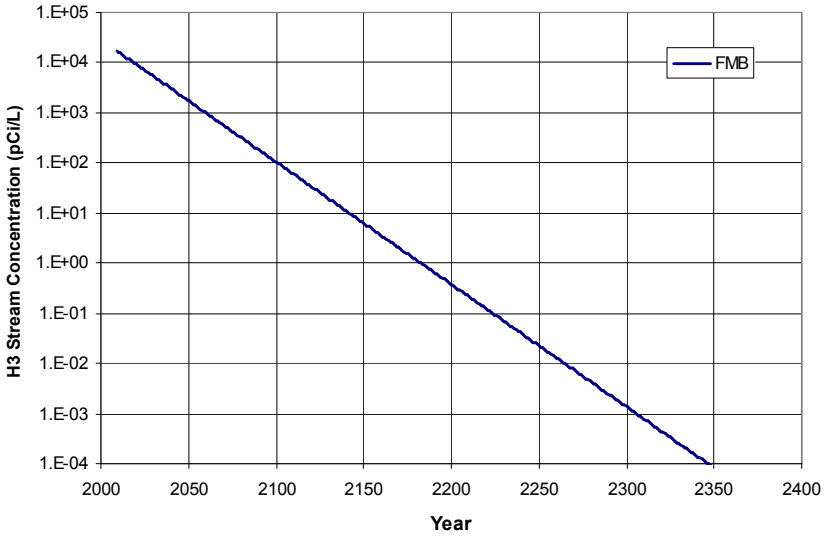


Figure 9-104. Projected FMB Tritium Concentrations

9.3.7.3 Streambed Sediments

The streambed sediment source release mechanism, as described in Section 9.3.1.6.8, is the only source release mechanism used for the eight IOUs: FMB, LTR, PB, SR-A, SR-B, SC, UTR and SpdTbr (see Table C-1 for source identification by abbreviated name). Except for UTR, radionuclide release from streambed sediments was the largest contributor to the dose at each POA. With the exception of the Tims Branch/Steed Pond IOU, the only significant radionuclide was Cs137. This is the only radionuclide that appears in sufficient quantities in stream sediment samples that will deliver a significant dose at the CA POAs. In the Tims Branch/Steed Pond IOU there are significant quantities of depleted uranium (DU) in the streambed sediments, but no Cs137. The specific uranium isotopes of concern are U238, U235 and U234.

To validate the sediment model, simulated results from the GoldSim^{TM} model were compared against the results of stream water samples collected in each of the SRS streams as part of the ongoing environmental monitoring program at SRS. The surface water analytical results for each year, from 2003 to 2007, are presented in the SRS Environmental Report for each of those years. Typically, the mean, maximum and minimum computed from multiple sample analyses associated with a specific sampling station on a specific stream are presented for each radionuclide of interest. Because multiple sampling stations may exist on an individual SRS stream, the most downstream sampling station was selected and the mean value for each year for Cs137 and U238, U235 and U234 was selected to compare against CA GoldSim^{TM} model results.

Results of the CA GoldSim™ model simulations of stream sediment sources are presented below, for each of the SRS IOUs. Two graphs are shown for each radionuclide associated with an IOU, the first (on the left) being focused on the time-period over which SRS reports average stream concentrations based on samples collected from each stream (2003 to 2007) as part of the environmental monitoring program. The second graph (on the right) illustrates the simulated stream concentration projected into the future. For UTR, LTR, FMB, PB, and SC, Cs137 is relatively tightly bound to the sand in streambed sediments (K_d = 50 ml/g) but has a relatively short half-life (30 yrs). Therefore, the rapid decline in the stream concentration of Cs137 over time is due to decay.

9.3.7.3.1 Upper Three Runs

Measured annual average stream concentrations for UTR and the GoldSim™ model simulated concentrations are shown in Figure 9-105. Simulated concentrations match closely with measured results during the period of overlap, falling within the measured range. The simulation results indicate that Cs137 stream concentrations will diminish to levels below 1.0E-02 pCi/L within 100 years and below 1.0E-03 pCi/L within 200 years.

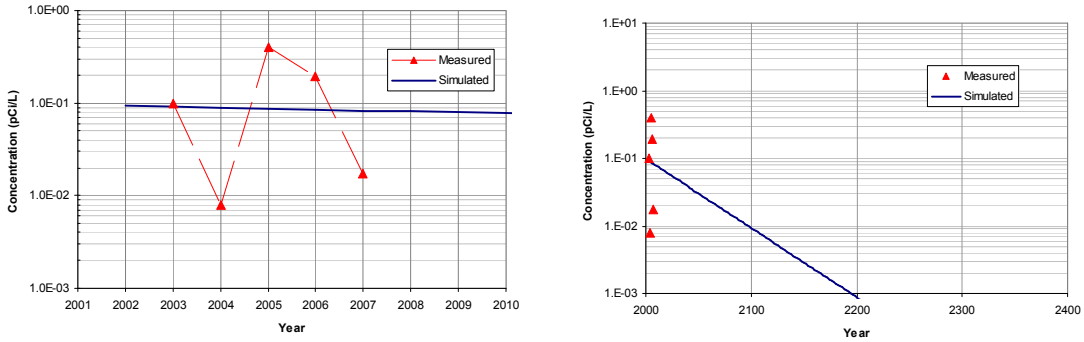


Figure 9-105. Upper Three Runs Cs137 Current and Projected Stream Concentrations

9.3.7.3.2 Lower Three Runs

Measured annual average stream concentrations for LTR and the GoldSim™ model simulated concentrations are illustrated in Figure 9-106. Simulated concentrations match closely with measured results during the period of overlap, falling within the measured range. The simulation results indicate that Cs137 stream concentrations will diminish to levels below 1.0E-02 pCi/L within 250 years and below 1.0E-03 pCi/L within 350 years.

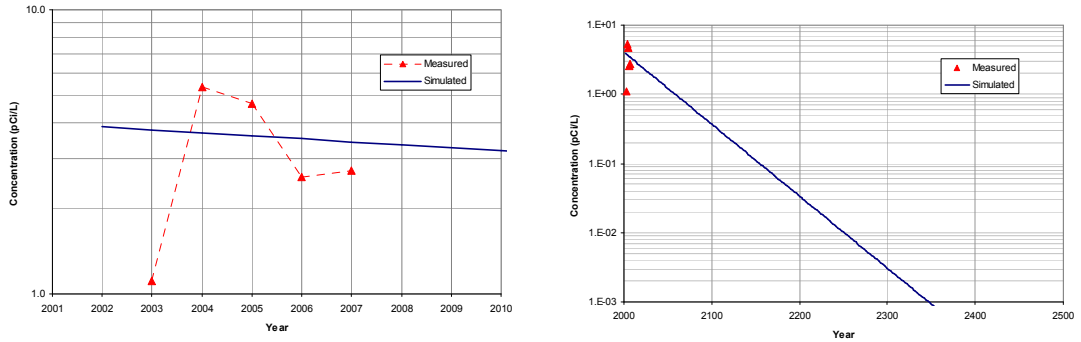


Figure 9-106. Lower Three Runs Cs137 Current and Projected Stream Concentrations

9.3.7.3.3 <u>Fourmile Branch</u>

Measured annual average stream concentrations for FMB and the GoldSim^{TM} model simulated concentrations are illustrated in Figure 9-107. Simulated concentrations match closely with measured results during the period of overlap, falling within the measured range. The simulation results indicate that Cs137 stream concentrations will diminish to levels below 1.0E-02 pCi/L within approximately 230 years and below 1.0E-03 pCi/L within approximately 330 years.

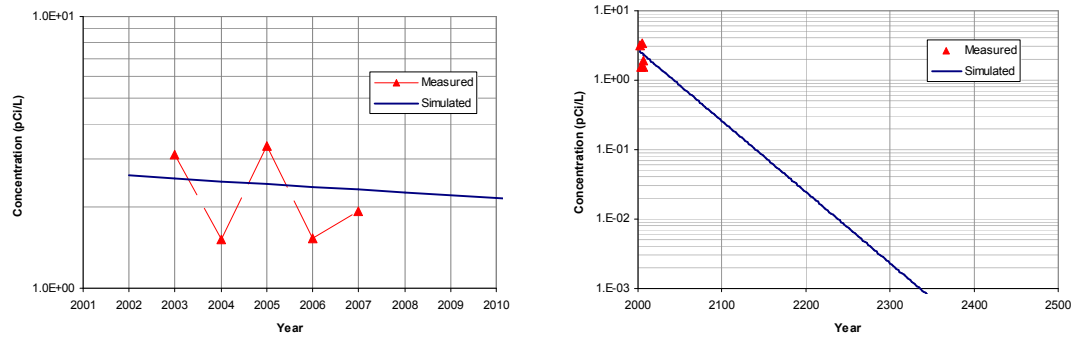


Figure 9-107. Fourmile Branch Cs137 Current and Projected Stream Concentrations

9.3.7.3.4 Pen Branch

Measured annual average stream concentrations for PB and the GoldSim™ model simulated concentrations are illustrated in Figure 9-108. Simulated concentrations match closely with measured results during the period of overlap, falling within the measured range. The simulation results indicate that Cs137 stream concentrations will diminish to levels below 1.0E-02 pCi/L within approximately 90 years and below 1.0E-03 pCi/L within approximately 200 years.

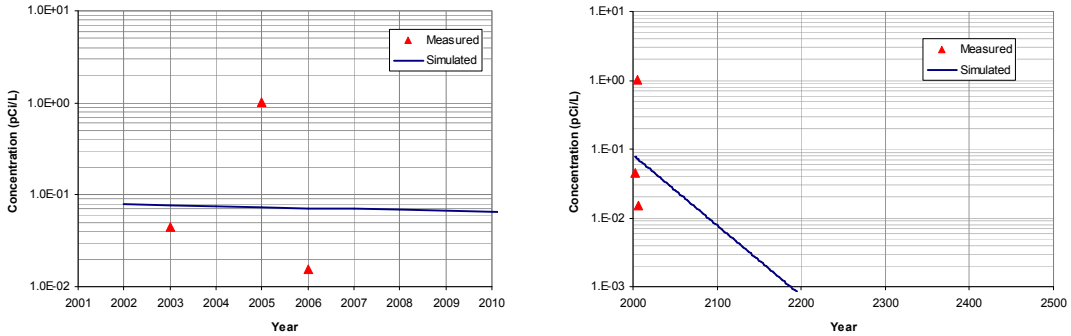


Figure 9-108. Pen Branch Cs137 Current and Projected Stream Concentrations

9.3.7.3.5 Steel Creek

Measured annual average stream concentrations for SC and the GoldSim™ model simulated concentrations are illustrated in Figure 9-109. Simulated concentrations match closely with measured results during the period of overlap, falling within the measured range. The simulation results indicate that Cs137 stream concentrations will diminish to levels below 1.0E-02 pCi/L within 150 years and below 1.0E-03 pCi/L within 250 years.

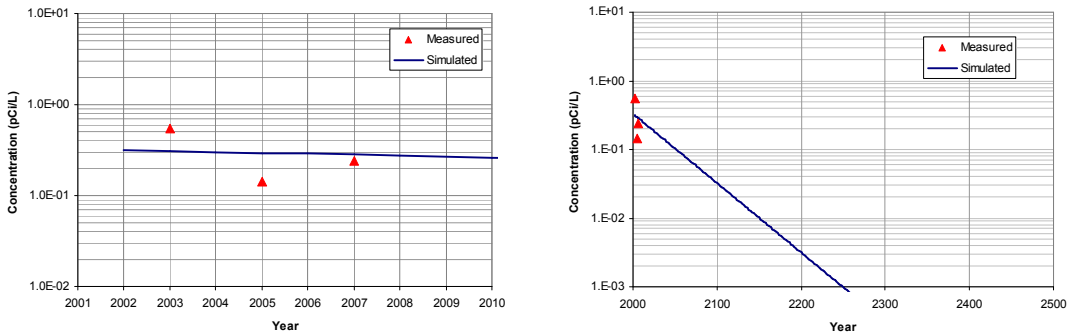


Figure 9-109. Steel Creek Cs137 Current and Projected Stream Concentrations

9.3.7.3.6 Tims Branch and Steed Pond

Tims Branch and Steed Pond received waste water containing DU, which contains U238, U235 and U234. Significant quantities of these radionuclides are bound up in sediments of Steed Pond and within Tims Branch channel sediments. Because Tims Branch discharges into UTR, the GoldSim™ Base Case model results are compared against annual average concentration measurements of U238, U235 and U234 obtained from UTR. Measured and simulated stream concentrations for each of these isotopes are presented in Figure 9-110 through Figure 9-112.

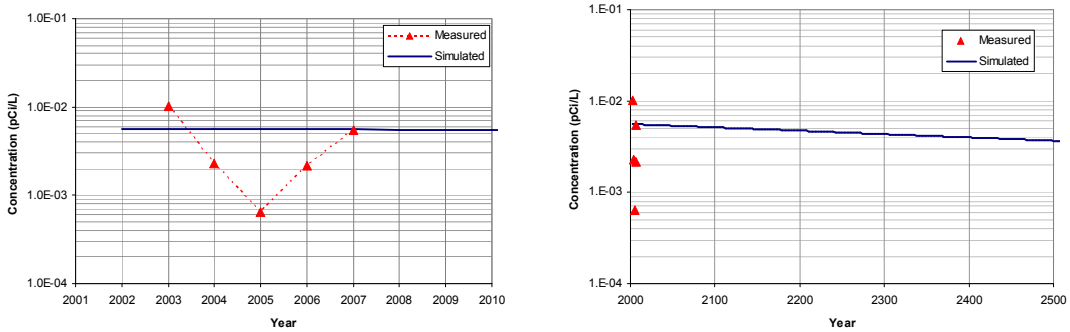


Figure 9-110. Tims Branch/Steed Pond U234 Current and Projected Stream Concentrations

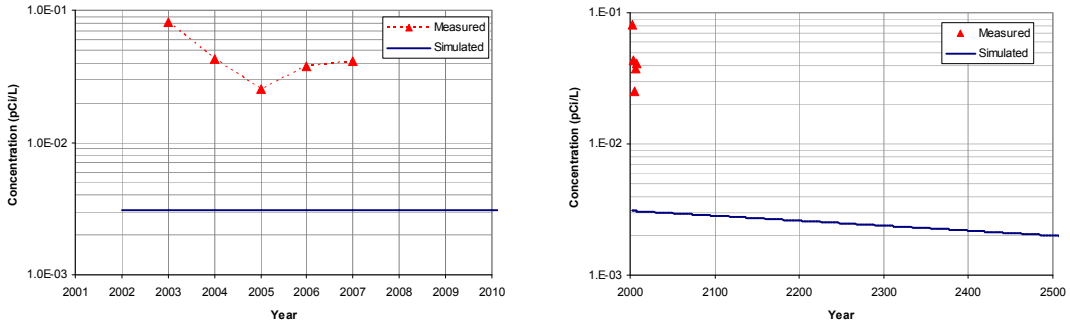


Figure 9-111. Tims Branch/Steed Pond U235 Current and Projected Stream Concentrations

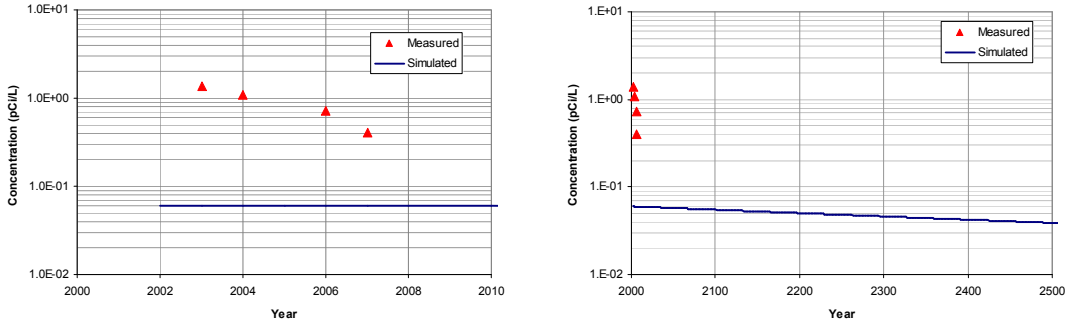


Figure 9-112. Tims Branch/Steed Pond U238 Current and Projected Stream Concentrations

In Figure 9-110, the simulated concentrations of U234 fall within the range of measured concentrations, which range from 6.0E-04 to 1.0E-02 pCi/L. Simulated concentrations decline slowly in the future due to the strong affinity of uranium to soil sediment particles and the long half-life of U234. In Figure 9-111, the simulated concentrations of U235 fall well below the range of measured concentrations, which range from 2.5E-02 to 8.0E-02 pCi/L. The simulated stream concentration of 3.0E-3 pCi/L is approximately one order of magnitude lower than the measured values. Simulated concentration projections decline slowly from this concentration in the future due to the strong affinity of uranium to soil sediment particles and the long half-life of U235. In Figure 9-112, the simulated concentrations of U238 also fall well below the range of measured concentrations, which range from 4.0E-01 to 1.0E+00 pCi/L. The simulated stream concentration of 6.0E-02 pCi/L is approximately one order of magnitude lower than the measured values. Simulated concentration projections decline slowly from this concentration in the future due to the strong affinity of uranium to soil sediment particles and the long half-life of U238.

The most likely explanation of the significant discrepancy between measured and modeled stream concentrations is the dissolution of uranium from entrained particulate matter in samples collected from site streams for analysis. SRS is required to collect unfiltered surface water samples which are then acidified to a pH < 2 prior to chemical analysis. This hypothesis is substantiated by research conducted to analyze the transport of uranium in Tims Branch (Baston et. al. 1996) in which the differences in uranium concentration for filtered samples ranged from one to three orders of magnitude lower than unfiltered samples, depending on whether samples were collected during times of base flow or at times of elevated flow rates during storms. In addition, the report suggests that the primary mechanism for uranium export from Steed Pond and Tims Branch is by particulate transport in surface water.

9.3.7.4 Summary of Source Release Results

In summary, source release modeling gave the following general results:

- Utilizing PA fluxes, which account for waste forms and barriers, produces a significant decrease in projected dose.
- Accounting for metal corrosion produces a significant decrease in projected dose.
- Accounting for tritium diffusion out of concrete produces a significant decrease in projected dose.
- Accounting for concrete barrier K_ds typically produced only a minor decrease in the projected maximum dose but introduced a significant delay in the peak time.
- Accounting for metal casks reduces the dose from radionuclides with half-lives significantly less than the failure time of the cask but does little for longer lived radionuclides.
- Reductions in infiltration through caps generally produced minor reductions in maximum doses, particularly for long-lived radionuclides, for which the assumed time to failure of the closure cap is short relative to the radionuclide half-life.
- Tritium plumes produced only small contributions to POA doses that were greatest at the start of the CA assessment period and rapidly declined thereafter.
- Cesium releases from streambed sediment gave significant contributions to the total dose at all POAs. Except for UTR, Cs137 from the stream beds caused the maximum dose at each POA, and it occurred at the start of the CA assessment period and quickly diminished.

9.3.8 Composite Analysis Base Case Deterministic Results

Results are presented below for the base-case CA deterministic calculations obtained by combining the generic (108 sources) and source release (44 sources) modeling discussed in the previous sections (see Figure 9-1 and Figure 9-2). Results are presented as tabular summaries of maximum doses for all sources.

9.3.8.1 Maximum Doses for All Sources

Base case modeling results are summarized in Table 9-24 and Table 9-25 where maximum doses for each source are presented. Three sources give doses greater than 1.0 mrem/yr and another four sources give doses greater than 0.1 mrem/yr. The sum of all maximum doses is 7.51 mrem/yr which is about 20% of the value of 37.66 mrem/yr obtained with generic surface release modeling. This estimate of the total CA dose is still conservative because it assumes that all of the maximum doses occur at the same time and at the same place. In Table 9-24 and Table 9-25, the sources are grouped by area with POAs denoted by the POA identification number (see Table 9-7). Table 9-26 through Table 9-30 present the same results grouped by POA and sorted from lowest to highest value. In all of the tables, cells shaded orange indicate doses greater than 1.0 mrem/yr and cells shaded yellow indicate doses greater than 0.1 mrem/yr.

The grouping and numbering of sources in Table 9-24 and Table 9-25 follow the order:

A-Area	1 - 4	J-Area	83	S-Area	114 - 117
C-Area	5 - 10	K-Area	84 - 89	T-Area	118 - 122
D-Area	11 - 15	L-Area	90 - 95	Z-Area	123 - 127
E-Area	16 - 36	M-Area	96 - 97	MOXFF	128
F-Area	37 - 52	N-Area	98	Spills	129 - 137
H-Area	53 - 73	P-Area	99 - 104	GOU	138 - 144
Tritium Area	74 - 82	R-Area	105 - 113	IOU	145 - 152

Table 9-24. Maximum Doses from Base Case Modeling for Sources 1 – 76

Number	POA	Unit Name	Max Dose mrem/yr	Number	POA	Unit Name	Max Dose mrem/yr
1	1	A-Area Seep	3.00E-99	39	3	F-Area IPSL FMB	1.94E-05
2	5	773-A	1.26E-12	40	1	221-F	2.75E-03
3	1	776-A	3.00E-67	41	1	211-F	6.15E-05
4	5	778-A	5.37E-82	42	1	281-3F	8.34E-03
5	3	105-C SS	8.42E-08	43	3	F-Area Seep	2.24E-03
6	3	105-C Al	9.25E-14	44	1	FTF Type I	2.16E-06
7	3	105-C Con	1.91E-04	45	1	FTF Type IV	1.27E-07
8	3	105-C Surf	5.75E-02	46	1	FTF Type III	1.43E-03
9	3	C-Area Seep	1.39E-20	47	1	FTF Type IIIA UTR	3.07E-06
10	3	C-Area Dirt	1.71E-23	48	3	FTF Type IIIA FMB	1.03E-05
11	5	420-D	6.10E-15	49	1	FTF Equip	1.08E-05
12	5	420-2D	2.07E-08	50	1	772-F	1.82E-06
13	5	421-D	4.89E-24	51	1	772-1F,-4F	1.87E-06
14	5	421-2D	6.80E-14	52	1	F-Area Old Seep	1.16E-02
15	5	772-D	4.15E-09	53	1	HTF Type I UTR E	1.07E-04
16	1	E-Area CIG	2.09E-02	54	1	HTF Type I UTR W	8.44E-06
17	1	E-Area ET	2.70E-04	55	1	HTF Type II	9.81E-05
18	1	E-Area ILV	2.66E-05	56	1	HTF Type IV UTR	1.49E-06
19	1	E-Area LAWV	2.84E-06	57	3	HTF Type IV FMB	1.53E-04
20	1	NR Pad 1 Part 1	3.58E-04	58	1	HTF Type III UTR E	1.29E-05
21	1	NR Pad 1 Part 2	4.80E-04	59	1	HTF Type III UTR W	4.36E-07
22	1	NR Pad 1 Part 3	6.57E-05	60	3	HTF Type III FMB	2.49E-03
23	1	NR Pad 1 Part 4	1.49E-05	61	1	HTF Type IIIA UTR E	2.67E-04
24	1	NR Pad 1 Part 5	1.20E-03	62	1	HTF Type IIIA UTR W	4.43E-06
25	1	NR Pad 1 Part 6	3.66E-03	63	3	HTF Type IIIA FMB	3.89E-03
26	1	NR Pad 1 Crud	5.90E-04	64	1	HTF Equip UTR	4.55E-06
27	1	NR Pad 2	3.65E-02	65	3	HTF Equip FMB	5.88E-04
28	1	E-Area ST East	3.23E-04	66	1	H-Area IPSL UTR	3.69E-03
29	1	E-Area ST Cent	1.33E-02	67	3	H-Area IPSL FMB	4.36E-03
30	1	E-Area ST West	1.36E-04	68	1	281-1H, 2H	8.17E-07
31	1	TRU Pad	1.19E-12	69	1	281-8H	6.07E-04
32	3	LLRWDF FMB	1.79E-02	70	3	H-Area Seep	1.03E-01
33	1	LLRWDF UTR	1.80E-02	71	1	H-Canyon	1.04E+00
34	1	MWMF	1.35E-01	72	1	RBOF	5.72E-08
35	3	ORWBG	7.28E-03	73	1	H-Area Sand	1.20E-08
36	3	ORWBG AgI	1.63E-01	74	1	232-H	1.11E-02
37	1	F-Area MS	1.67E-06	75	1	HANM	6.07E-02
38	1	F-Area IPSL UTR	4.89E-04	76	1	HAOM	2.02E-02

Table 9-25. Maximum Doses from Base Case Modeling for Sources 77 – 152

Number	POA	Unit Name	Max Dose mrem/yr	Number	POA	Unit Name	Max Dose mrem/yr
77	1	234-7H	4.21E-03	115	1	511-S LPPP	3.76E-04
78	1	236-H	7.13E-04	116	1	294-S Sand L	9.83E-07
79	1	237-H	5.33E-05	117	1	294-S Sand S	3.43E-06
80	1	TEF	3.38E-03	118	5	TNX BG	7.78E-06
81	1	264-2H	1.48E-02	119	5	T-Area IPSL	3.90E-06
82	1	TPBAR	2.44E-02	120	5	TNX Gully	4.43E-06
83	1	J-Area SWPF	1.34E-05	121	5	TNX Outfall	4.61E-04
84	4	KAC SS	4.16E-06	122	5	677-T	2.76E-07
85	4	KAC Al	4.46E-11	123	1	Z-Area Vault 1	7.71E-03
86	4	KAC Con	2.17E-03	124	1	Z-Area Vault 4	4.93E-02
87	4	KAC Surf	5.30E-02	125	1	Z-Area Vault 2 NL	1.06E-09
88	4	K-Area Seep	7.11E-05	126	1	Z-Area Vault 2 NS	3.58E-10
89	4	K-Area Dirt	5.02E-17	127	1	Z-Area Vault 2 S	1.19E-09
90	4	LAC SS	1.42E-05	128	1	MOXFF	3.62E-05
91	4	LAC Al	3.61E-07	129	1	T-13 Spill	1.89E-05
92	4	LAC Con	8.24E-03	130	1	T-9 Spill	6.00E-06
93	4	LAC Surf	7.57E-02	131	1	T-16 Spill	4.73E-05
94	4	L-Area Seep	2.17E-04	132	1	T-37 Spill UTR	2.57E-05
95	4	L-Area Dirt	2.04E-06	133	3	T-37 Spill FMB	8.20E-04
96	1	M-Area HWMF	5.25E-10	134	1	281-3F Spill	1.83E-06
97	1	M-Area MIPSL	3.99E-12	135	1	T-3 Spill	1.61E-06
98	1	N-Area Seep	1.00E-99	136	1	T-8 Spill	1.78E-04
99	4	105-P SS	1.59E-05	137	1	281-3H Spill	8.97E-06
100	4	105-P Al	6.46E-07	138	3	FMB GOU	6.81E-03
101	4	105-P Con	1.54E-02	139	5	D-Area GOU	1.57E-80
102	4	105-P Surf	1.14E-01	140	4	K-Area GOU	2.47E-03
103	4	P-Area Seep	2.80E-06	141	4	L-Area GOU	1.20E-11
104	4	P-Area Dirt	6.61E-04	142	4	P-Area GOU	5.73E-05
105	2	105-R SS	1.91E-06	143	2	R-Area GOU	2.68E-07
106	2	105-R Al	9.33E-10	144	5	T-Area GOU	1.00E-99
107	2	105-R Con L	4.15E-07	145	3	FMB IOU	1.97E+00
108	2	105-R Con S	3.26E-03	146	2	LTR IOU	2.94E+00
109	2	105-R Surf L	4.10E-04	147	4	PB IOU	6.00E-02
110	2	105-R Surf S	4.68E-02	148	5	SR-A IOU	1.76E-02
111	2	R-Area Seep	3.84E-05	149	5	SR-B IOU	1.38E-02
112	2	R-Area Dirt L	1.46E-48	150	4	SC IOU	2.43E-01
113	2	R-Area Dirt S	1.31E-06	151	1	UTR IOU	7.19E-02
114	1	221-S	7.11E-04	152	1	SpdTbr IOU	2.03E-03

Table 9-26. Maximum Doses from Each Source to UTR POA

Number	Unit Name	Max Dose mrem/yr	Number	Unit Name	Max Dose mrem/yr
98	N-Area Seep	1.00E-99	22	NR Pad 1 Part 3	6.57E-05
1	A-Area Seep	3.00E-99	55	HTF Type II	9.81E-05
3	776-A	3.00E-67	53	HTF Type I UTR E	1.07E-04
31	TRU Pad	1.19E-12	30	E-Area ST West	1.36E-04
97	M-Area MIPSL	3.99E-12	136	T-8 Spill	1.78E-04
126	Z-Area Vault 2 NS	3.58E-10	61	HTF Type IIIA UTR E	2.67E-04
96	M-Area HWMF	5.25E-10	17	E-Area ET	2.70E-04
125	Z-Area Vault 2 NL	1.06E-09	28	E-Area ST East	3.23E-04
127	Z-Area Vault 2 S	1.19E-09	20	NR Pad 1 Part 1	3.58E-04
73	H-Area Sand	1.20E-08	115	511-S LPPP	3.76E-04
72	RBOF	5.72E-08	21	NR Pad 1 Part 2	4.80E-04
45	FTF Type IV	1.27E-07	38	F-Area IPSL UTR	4.89E-04
59	HTF Type III UTR W	4.36E-07	26	NR Pad 1 Crud	5.90E-04
68	281-1H, 2H	8.17E-07	69	281-8H	6.07E-04
116	294-S Sand L	9.83E-07	114	221-S	7.11E-04
56	HTF Type IV UTR	1.49E-06	78	236-H	7.13E-04
135	T-3 Spill	1.61E-06	24	NR Pad 1 Part 5	1.20E-03
37	F-Area MS	1.67E-06	46	FTF Type III	1.43E-03
50	772-F	1.82E-06	152	SpdTbr IOU	2.03E-03
134	281-3F Spill	1.83E-06	40	221-F	2.75E-03
51	772-1F,-4F	1.87E-06	80	TEF	3.38E-03
44	FTF Type I	2.16E-06	25	NR Pad 1 Part 6	3.66E-03
19	E-Area LAWV	2.84E-06	66	H-Area IPSL UTR	3.69E-03
47	FTF Type IIIA UTR	3.07E-06	77	234-7H	4.21E-03
117	294-S Sand S	3.43E-06	123	Z-Area Vault 1	7.71E-03
62	HTF Type IIIA UTR W	4.43E-06	42	281-3F	8.34E-03
64	HTF Equip UTR	4.55E-06	74	232-H	1.11E-02
130	T-9 Spill	6.00E-06	52	F-Area Old Seep	1.16E-02
4	HTF Type I UTR W	8.44E-06	29	E-Area ST Cent	1.33E-02
137	281-3H Spill	8.97E-06	81	264-2H	1.48E-02
49	FTF Equip	1.08E-05	33	LLRWDF UTR	1.80E-02
58	HTF Type III UTR E	1.29E-05	76	HAOM	2.02E-02
83	J-Area SWPF	1.34E-05	16	E-Area CIG	2.09E-02
23	NR Pad 1 Part 4	1.49E-05	82	TPBAR	2.44E-02
129	T-13 Spill	1.89E-05	27	NR Pad 2	3.65E-02
132	T-37 Spill UTR	2.57E-05	124	Z-Area Vault 4	4.93E-02
18	E-Area ILV	2.66E-05	75	HANM	6.07E-02
128	MOXFF	3.62E-05	151	UTR IOU	7.19E-02
131	T-16 Spill	4.73E-05	34	MWMF	1.35E-01
79	237-H	5.33E-05	71	H-Canyon	1.04E+00
41	211-F	6.15E-05			

Table 9-27. Maximum Doses from Each Source to LTR POA

Number	Unit Name	Max Dose mrem/yr	Number	Unit Name	Max Dose mrem/yr
112	R-Area Dirt L	1.46E-48	111	R-Area Seep	3.84E-05
106	105-R Al	9.33E-10	109	105-R Surf L	4.10E-04
143	R-Area GOU	2.68E-07	108	105-R Con S	3.26E-03
107	105-R Con L	4.15E-07	110	105-R Surf S	4.68E-02
113	R-Area Dirt S	1.31E-06	146	LTR IOU	2.94E+00
105	105-R SS	1.91E-06			

Table 9-28. Maximum Doses from Each Source to FMB POA

Number	Unit Name	Max Dose mrem/yr	Number	Unit Name	Max Dose mrem/yr
10	C-Area Dirt	1.71E-23	60	HTF Type III FMB	2.49E-03
9	C-Area Seep	1.39E-20	63	HTF Type IIIA FMB	3.89E-03
6	105-C Al	9.25E-14	67	H-Area IPSL FMB	4.36E-03
5	105-C SS	8.42E-08	138	FMB GOU	6.81E-03
48	FTF Type IIIA FMB	1.03E-05	35	ORWBG	7.28E-03
39	F-Area IPSL FMB	1.94E-05	32	LLRWDF FMB	1.79E-02
133	T-37 Spill FMB	8.20E-04	8	105-C Surf	5.75E-02
57	HTF Type IV FMB	1.53E-04	70	H-Area Seep	1.03E-01
7	105-C Con	1.91E-04	36	ORWBG AgI	1.63E-01
65	HTF Equip FMB	5.88E-04	145	FMB IOU	1.97E+00
43	F-Area Seep	2.24E-03			

Table 9-29. Maximum Doses from Each Source to SC/PB POA

Number	Unit Name	Max Dose mrem/yr	Number	Unit Name	Max Dose mrem/yr
89	K-Area Dirt	5.02E-17	94	L-Area Seep	2.17E-04
141	L-Area GOU	1.20E-11	104	P-Area Dirt	6.61E-04
85	KAC Al	4.46E-11	86	KAC Con	2.17E-03
91	LAC Al	3.61E-07	140	K-Area GOU	2.47E-03
100	105-P Al	6.46E-07	92	LAC Con	8.24E-03
95	L-Area Dirt	2.04E-06	101	105-P Con	1.54E-02
103	P-Area Seep	2.80E-06	87	KAC Surf	5.30E-02
84	KAC SS	4.16E-06	147	PB IOU	6.00E-02
90	LAC SS	1.42E-05	93	LAC Surf	7.57E-02
99	105-P SS	1.59E-05	102	105-P Surf	1.14E-01
142	P-Area GOU	5.73E-05	150	SC IOU	2.43E-01
88	K-Area Seep	7.11E-05			

Table 9-30. Maximum Doses from Each Source Directly to Savannah River

Number	Unit Name	Max Dose mrem/yr	Number	Unit Name	Max Dose mrem/yr
144	T-Area GOU	1.00E-99	12	420-2D	2.07E-08
4	778-A	5.37E-82	122	677-T	2.76E-07
139	D-Area GOU	1.57E-80	119	T-Area IPSL	3.90E-06
13	421-D	4.89E-24	120	TNX Gully	4.43E-06
11	420-D	6.10E-15	118	TNX BG	7.78E-06
14	421-2D	6.80E-14	121	TNX Outfall	4.61E-04
2	773-A	1.26E-12	149	SR-B IOU	1.38E-02
15	772-D	4.15E-09	148	SR-A IOU	1.76E-02

9.3.8.2 Results for Major Sources

In Appendix E, results are presented as graphs showing the time behavior of the dose for a limited number of sources over time for each POA. A total of 152 sources were analyzed as part of the CA. Results for each source are best presented as a set of four graphs showing the dose from the most significant radionuclides, residential dose by pathway, recreational dose by pathway, and total dose. Graphs showing the radionuclide doses plot up to the 10 radionuclides that give the highest doses during the CA assessment period. The radionuclides are listed in the figure legends in order from the one giving the highest dose to the one giving the 10th highest dose. Presenting this level of information for all 152 sources would require 608 plots which would make the report cumbersome and make it difficult for the reader to find the significant results. To reduce the number of plots, detailed results are only presented for those sources that gave a maximum base-case dose greater than 0.01 mrem/yr at any time during the 1,000 year CA assessment period from 2025 to 3025. Based on this criterion (1% of 1% of the primary dose limit), there are 29 major sources for which graphical results from the dose calculations are provided.

Figures E-1 through E-116 in Appendix E, which show these results, are arranged as follows:

- Figures E-1 through E-56 show results for the 14 major sources that drain into UTR.
- Figures E-57 through E-64 show results for the two major sources that drain into LTR.
- Figures E-65 through E-84 show results for the five major sources that drain into FMB.
- Figures E-85 through E-108 show results for the six major sources that drain into SC and PB.
- Figures E-109 through E-116 show results for the two major sources that drain directly into the Savannah River.

Within these groups of figures, the sources are arranged in alphabetical order. A list of the major sources for each POA is given in Table 9-27 in Section 9.3.8.1.

9.3.8.3 Results for Performance Assessment Facilities

To assess the impact of the PA facilities on the CA dose, the PA facilities were separated out and the results for total dose plotted separately. Figure 9-113 shows the contribution from the HTF, FTF, ELLWF, and Z-Area facilities to the total CA dose at the UTR POA. For comparison, the maximum CA dose at UTR was 1.06 mrem/yr. In the years 2210 and 2615 the PA facilities contribute 55% and 50% of the total dose at UTR, respectively. These are the largest percentages during the CA period of assessment. By 3025 the contribution to the total dose at UTR from the PA facilities has fallen to about 4% but it is increasing.

Figure 9-114 shows the contribution from the HTF and FTF to the total CA dose at the FMB POA. In Figure 9-114, the total PA dose follows the HTF curve since the contribution from the FTF is negligible. For comparison, the maximum CA dose at FMB was 2.14 mrem/yr. In the year 2310 the PA facilities contribute about 26% of the total dose at FMB which is the maximum percentage during the CA assessment period. By 3025 the contribution to total dose at FMB from the PA facilities is less than 1% but it is increasing.

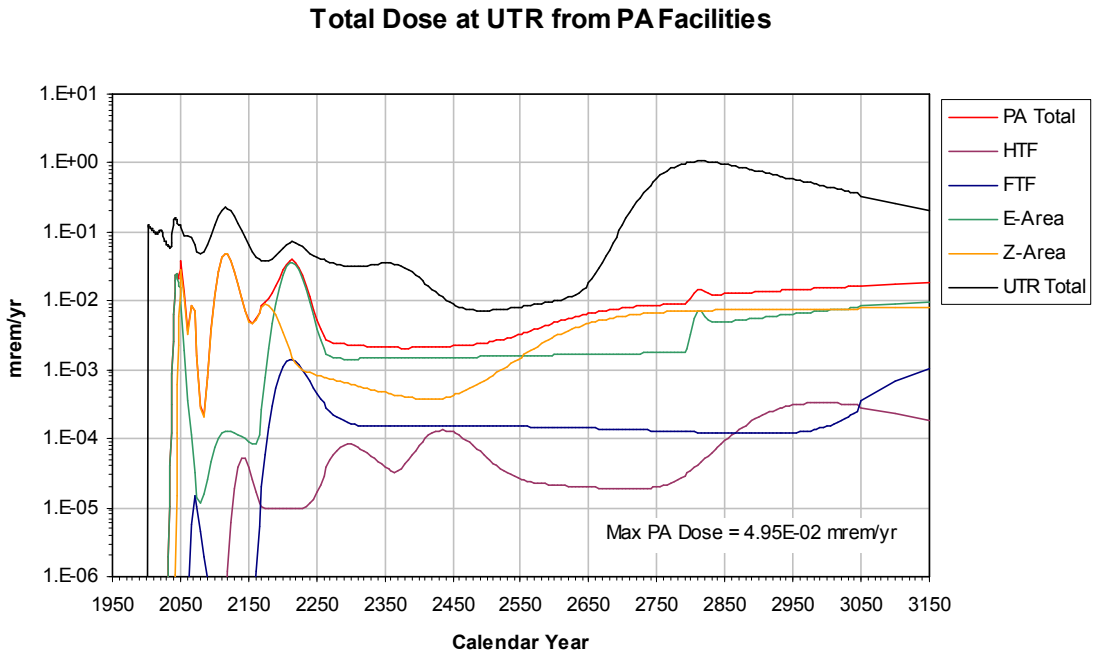


Figure 9-113. Direct Contribution from PA Facilities to CA Dose at UTR POA

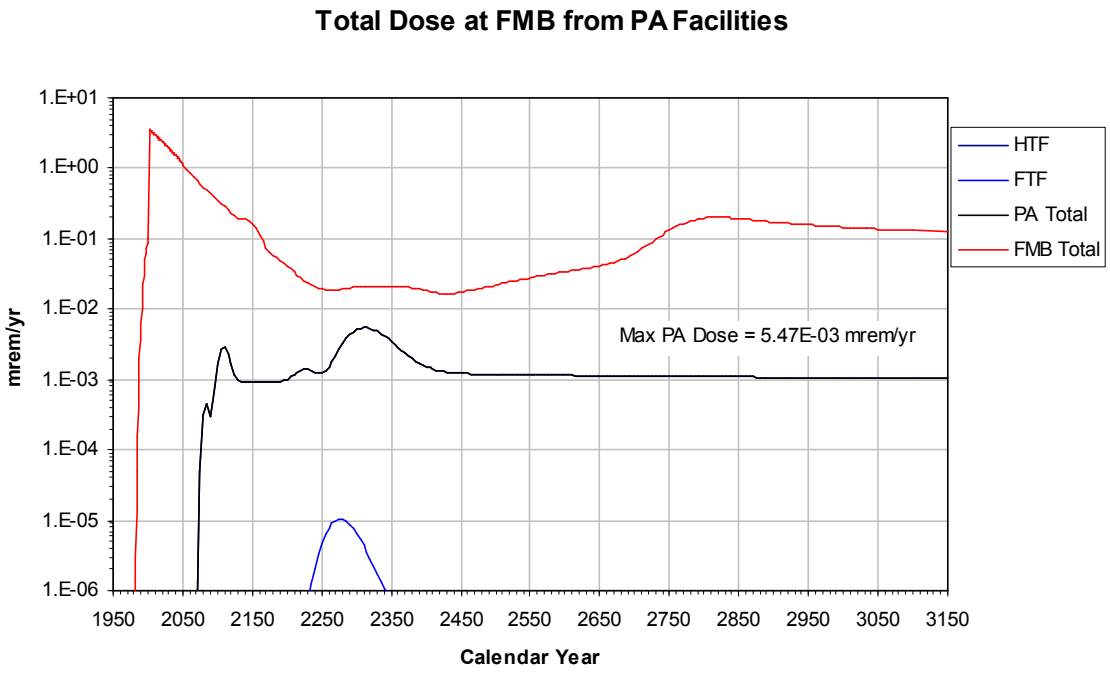


Figure 9-114. Direct Contribution from PA Facilities to CA Dose at FMB POA

9.3.8.4 Results by POA

Figure 9-115 through Figure 9-119 plot the total dose from direct discharges to each POA over the CA assessment period. Additional figures in Appendix E (Figures E-117 through E-136) show the contribution to the dose from the 10 most significant sources, the 10 most significant radionuclides, and from residential and recreational dose pathways. The doses plotted in these figures do not include contributions from upstream sources. Therefore, the Savannah River doses plotted in Figure 9-119 (and Figure E-33 through Figure E-36) are based on discharges directly into the river from the flood plain and swamp, TNX, D-Area, A-Area, and M-Area and do not correspond to a total dose at the U.S. Highway 301 Bridge POA (the total cumulative dose at the U.S. Highway 301 Bridge POA is discussed in Section 10.0). The maximum dose over the 1,000 year CA period of assessment is indicated on the graphs below and listed in Table 9-31 in the next section.

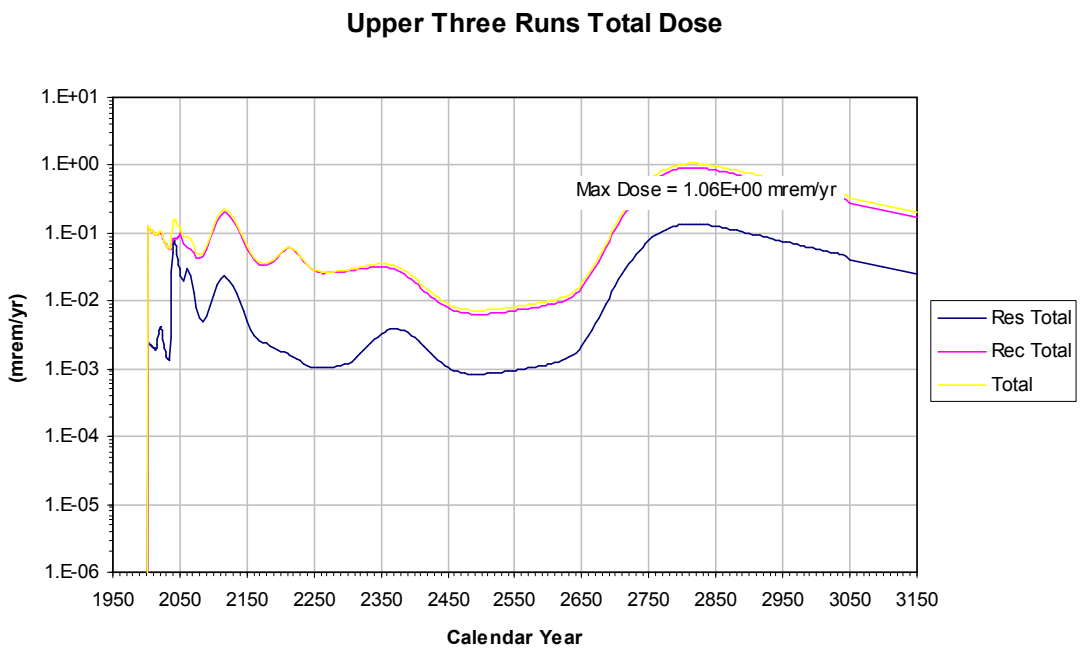


Figure 9-115. Total Direct Dose to UTR POA

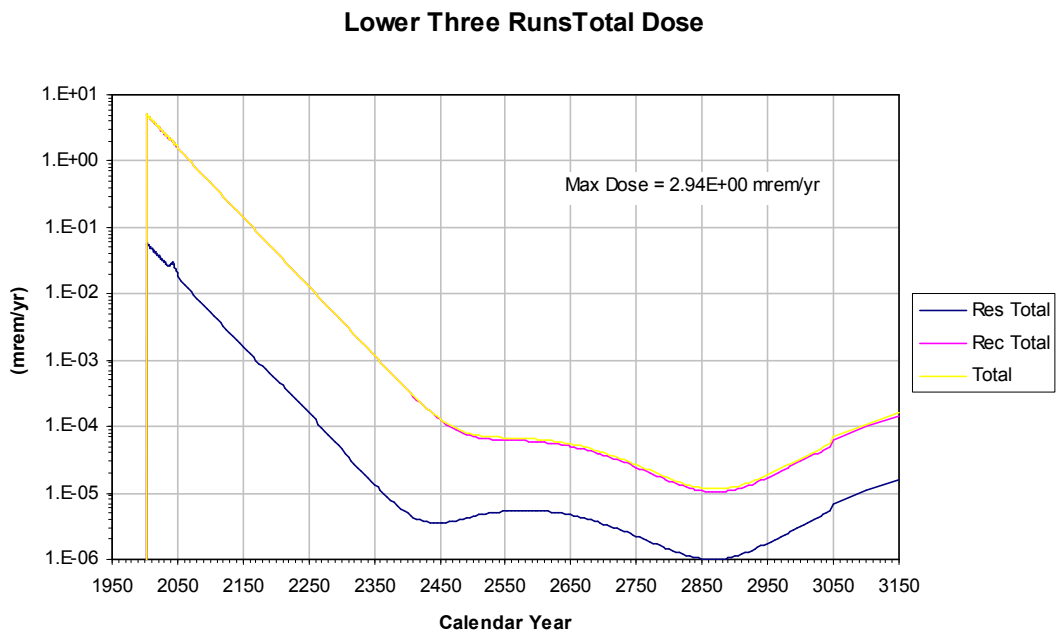


Figure 9-116. Total Direct Dose to LTR POA

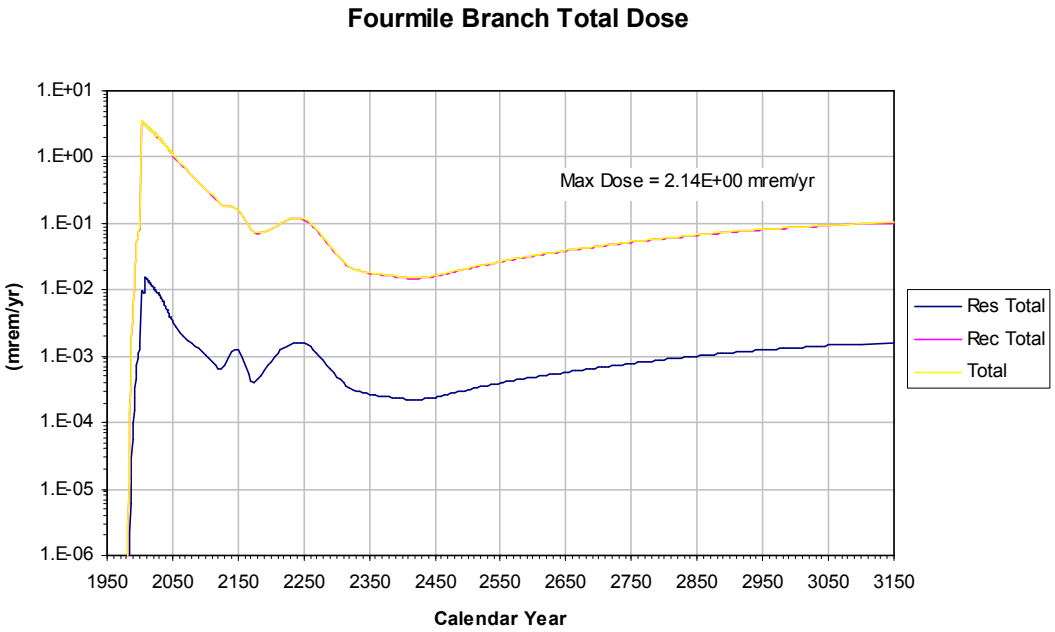


Figure 9-117. Total Direct Dose to FMB POA

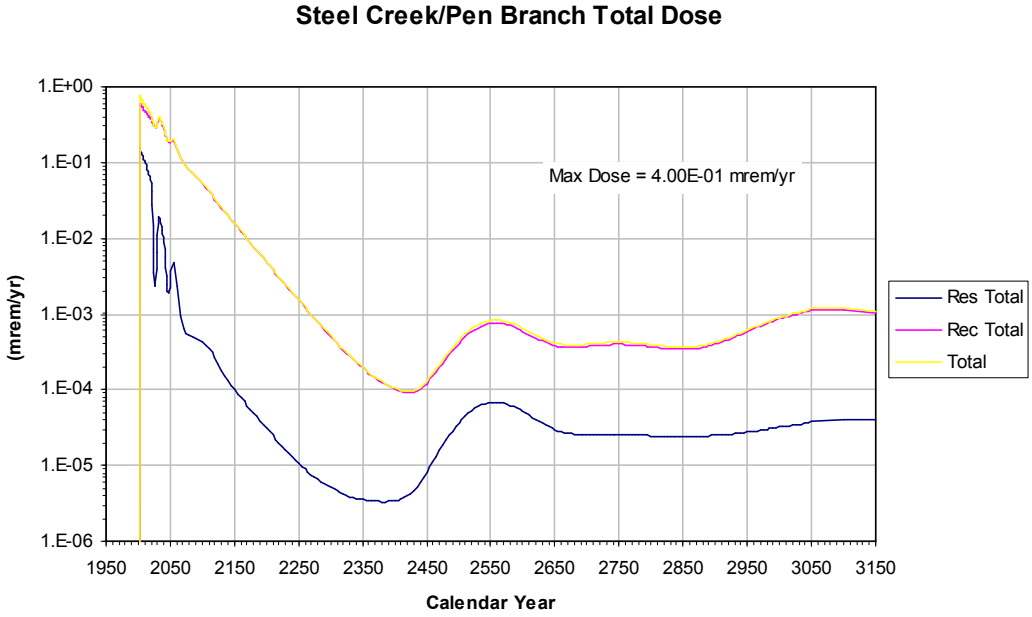


Figure 9-118. Total Direct Dose to SC/PB POA

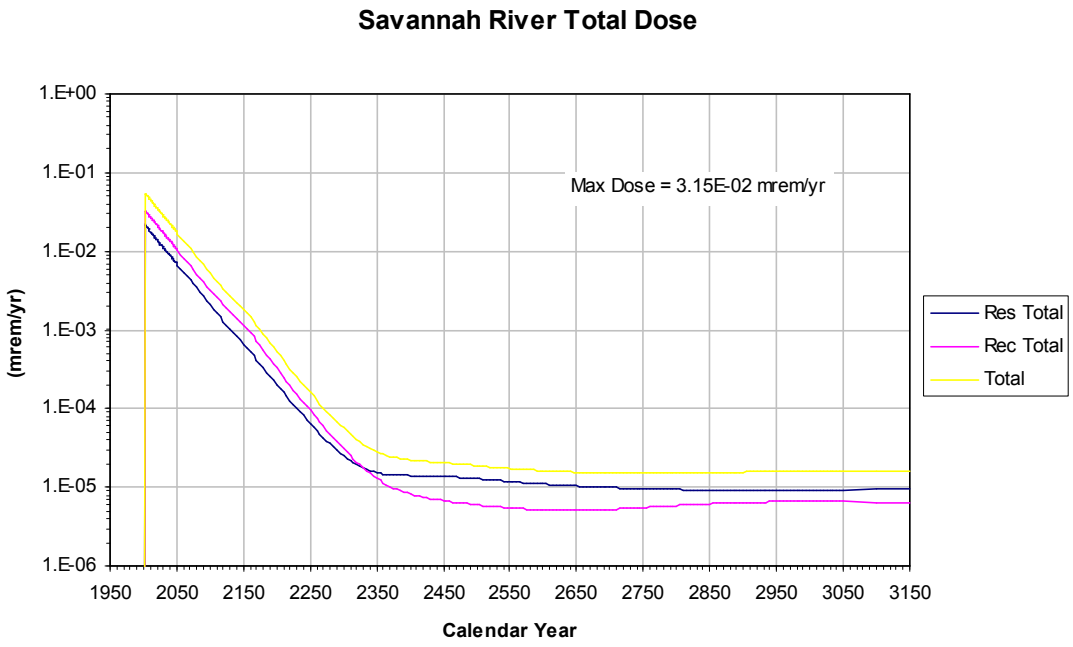


Figure 9-119. Total Direct Dose to Savannah River

9.3.9 Summary of Base Case Deterministic Modeling

The SRS CA model used to perform deterministic base case calculations, and the results of those calculations, have been described in this section. The results represent the deterministic analysis of 152 different sources at five POAs. Doses were calculated at each POA assuming recreational use of stream water and residential use of river water. Overall results from the analysis are summarized in Table 9-31 below. The first two columns in Table 9-31 show the sum of maximum doses from each source at each POA (regardless of the time when the maximum was reached) and the maximum dose from direct discharges to each POA. Dose values in the last column of Table 9-31 show the cumulative dose at each POA accounting for upstream doses in the Savannah River. Figure 9-120 shows a site map with the location of the POAs indicated along with the maximum total dose at each POA.

Table 9-32 gives a summary of the results showing the 29 sources that produced doses greater than 0.01 mrem/yr grouped by POA and sorted by increasing dose. Table 9-33 shows the 17 major sources having doses greater than 0.05 mrem/yr. This table is included to better focus on the few sources (about 10% of all sources) that have the greatest impact on dose. Cells in Table 9-32 and Table 9-33 shaded orange indicate doses greater than 1.0 mrem/yr while cells shaded yellow indicate doses greater than 0.1 mrem/yr. Table 9-34 gives a summary of the results showing the radionuclides that produced doses greater than 0.01 mrem/yr at each POA. At the UTR POA, about 65% of the maximum dose is caused by Np237 originating from H-Canyon (including HB-Line). At all of the other points of assessment, the maximum dose is primarily caused by Cs137 released from stream and river sediments (IOUs). The highest estimated dose of 3.0 mrem/yr occurs at the LTR POA from Cs137 released from the stream bed. This dose is 10% of the 30 mrem/yr constraint. In fact, Table 9-33 shows that seven of the 17 highest doses come from the IOUs. Table 9-33 also shows that contamination on surfaces in the reactor buildings accounts for another five of the 17 highest doses. Cl36 is the radionuclide primarily causing the maximum surface contamination doses. All of the surface contamination sources were analyzed using the conservative generic release directly to the vadose zone. The remaining five major dose sources were: H3 from the HANM, I129 from the HSB, I129 from the ORWBG, I129 from the MWMF, and Np237 from H-Canyon/HB-Line.

Table 9-35 gives a summary of the results showing the dose pathways that produced doses greater than 0.01 mrem/yr. In all cases, the dominant contribution to total dose came from the recreational dose, based on radionuclide concentrations at the stream mouth, rather than the residential dose, based on the much more dilute river concentrations. The dose pathway giving the most significant contribution to the recreational dose was ingestion of fish. The dose for direct discharge into the Savannah River was the only case where the recreational and residential doses were comparable (see Figure 9-119). In Table 9-32 through Table 9-35, the percent of total given below the sum of the maximum doses at each POA is the percent of the sum of all maximum doses at that POA that the number above accounts for. While this is not a perfect measure, since adding peak doses does not account for differences in timing of the peaks, the percent of the total gives an estimate of the total dose accounted for.

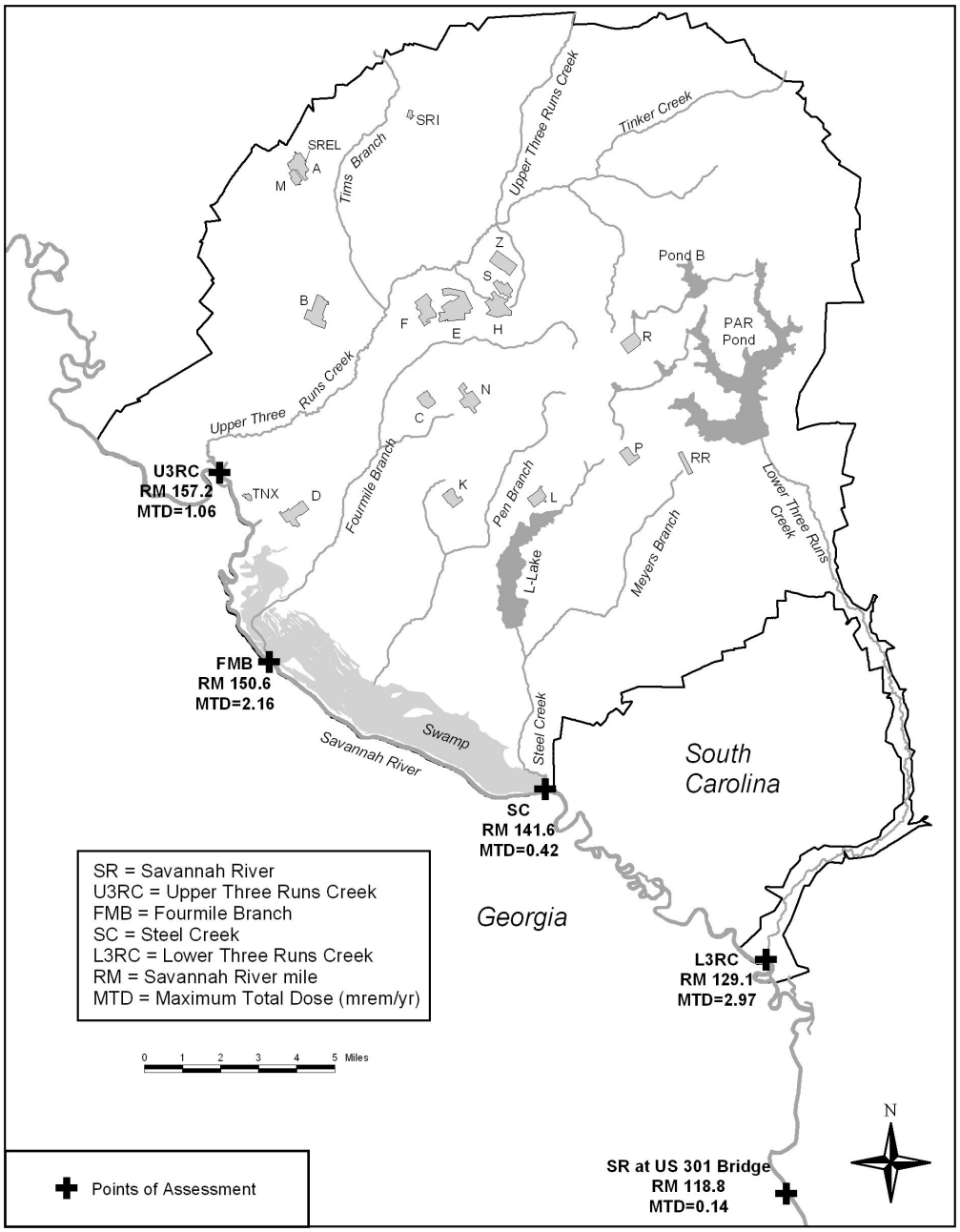


Figure 9-120. Site Map Showing CA Points of Assessment and Maximum Total Doses

Figure 9-121 plots the total cumulative doses at each POA over the somewhat longer time period from the start of SRS operations to 3950. This plot shows that the total dose is declining immediately beyond the CA assessment period. In the year 2210 the PA facilities contribute 55% of the total dose at UTR which is the largest percentage during the CA period of assessment. By 3025 the contribution to the total dose at UTR from the PA facilities has fallen to about 4%. In the year 2310 the PA facilities contribute about 26% of the total dose at FMB which is the maximum percentage during the CA period of assessment. By 3025 the contribution to total dose at FMB from the PA facilities is less than 1%.

Table 9-31. Maximum Doses at Each POA

Point of Assessment	Sum of Direct Maximum Doses [1] mrem/yr	Direct Maximum Dose [2] mrem/yr	Cumulative Maximum Dose [3] mrem/yr
Upper Three Runs	1.58	1.06	1.06
Fourmile Branch	2.34	2.14	2.16
Steel Creek/Pen Branch	0.57	0.40	0.42
Lower Three Runs	2.99	2.94	2.97
Savannah River	0.03	0.03	0.14[4]

[1] Sum of direct maximum doses from each source at each POA (regardless of the time when the maximum was reached)

[2] Maximum dose from direct discharges to each POA accounting for timing of release from each source

[3] Cumulative maximum dose at each POA accounting for both timing of release from each source and cumulative impact including all upstream SRS sources

[4] Recreational dose is calculated only from sources that discharge directly to the Savannah River via groundwater, while the residential dose is the accumulation of all sources in the Savannah River. Total cumulative dose for both recreational and residential scenarios was performed as a sensitivity as outlined in Sections 9.4.2.3.4 and 9.4.2.3.5. This note is also applicable to the Savannah River maximum total dose (MTD) shown in Figure 9-120 and Figure 9-121.

Table 9-32. Composite Analysis Sources Giving Doses > 0.01 mrem/yr

Source Name	Max Dose mrem/yr	Source Name	Max Dose mrem/yr
232-H	1.11E-02	LLRWDF FMB	1.79E-02
F-Area Old Seep	1.16E-02	105-C Surf	5.75E-02
E-Area ST Cent	1.33E-02	H-Area Seep	1.03E-01
264-2H	1.48E-02	ORWBG AgI	1.63E-01
LLRWDF UTR	1.80E-02	FMB IOU	1.97E+00
HAOM	2.02E-02	Fourmile Branch	2.31E+00
E-Area CIG	2.09E-02		99% of Total
TPBAR	2.44E-02		
NR Pad 2	3.65E-02	105-P Con	1.54E-02
Z-Area Vault 4	4.93E-02	KAC Surf	5.30E-02
HANM	6.07E-02	PB IOU	6.00E-02
UTR IOU	7.19E-02	LAC Surf	7.57E-02
MWMF	1.35E-01	105-P Surf	1.14E-01
H-Canyon	1.04E+00	SC IOU	2.43E-01
Upper Three Runs	1.53E+00	Steel Creek/Pen Branch	5.61E-01
	97% of Total		98% of Total
105-R Surf S	4.68E-02	SR-B IOU	1.38E-02
LTR IOU	2.94E+00	SR-A IOU	1.76E-02
Lower Three Runs	2.99E+00	Savannah River	3.14E-02
	99% of Total		98% of total

Table 9-33. Composite Analysis Sources Giving Doses > 0.05 mrem/yr.

Source Name	Max Dose mrem/yr	Source Name	Max Dose mrem/yr
		KAC Surf	5.30E-02
HANM	6.07E-02	PB IOU	6.00E-02
UTR IOU	7.19E-02	LAC Surf	7.57E-02
MWMF	1.35E-01	105-P Surf	1.14E-01
H-Canyon	1.04E+00	SC IOU	2.43E-01
Upper Three Runs	1.31E+00	Steel Creek/Pen Branch	5.46E-01
	83% of Total		95% of Total
105-R Surf S	4.68E-02	SR-B IOU	1.38E-02
LTR IOU	2.94E+00	SR-A IOU	1.76E-02
Lower Three Runs	2.99E+00	Savannah River	3.14E-02
	99% of Total		98% of Total
105-C Surf	5.75E-02		
H-Area Seep	1.03E-01		
ORWBG AgI	1.63E-01		
FMB IOU	1.97E+00		
Fourmile Branch	2.29E+00		
	98% of Total		

Table 9-34. Radionuclides Giving Doses > 0.01 mrem/yr

Radionuclide	Max Dose mrem/yr	Radionuclide	Max Dose mrem/yr
Nb93m	2.30E-02	Cl36	5.74E-02
Nb94	3.47E-02	I129	1.64E-01
Tc99	4.80E-02	Cs137	1.97E+00
Cs137	7.19E-02	Fourmile Branch	2.19E+00
H3	9.15E-02		94% of Total
I129	1.49E-01		
Np237	1.05E+00	I129	1.24E-02
Upper Three Runs	1.47E+00	H3	1.94E-02
	93% of Total	Cl36	1.16E-01
		Cs137	3.03E-01
Cl36	4.50E-02	Steel Creek/Pen Branch	4.51E-01
Cs137	2.94E+00		79% of Total
Lower Three Runs	2.99E+00		
	99% of Total	Cs137	3.15E-02
		Savannah River	3.15E-02
			99% of Total

Table 9-35. Dose Pathways Giving Doses > 0.01 mrem/yr

Dose Pathway	Max Dose mrem/yr	Dose Pathway	Max Dose mrem/yr
Milk Ingestion	1.33E-02		
Water Ingestion	4.20E-02		
Vegetable Ingestion	9.43E-02	Vegetable Ingestion	1.07E-02
Fish Ingestion	9.24E-01	Fish Ingestion	3.80E-01
Upper Three Runs	1.07E+00	Steel Creek/Pen Branch	3.91E-01
	68% of Total		68% of Total
Beef Ingestion	1.05E-02		
Vegetable Ingestion	1.90E-02		
Fish Ingestion	2.90E+00	Fish Ingestion	1.91E-02
Lower Three Runs	2.93E+00	Savannah River	1.91E-02
	98% of Total		60% of Total
Fish Ingestion	2.13E+00		
Fourmile Branch	2.13E+00		
	91% of Total		

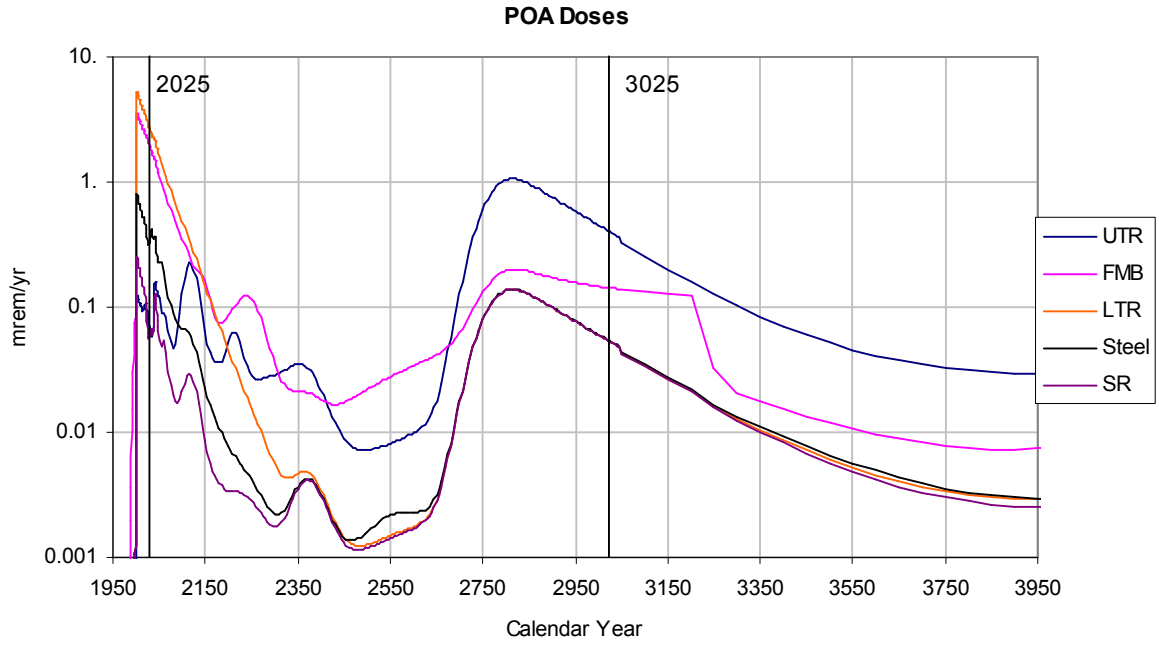


Figure 9-121. Total Cumulative Dose at Each POA from 1950 to 3950

9.4 SENSITIVITY AND UNCERTAINTY

In this section, results from sensitivity and probabilistic calculations designed to provide a measure of the uncertainty in the nominal deterministic dose results are presented. The discussion and data presented were taken largely from *Savannah River Site Composite Analysis: Sensitivity and Uncertainty Calculations* (Smith et al. 2009b). Our working definition of sensitivity analysis was testing the model response to discrete variations in model parameters where a distribution of possible values either could not be defined or did not exist. Similarly, we defined uncertainty analysis as testing the model response to random variations in parameters where a mean value and stochastic distribution were defined.

The CA GoldSim™ model, which was used for the deterministic calculations described in Section 9.3 and by Smith et al. (2009a), was also used in the sensitivity calculations (GTG 2007). A variation of this generic model that combined up to five sources into a single calculation, but was otherwise identical in structure to the generic model, was used to perform the probabilistic calculations.

9.4.1 Sensitivity Scenario Selection

The sensitivity scenarios were chosen to address the requirements in the LFRG Manual (DOE 2008) to consider factors such as release rates, radionuclide inventories, alternative points of assessment, groundwater divides, stream flow variation, and alternative disposal actions in sensitivity and uncertainty analysis (Phifer et al. 2008a). The CA Core Team concurred with the following set of CA sensitivities representing changes from the base case deterministic model (i.e., the combined generic and source release models) that were evaluated:

- Source Release Model Sensitivity: Comparison of Generic Model results to Source Release Model results for those facilities/waste sites that have both Generic Model results and Source Release Model results.
- Alternate Industrial POA Sensitivity: Both recreational and residential exposure scenarios were considered for all sources that discharge to UTR and FMB using flow rates measured near the industrial area. These flow rates are upstream of the point where the streams discharge into the Savannah River and are therefore lower. The alternative POA includes all sources that discharge into the stream that the alternative POA is within, even if the source is downstream of the POA, because the public would have access to the entire stream from the POA to the stream mouth. Because dose is directly proportional to concentration which is inversely proportional to flow, it was possible to calculate doses at the alternative POAs without performing additional GoldSim™ runs.

- Low-Flow Sensitivity #1: Low flow rate sensitivities were conducted using the seven-day period low flow with 10 year recurrence (7Q10) flow rates provided by Jones (2009) with the recreational exposure scenario using the stream flow rates and the residential exposure scenario using the Savannah River flow rates. Because dose is directly proportional to concentration which is inversely proportional to flow, it was possible to calculate doses from different flow rates without performing additional GoldSim™ runs.
- Low-Flow Sensitivity #2: Low flow rate sensitivities were conducted to determine the flow rates that would be required to produce a dose of 30 mrem/yr:
  1. For the combined recreational and residential exposure scenario at the UTR and FMB POAs using a single flow rate (in conjunction with the Alternative POA sensitivity).
  2. For the recreational exposure scenario separately at each POA, and residential exposure scenario at the U.S. Highway 301 Bridge POA, where the radionuclides from all sources are mixed together.
- Alternative Recreational Exposure Scenario Sensitivity: Evaluate the recreational exposure scenario in the Savannah River, rather than in the streams, using a flow rate of 10,175 cfs at the U.S. Highway 301 Bridge POA. This scenario combines all of the input from upstream POAs at the U.S. Highway 301 Bridge POA. Because dose is directly proportional to concentration which is inversely proportional to flow, it was possible to calculate doses at this alternative POA without performing additional GoldSim™ runs.
- Inventory Sensitivity #1: Evaluate alternatives relative to the assumptions associated with the inventories of specific major facilities, where significant decisions regarding facility disposition remain to be made. This study was made for the following facilities:
  - FAMS: The CA Base Case deterministic model assumed an end-state inventory that is 1% of the Safety Analysis Report (SAR) inventory for FAMS. Two FAMS inventory sensitivities were run: a) an end-state inventory that is 10% of the SAR inventory; and b) an end-state inventory that is 100% of the SAR inventory.
  - H-Canyon and HB-Line: The CA Base Case deterministic model assumed an end-state inventory that is 1% of the Safety Analysis Report (SAR) inventory for H-Canyon and HB-Line. One H-Canyon and HB-Line inventory sensitivity was run: an end-state inventory that is 10% of the SAR inventory.
  - TRU Pad 1: The CA Base Case deterministic model assumed an end-state inventory that is 100% of the current inventory stored on TRU Pad 1 even though the current plans are to ship this waste to the Waste Isolation Pilot Plant (WIPP) (see Section 3.4.5.1). Two TRU Pad 1 inventory sensitivities were run: a) Assume that 100% of the inventory is shipped to WIPP (i.e., no inventory); and b) Assume that 100% of the inventory is transferred to HB-Line, which is considered to be a part of H-Canyon for purposes of the CA (i.e., relocate the inventory on SRS).

- Inventory Sensitivity #2: This sensitivity determined the inventory required to produce doses of 30 mrem/yr for individual source locations within each POA while keeping all of the other sources for that POA at the base case inventory. This sensitivity was performed for the source locations with the highest calculated doses associated with each POA. This sensitivity was performed from the most significant to the least significant source location, in terms of dose, until the inventory factor exceeded 1,000, at which point no more source locations were evaluated for that POA. However, this sensitivity was done for all PA facilities (i.e., ELLWF, FTF and HTF, and SDF). Because dose is directly proportional to inventory, it was possible to calculate doses from different inventories without performing additional GoldSim™ runs. One exception to this approach was the solubility limited dose from AgI in the ORWBG, which did not have a linear response to changes in inventory once the solubility limit was reached.
- Groundwater Divide Sensitivity: Evaluate the impact of sources located near the groundwater divide within the GSA discharging to the other stream (either UTR or FMB). This analysis was performed for the MWMF, FTF, and HTF.
- Alternative End State Date Sensitivity: Evaluate the impact of the site end-state date being 2050 rather than 2025 as currently assumed in the CA Base Case deterministic model.
- C14 Fish Bioaccumulation Factor (BCF) Sensitivity: Run the base case deterministic model (combination of Generic and Source Release Models) with a C14 BCF for top predatory fish of 4,500 L/kg (NCRP 1985) and 50,000 L/kg (Lee and Coffield 2008 from Staven et al. 2003) rather than the 3 L/kg (Hinton et al. 2009) used in the CA Base Case deterministic model.
- Clay Sensitivity: The absence of clay was evaluated as a sensitivity variable by taking a limited number of sources with either the longest aquifer clay length or the greatest dose impact and clay in the aquifer flow path and making the entire aquifer zone sand.
- Maximum Calculated Dose Sensitivity: The maximum calculated dose and the time of the maximum beyond the 1,000 year period used for the CA assessment period was determined regardless of the time at which the maximum occurred, within reason. The ELLWF PA determined the maximum dose at a 100 m well from 35,000 years (STs) to 100,000 years (ILV). The FTF PA maximum dose at a 100 m well was determined over 40,000 years and the SDF maximum dose at a 100 m well was determined over 40,000 years. These PA times were evaluated to determine a reasonable cut off time.

9.4.2 Sensitivity Analysis

9.4.2.1 Linearity Test

In anticipation of conducting the sensitivity analyses listed in Section 9.4.1, it was necessary to establish that, as expected, doses calculated using the CA GoldSim™ model vary linearly with changes in the source inventory and simultaneous changes in stream and river flow rates. That is, changing either the total inventory or both stream and river flow rates by a fixed factor should give a proportional change in the calculated total dose. The total dose should be directly proportional to inventory and inversely proportional to changes in stream and river flow. The recreational dose should be linear in stream flow and the residential dose linear in river flow. The only CA calculation for which linearity with inventory does not apply is the dose from silver iodide adsorbed on berl saddles in the Old Radioactive Waste Burial Grounds (ORWBG). This source was created specifically to handle I129 absorbed on berl saddles which makes up most of the I129 inventory in the ORWBG. It is assumed that this I129 is in the form of AgI and that the release of AgI is solubility limited. Therefore, increasing the I129 inventory in this source will not increase the concentration of I129 in the groundwater once the solubility limit is reached in the waste zone. Higher inventories of I129 would reach a maximum dose and level off until the inventory is depleted. The maximum dose would not increase above the solubility limit allowed dose. For purposes of the sensitivity studies described in Section 9.4.1, it was assumed that the dose from this source is proportional to the inventory of I129 which is a conservative assumption. The only other deviation from linear behavior for other sources could possibly occur if some nonlinearity were introduced in the dose calculation module which was developed separately from the transport model under subcontract to Neptune, Inc. If the linear behavior holds, many of the sensitivity analyses can be performed by using existing results from the base case calculations without making additional GoldSim™ model runs.

To test that the calculated dose is indeed linear in inventory and stream and river flow, the following trial cases were performed using the H-Canyon source release model:

- Nominal inventory with nominal stream and river flows
- Ten times the nominal inventory with nominal stream and river flows
- One tenth the nominal inventory with nominal stream and river flows
- Nominal inventory with ten times the nominal stream and river flows
- Nominal inventory with one tenth the nominal stream and river flows

Total doses calculated for these five cases are shown in Figure 9-122 and Figure 9-123. Figure 9-122 shows the variation in total dose with inventory where the expected directly proportional linear behavior is found. Similarly, Figure 9-123 shows the variation in total dose with changes in stream and river flow. Again the expected inverse linear response is observed. Calculation of dose ratios from the GoldSim™ results in an Excel spreadsheet confirmed that the ratios were identically 10.0 and 0.1 to at least five significant figures for nominal doses over the range 1.6E-89 and 3.6E-03 mrem/yr. It is concluded that the analysis performed here sufficiently demonstrates a linear response of total dose to variations in inventory and simultaneous variations in stream and river flow. As noted above, the only exception to this behavior will be the solubility limited dose from silver iodide in the ORWBG which will not have a linear response to changes in inventory once the solubility limit is reached.

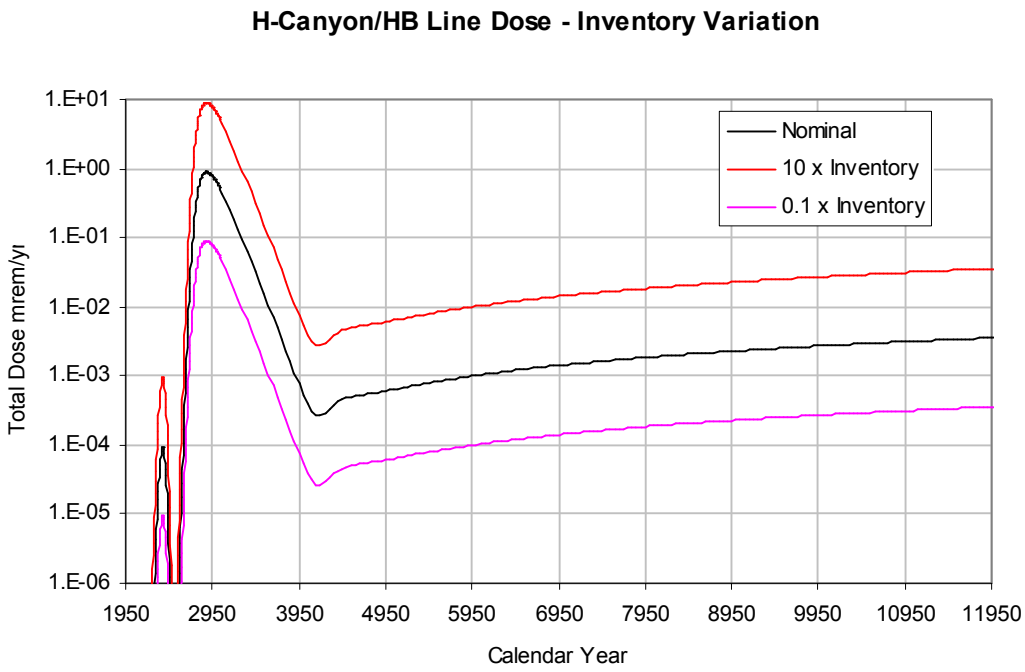


Figure 9-122. Demonstration of Linearity in Dose Response to Changes in Inventory

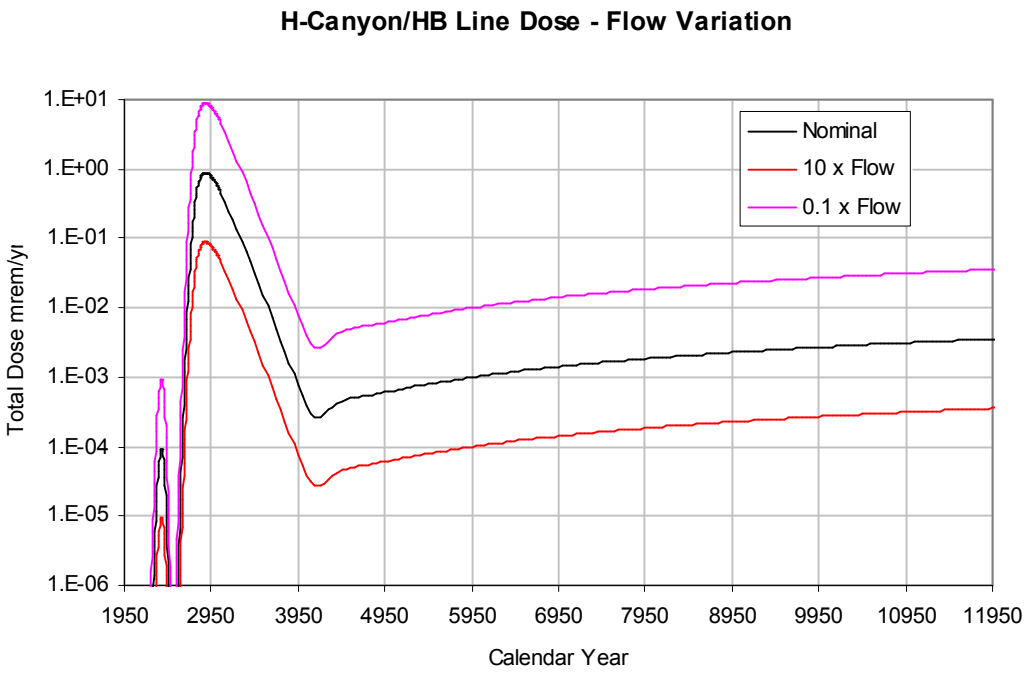


Figure 9-123. Demonstration of Linearity in Dose Response to Changes in Stream and River Flow

9.4.2.2 Generic and Source Release Results

As described in Section 9.3.6 (and by Smith et al. 2009a), all CA sources were initially evaluated using what was termed generic release modeling where the entire inventory was instantaneously released into the top cell of the vadose zone. Any source producing a total dose greater than 0.1 mrem/yr with generic release modeling was rerun using a more detailed model of the actual source release mechanism. A comparison of results obtained using the generic release model to those obtained using source release modeling for those sources where both types of calculations were performed provides a measure of the sensitivity to applying source release modeling. Table 9-23 in Section 9.3.7 gives a comparison of the maximum total doses during the CA assessment period obtained with both types of modeling. The predominant radionuclides listed are those that gave a dose greater than 0.1 mrem/yr with generic release modeling or, if no radionuclide met this criterion, the radionuclide giving the highest dose with generic release modeling. In general, source release modeling gave the following results:

- Using radionuclide fluxes to the water table from PA calculations for the E-Area ILV, E-Area LAWV and Z-Area Vault 2 (sources 1, 2 and 5 in Table 9-23), which will account for the waste forms and barriers, produced a significant decrease in projected dose.
- Accounting for metal corrosion in reactor vessel stainless steel and the E-Area NRCDA Pad 1 (sources 8 – 13 in Table 9-23) produced a significant decrease in projected dose. The remaining dose in the reactor area buildings was primarily from surface contamination.
- Accounting for tritium diffusion out of concrete pads from tritium area building slabs (sources 15 – 19 in Table 9-23) produced a significant decrease in projected dose.
- Accounting for reduction in the infiltration through engineered caps alone (sources 3, 4, 6, 7 and 14 in Table 9-23) produced relatively small reductions in maximum CA doses, particularly for long-lived radionuclides. In two cases (sources 4 and 7 in Table 9-23) the maximum dose during the CA assessment period increased. This behavior was caused by the dose from mobile isotope I129 which peaked prior to the start of the CA assessment period in 2025. Accounting for the delay in transport time from cap placement shifted the peak dose to a later time causing more of the leading edge of the dose to fall within the CA assessment period thereby increasing the maximum CA dose.

The total doses given in Table 9-23 simply sum the maximum doses listed for each source irrespective of the POA or the timing of the maximum. As such, the totals only give a rough measure of the impact of applying source release modeling. In general, the differences between results from generic source release modeling and results from the detailed source release modeling are significant and using the generic source release approach substantially overestimates the resulting doses.

9.4.2.3 Stream and River Flow

The CA sensitivity studies documented in this section of the report were performed using output from the deterministic GoldSim™ CA Model runs described in Sections 9.3.6, 9.3.7, and 9.3.8 documented by Smith et al. (2009a). Based on the demonstrated linear response of dose to changes in stream and river flow (see Section 9.4.2.1), no additional GoldSim™ CA Model runs were required for the following sensitivity studies. For sensitivity cases involving predetermined flow changes, base case CA doses were multiplied by the ratio of the base case flows to the alternative flows. For sensitivity studies determining the flow required to produce a dose of 30 mrem/yr, base case CA doses were multiplied by appropriate factors and the Microsoft ® Excel *Goal Seek* function was used to calculate the flow multiplier giving the 30 mrem/yr dose. Nominal average stream and river flow rates used in the base case calculations are shown in Table 9-17 in Section 9.3.4.5 (Jones 2009). Alternate flow rates used in each sensitivity study are given in the subsections where they apply.

Results from all of the stream and river flow sensitivity studies are summarized and compared to corresponding base case results in Table 9-42 in Section 9.4.2.3.5.

9.4.2.3.1 Alternative Industrial POA

Both recreational and residential exposure scenarios were considered for all sources that discharge to UTR and FMB using the lower flow rates shown in Table 9-36 (Jones 2009) for the alternative UTR (02197310, UTR just downstream of F-Area) and FMB (02197342, FMB just downstream of C-Area) industrial POAs shown in Figure 9-124. The alternative POAs were chosen at the boundary of the site industrial zone consistent with the approach described in *Comprehensive SRS Composite Analysis Program Points of Assessment and Assessment Period* (Phifer and Wilhite 2008, Section 3.5). The analysis included the dose from all sources that discharge into the stream that the alternative industrial POA is on, even if the source is downstream of the POA, because the public would have access to the stream from the POA to the stream mouth. Dose is directly proportional to concentration which is inversely proportional to stream/river flow. Therefore, doses at the alternate POAs required no additional CA model runs.

Table 9-36. Average Annual Flow Rates at Alternative POA Locations

Stream	Station Number and Location	Average Annual Flow Rate (cfs)
Upper Three Runs (near Industrial Area)	02197310 (UTR just downstream of F-Area)	209
Fourmile Branch (near Industrial Area)	02197342 (FMB just downstream of C-Area)	16

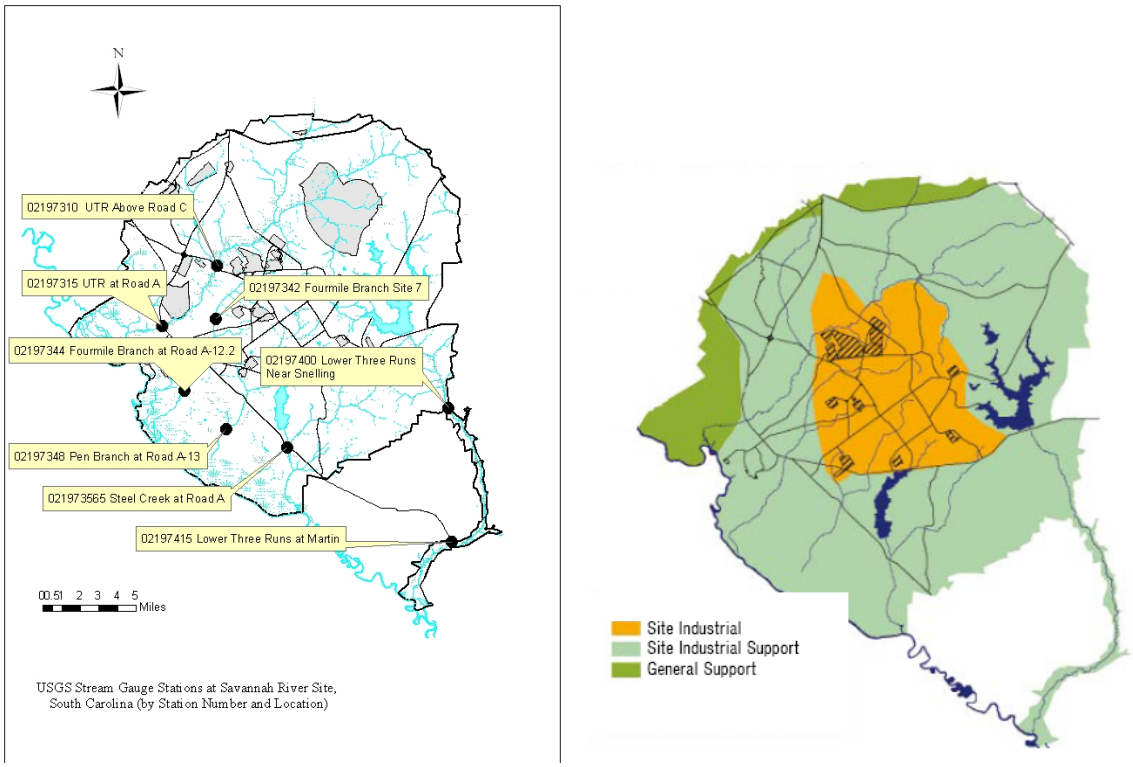


Figure 9-124. Alternative UTR and FMB Industrial POAs

Nominal doses for the alternate POAs are given in Table 9-37. The results provided are the maximum doses for the 1,000-year CA assessment period, defined as from year 2025 to year 3025. The residential doses in Table 9-37 were calculated using the stream flow (not river flow) and are therefore significantly higher than the corresponding base case results. Maximum total cumulative doses for the base case are given in Table 9-31. Though not shown here, doses were also calculated during the site operational period prior to 2025 and for 10,025 years past the end-state date of 2025. In no case did the maximum total dose reach 30 mrem/yr throughout the entire period analyzed (1950 to 12050). The maximum total dose in Table 9-37 may not necessarily correspond in time with either the maximum residential dose or the maximum recreational dose, and therefore may not correspond to the sum of the maximum residential dose and the maximum recreational dose. However, in this case, the maximum doses all occur simultaneously.

Table 9-37. Maximum Doses Evaluated at Alternative POA Locations

Alternate POA Stream	Max Residential Dose		Max Recreational Dose		Max Total Dose	
	(mrem/yr)	Year	(mrem/yr)	Year	(mrem/yr)	Year
Upper Three Runs	6.07	2810	1.05	2810	7.12	2810
Fourmile Branch	5.47	2025	4.27	2025	9.74	2025

9.4.2.3.2 Low Stream and River Flow

A low flow rate sensitivity study was conducted using the 7Q10 flow rates shown in Table 9-38 (Jones 2009). As before, the recreational exposure scenario was calculated using the stream flow rates and the residential exposure scenario using the Savannah River flow rates. Dose is directly proportional to concentration which is inversely proportional to stream/river flow. For the purposes of this sensitivity study, groundwater transport is assumed to be unaffected by the conditions which lead to the low flow in the streams and river. In reality, the conditions which lead to reduced stream and river flows over a reasonably long period of time might also be expected to reduce groundwater flows and therefore slow the transport of contaminants to the seep. This action would serve to further reduce radionuclide concentrations in the streams. By making these assumptions, this sensitivity study can be conducted with no additional CA model runs and would also be expected to be conservative.

Table 9-38. 7Q10 Stream and River Flow Rates

Stream Name	Location	7Q10 Flow Rate for Low-Flow Sensitivity Modeling (cfs)
Savannah River	Augusta, GA[1]	4,010
Savannah River	U.S. Highway 301 Bridge[2]	4,360
Upper Three Runs	Highway 125	102
Fourmile Branch	Near D-Area	8.31
Steel Creek-Pen Branch	Highway 125/Near Savannah River Swamp	12.88
Lower Three Runs	Martin, SC[3]	48.02

[1]Flow used for all POAs except for U.S. Highway 301 Bridge POA.
[2]Flow used for U.S. Highway 301 Bridge POA.
[3]Modified with Snelling, SC data (Jones 2009).

The cumulative doses at each POA obtained using the 7Q10 low flows are given in Table 9-39. Corresponding results for the base case total dose are given in Table 9-31. The results provided are the maximum doses for the 1,000 year CA assessment period from year 2025 to year 3025. The recreational doses are calculated using concentrations at the mouth of the named stream. The residential dose calculation assumes dilution in the Savannah River and the addition of all upstream sources (e.g., residential dose for FMB would include the contribution from UTR and from sources which flow directly to the Savannah River, which for the purposes of this study, were assumed to enter the river between UTR and FMB).

Table 9-39. Maximum Cumulative Doses with 7Q10 Flow Rates

Source Stream	Max Residential Dose (mrem/yr)	Max Recreational Dose (mrem/yr)	Max Total Dose (mrem/yr)
Upper Three Runs	0.32	2.15	2.47
Savannah River	0.32	0.04	0.32
Fourmile Branch	0.32	8.21	8.27
Steel Creek-Pen Branch	0.32	2.63	2.71
Lower Three Runs	0.32	9.92	10.1

Maximum total doses in Table 9-39 do not necessarily correspond in time with either the maximum residential dose or the maximum recreational dose, and therefore may not correspond to the sum of the maximum residential dose and the maximum recreational dose.

The results of this study show that the maximum total doses under the 7Q10 low flows are all significantly less than 30 mrem/yr. For each of the source streams, with the exception of UTR, higher doses than those found during the CA assessment period were calculated before the assessment period began (i.e., doses were declining at the start of the CA assessment period, see Figure 9-132 through Figure 9-136). In no case did the maximum dose during any time period reach or exceed 30 mrem/yr.

9.4.2.3.3 Flows Required to Reach a Dose of 30 mrem/yr

In this sensitivity study, the flows required to reach a maximum total dose (including recreational and residential doses) of 30 mrem/yr at the various points of assessment were determined. The first part of this study determined the flows necessary to produce the 30 mrem/yr maximum dose at the UTR and FMB alternative POAs. The results are given in Table 9-40 which also compares the flows to the annual average and 7Q10 flows.

Table 9-40. Alternative POA Stream Flows Required to Reach 30 mrem/yr Dose Constraint

Alternate POA Stream	Stream Flow Required to Reach a Total Dose of 30 mrem/yr (cfs)	Percentage of Annual Average Flow	Percentage of 7Q10 Flow
Upper Three Runs	49.6	20.9%	48.6%
Fourmile Branch	5.19	16.2%	62.4%

The second part of this low flow sensitivity study determined the stream flow rates required to produce a dose of 30 mrem/yr from the recreational exposure scenario at each POA. The results for this study are given in Table 9-41. In this study all flows required to increase the maximum recreational dose during the assessment period are considerably less than the corresponding 7Q10 flows.

Table 9-41. POA Stream Flows Required to Reach 30 mrem/yr Dose Constraint

POA Stream	Stream Flow Required to Reach Maximum Recreational Dose of 30 mrem/yr (cfs)	Percentage of Annual Average Flow
Upper Three Runs	7.31	3.08%
Savannah River	5.86	0.06%
Fourmile Branch	2.28	7.13%
Steel Creek-Pen Branch	1.13	1.27%
Lower Three Runs	15.9	9.70%
U.S. Highway 301 Bridge	25.6	0.25%

The final low flow study determined the Savannah River flow needed to produce a maximum residential dose at the U.S. 301 Bridge of 30 mrem/yr. The necessary flow was found to be 42.6 cfs, which is more than 100 times less than the 7Q10 flow at that point on the river.

9.4.2.3.4 Alternative Recreational Exposure

This sensitivity study calculated the maximum recreational dose at the U.S. 301 Bridge under average flow conditions (see Table 9-17 in Section 9.3.4.5). The maximum recreational dose at that location was found to be 7.54E-02 mrem/yr during the 1,000-year CA assessment period. The maximum recreational dose found prior to the CA assessment period (i.e., before 2025) at this location under normal river flow conditions was about twice the maximum dose during the assessment period.

9.4.2.3.5 Stream and River Flow Sensitivity Summary

Table 9-42 provides a summary of the stream and river flow sensitivities performed relative to the base case deterministic results presented in Section 9.3, and Smith et al. (2009a). The linear response of dose to changes in stream and river flow (see Section 9.4.2.1) is clearly shown by the data in Table 9-42, where the dose is inversely proportional to the flow rate. As shown under base case and 7Q10 flow rate conditions, the dose is dominated by the recreational dose, primarily due to evaluation of the recreational dose at the stream mouth prior to mixing with the Savannah River as opposed to evaluation of the residential dose after mixing with the Savannah River. As shown in the table, the use of the 7Q10 flow rates rather than the average flow rates results in an increased dose. However, even conservatively assuming a constant yearly flow rate at the 7Q10 value at most results in a maximum total dose of approximately 10 mrem/yr at Lower Three Runs (LTR), which is one third of the 30 mrem/yr dose constraint. For the recreational doses at the mouths of the streams to reach the 30 mrem/yr dose constraint, the flow in UTR, FMB, Steel Creek/Pen Branch (SC/PB), and LTR would have to have a long-term reduction in flow to approximately 3%, 7%, 1%, and 10%, respectively, of their average flow rates. The maximum recreational, residential and total doses shown in Table 9-42 may not occur at the same time. For this reason, the sum of the maximum recreational and residential doses may not equal the maximum total dose.

Consideration of dose at alternative industrial POAs, where both the recreational and residential doses are determined based upon the stream flow rates, results in a residential dose that is slightly greater than the recreational dose, in contrast to the base case or 7Q10 doses discussed above. Also, as shown in the table, consideration of the average flow at the alternative industrial POAs at most results in a maximum total dose of approximately 10 mrem/yr at the FMB alternative industrial POA, which is one third of the 30 mrem/yr dose constraint. Additionally, the flow at the alternative UTR industrial POA would have to be reduced to about 20% of its average annual flow or roughly half its already extremely low 7Q10 flow rate; and the flow at the alternative FMB industrial POA would have to be reduced to about one sixth of its average annual flow or roughly 60% of its 7Q10 flow in order to reach the 30 mrem/yr dose constraint.

Recreational and residential doses calculated based upon Savannah River flow rates at or greater than the 7Q10 value result in total doses significantly less than 1 mrem/yr. In fact, to reach a maximum total dose equal to the 30 mrem/yr dose constraint, the flow in the Savannah River would have to be reduced to 56.8 cfs, which is 0.62 percent of the annual average Savannah River flow. Under Savannah River average flow conditions at U.S. Highway 301 Bridge, the maximum total dose from the accumulation of all SRS sources in the Savannah River is estimated to be 0.2 mrem/yr.

While dose is inversely proportional to the flow rate, the sustained reductions in flow rates from current annual average rates which would be required to result in doses at the 30 mrem/yr dose constraint, are so great as to be unrealistic. This result means that while dose is sensitive to flow rate, the flow rates are highly unlikely to be reduced to levels that would result in doses of any significant concern.

Table 9-42. Stream and River Flow Sensitivity Summary

POA	Sensitivity Case	Max Recreational		Max Residential [5]		Max Total Dose (mrem/yr)
		Flow Rate (cfs)	Dose (mrem/yr)	Flow Rate (cfs)	Dose (mrem/yr)	
Upper Three Runs	Base Case[1]	237	0.92	9,198 (SR)	0.14[6]	1.06
	7Q10 Flow	102	2.15	4,010 (SR)	0.32[6]	2.47
	Alt Industrial POA	209	1.05	209	6.07	7.12
	30 mrem/yr UTR flow at Alt Industrial POA	49.6	4.42	49.6	25.58	30
	30 mrem/yr recreational dose UTR flow	7.31	30	-	-	-
Fourmile Branch	Base Case[1]	32	2.13	9,198 (SR)	0.14[6]	2.16
	7Q10 Flow	8.31	8.21	4,010 (SR)	0.32[6]	8.27
	Alt Industrial POA	16	4.27	16	5.47	9.74
	30 mrem/yr FMB flow at Alt Industrial POA	5.19	13.15	5.19	16.85	30
	30 mrem/yr recreational dose FMB flow	2.28	30	-	-	-
Steel Creek/Pen Branch	Base Case[1]	89	0.38	9,198 (SR)	0.14[6]	0.42
	7Q10 Flow	12.88	2.63	4,010 (SR)	0.32[6]	2.71
	30 mrem/yr recreational dose SC/PB flow	1.13	30	-	-	-
Lower Three Runs	Base Case[1]	164	2.90	9,198 (SR)	0.14[6]	2.97
	7Q10 Flow	48.02	9.92	4,010 (SR)	0.32[6]	10.1
	30 mrem/yr recreational dose SC/PB flow	15.9	30	-	-	-
Savannah River	Base Case[1, 2]	9,198	0.02	9,198	0.14[6]	0.14
	7Q10 Flow[2]	4,010	0.04	4,010	0.32[6]	0.32
	30 mrem/yr recreational dose SR flow[2]	5.86	30	-	-	-
	30 mrem/yr recreational and residential doses SR flow[3]	25.6	30	42.5	30	-
	30 mrem/yr SR flow at SR-U.S. 301 Bridge	56.8	13.5	56.8	22.5	30
	Max dose SR-U.S. 301 Bridge average flow – cumulative[3,4]	10,175	0.08	10,175	0.13	0.17

[1] Smith et al. 2009a.
[2] Recreational dose is calculated from direct groundwater discharges to the Savannah River while the residential dose is the accumulation of all sources in the Savannah River.
[3] Recreational and residential dose are both the accumulation of all sources in the Savannah River.
[4] Maximum recreational dose occurs at 2025 primarily from Cs137 released from streambeds, maximum residential dose occurs at 2815 primarily from Np237 in UTR, maximum total dose occurs at 2041.
[5] Residential dose is a combination of the residential dose from the designated POA and all up-river POAs.
[6] Maximum cumulative residential doses are essentially identical at all POAs for the base case and 7Q10 flows because the largest dose occurs at UTR in the year 2815 while maximum residential doses at the other POAs occur near the beginning of the CA assessment period and have declined to small values by 2815. Therefore, since UTR is upstream of the other POAs, the maximum UTR dose at 2815 propagates downstream and becomes the maximum cumulative residential dose at all POAs.

9.4.2.4 Inventory

The CA sensitivity studies documented in this section of the report were performed using output from the deterministic GoldSim™ CA Model runs documented by Smith et al. (2009a). Based on the demonstrated direct proportionality of dose to changes in inventory shown in Section 9.4.2.1, no additional GoldSim™ CA Model runs were required for the following sensitivity studies. This approach is consistent with current Performance Assessment (PA) practice in E-Area where the flux of each radionuclide through the groundwater is assumed to be directly proportional to its inventory (WSRC 2008a). For sensitivity cases involving predetermined inventory changes, base case CA doses were multiplied by the ratio of the alternative inventories to the base case inventories. For sensitivity studies determining the inventory required to produce a dose of 30 mrem/yr, base case CA doses were multiplied by appropriate factors and the Microsoft® Excel *Goal Seek* function was used to calculate the inventory multiplier giving the 30 mrem/yr dose. In all of these studies, the inventory of each radionuclide assumed to remain within a facility was changed by the same factor.

9.4.2.4.1 F-Area Material Storage and H-Canyon/HB-Line

Alternatives exist for the inventories remaining in both F-Area Material Storage (FAMS) and H-Canyon/HB-Line at closure where significant decisions regarding facility disposition remain to be made. For FAMS, the CA Base Case inventory assumed that 1% of the SAR inventory remains at closure. The alternative cases assessed here assume 10% and 100% of the SAR inventory. The H-Canyon/HB-Line base case also assumed that 1% of the SAR inventory would remain at closure. The alternative assessed in this sensitivity analysis assumes that 10% of the SAR inventory would remain. Doses obtained for these facilities with this sensitivity analysis are given in Table 9-43. As expected, increasing the H-Canyon/HB-Line inventory by a factor of 10 increased the maximum dose proportionally to over 10 mrem/yr, which is one third of the 30 mrem/yr constraint. The FAMS facility did not produce a significant CA dose even with 100% of the SAR inventory remaining at closure.

Table 9-43. Sensitivity of Maximum Dose from FAMS and H-Canyon to Inventory

Facility and Percentage Assumed Inventory	Max Residential Dose (mrem/yr)	Max Recreational Dose (mrem/yr)	Max Total Dose (mrem/yr)
FAMS 1% (Base Case)	2.18E-07	1.45E-06	1.67E-06
FAMS 10%	2.18E-06	1.45E-05	1.67E-05
FAMS 100%	2.18E-05	1.45E-04	1.67E-04
H-Canyon/HB-Line 1% (Base Case)	0.14	0.91	1.04
H-Canyon/HB-Line 10%	1.36	9.08	10.4

9.4.2.4.2 Inventory Required to Reach a Dose of 30 mrem/yr

This sensitivity study determined the overall inventory multiplier that would be required to reach a dose of 30 mrem/yr for individual sources contributing to each POA while keeping all of the other sources for that POA at the base case inventory. A specific inventory multiplier was determined for all PA sources (i.e., ELLWF facilities, FTF, HTF, and SDF) and all other sources with an inventory multiplier of 1,000 or less. Those sources with an inventory multiplier greater than 1,000 are simply denoted as such and include all sources other than those listed in Table 9-44 and Table 9-45. The results for PA sources and those sources with an inventory multiplier of 1,000 or less are given in Table 9-44 and Table 9-45 as a number that the best estimate source inventory would have to be multiplied by to give a total dose at the POA of 30 mrem/yr. Of the 152 sources evaluated in this CA, 135 had inventory multipliers greater than 1,000; 14 had inventory multipliers between 100 and 1,000; and only the three sources highlighted in Table 9-44 and Table 9-45 (H-Canyon, FMB IOU and LTR IOU) had inventory multipliers less than 100.

Additionally the overall inventory multiplier required to produce a maximum total dose of 30 mrem/yr (i.e., the dose constraint) for the combined sources associated with each POA was also determined. The overall inventory multipliers were determined to be approximately 28, 14, 75, 10, and 950, respectively for the UTR, FMB, SC/PB, LTR, and Savannah River POA combined sources.

Table 9-44. Inventory Multipliers Required to Reach Dose of 30 mrem/yr at Upper Three Runs POA

Source	Inventory Multiplier	Source	Inventory Multiplier
E-Area ILV	1.13E+06	E-Area LAWV	1.05E+07
E-Area ST East	9.26E+04	E-Area CIG	1.43E+03
E-Area ST Cent	2.25E+03	E-Area ET	1.11E+05
E-Area ST West	2.20E+05	MWMF[2]	2.01E+02
Z-Area Vault 1	3.88E+03	H-Canyon	2.87E+01
Z-Area Vault 2[1]	1.14E+10	HANM	4.92E+02
Z-Area Vault 4	6.05E+02	UTR IOU[4]	4.17E+02
NR Pad 1[3]	5.31E+03	TRU Pad	2.49E+13
NR Pad 2	8.22E+02		
FTF Type IV	2.37E+08	FTF Type I	1.39E+07
FTF Type IIIA UTR	9.57E+06	FTF Type III	2.09E+04
HTF Type I UTR E	2.81E+05	FTF Equip	2.74E+06
HTF Type II Tanks	3.02E+05	HTF Type I UTR W	3.55E+06
HTF Type III UTR E	2.33E+06	HTF Type IV UTR	2.02E+07
HTF Type IIIA UTR E	1.10E+05	HTF Type III UTR W	6.88E+07
HTF Equip UTR	6.59E+06	HTF Type IIIA UTR W	6.77E+06
		UTR Overall Inventory (Applied to all sources)	2.82E+01

[1] Z-Area Vault 2 was divided into three separate units for the purposes of the CA Model. They have been re-combined into a single unit for this table.
[2] Includes part of LLRWDF going to UTR.
[3] Naval Reactor Pad 1 was divided into six separate units for the purposes of the CA Model. They have been re-combined into a single unit for this table.
[4] Integrator Operable Unit

Table 9-45. Inventory Multipliers Required to Reach Dose of 30 mrem/yr at Other POAs

POA	Source	Inventory Multiplier	Source	Inventory Multiplier
SR Direct Dose			SR Overall Inventory[1] (Applied to all sources)	9.53E+02
FMB	LLRWDF FMB	1.65E+03	ORWBG[2]	1.70E+02
	105-C Surf	5.20E+02	FMB IOU	1.51E+01
	FTF Type IIIA FMB	3.69E+06	HTF Type IV FMB	1.97E+05
	HTF Type III FMB	1.20E+04	HTF Type IIIA FMB	7.70E+03
	HTF Equip	5.08E+04	HTF IPSL FMB	6.66E+03
	H-Area Seep	2.91E+02		
			FMB Overall Inventory (Applied to all sources)	1.40E+01
SC/PB	KAC Surf	5.63E+02	PB IOU	4.96E+02
	LAC Surf	3.93E+02	SC IOU	1.23E+02
	105-P Surf	2.61E+02		
			SC/PB Overall Inventory (Applied to all sources)	7.51E+01
LTR	105-R Surf	6.01E+02	LTR IOU	1.02E+01
			LTR Overall Inventory (Applied to all sources)	1.02E+01

[1]No individual units had inventory multipliers less than 1,000 for the Savannah River direct dose contributions. Therefore, only the factor for all units is given.
[2]The ORWBG was divided into two units for the purposes of the CA Model (one for I129 on berl saddles and one for the remaining inventory). They have been re-combined for the purposes of this table.

9.4.2.4.3 TRU Pad 1

The CA Base Case deterministic model for TRU Pad 1 assumed an end-state inventory that is 100% of the current inventory stored on TRU Pad 1 even though the current plans are to ship this waste to WIPP (see Section 3.4.5.1). The TRU Pad 1 base case resulted in a maximum dose at the UTR POA of 1.2E-12 mrem/yr. Two inventory sensitivities were run for TRU Pad 1: a) assume that 100% of the inventory is shipped to WIPP (i.e., no inventory); and b) assume that 100% of the inventory is transferred to HB-Line, which is considered to be a part of H-Canyon for purposes of the CA (i.e., relocate the inventory on SRS). In the first sensitivity case, it was assumed that 100% of the TRU Pad 1 inventory was shipped off site. This scenario has a negligible impact on the CA dose during the period of assessment. The base case maximum dose from TRU Pad 1 between the years 2025 and 3025 is only 1.2E-12 mrem/yr because almost none of the plutonium radionuclides or daughters have migrated to the stream by that time. The second sensitivity case assumes that 100% of the TRU Pad 1 inventory is transferred to H-Canyon/HB-Line for final disposal. The aquifer path from H-Canyon to UTR (6088 ft) is shorter that the path from TRU Pad 1 (7620 ft) but the length of clay in the H-Canyon path (1123 ft) is longer than the clay region in the TRU Pad 1 path (117 ft). Because the paths are both long, the impact of placing the TRU Pad 1 inventory in H-Canyon on the CA dose is again negligible.

Figure 9-125 plots the total CA dose from TRU Pad 1, H-Canyon, and the combined inventories in H-Canyon. There is essentially no impact on the total dose until the year 4200. Beyond about the year 5000, the dose from the combined inventory remains approximately 2.24 times the dose from H-Canyon alone.

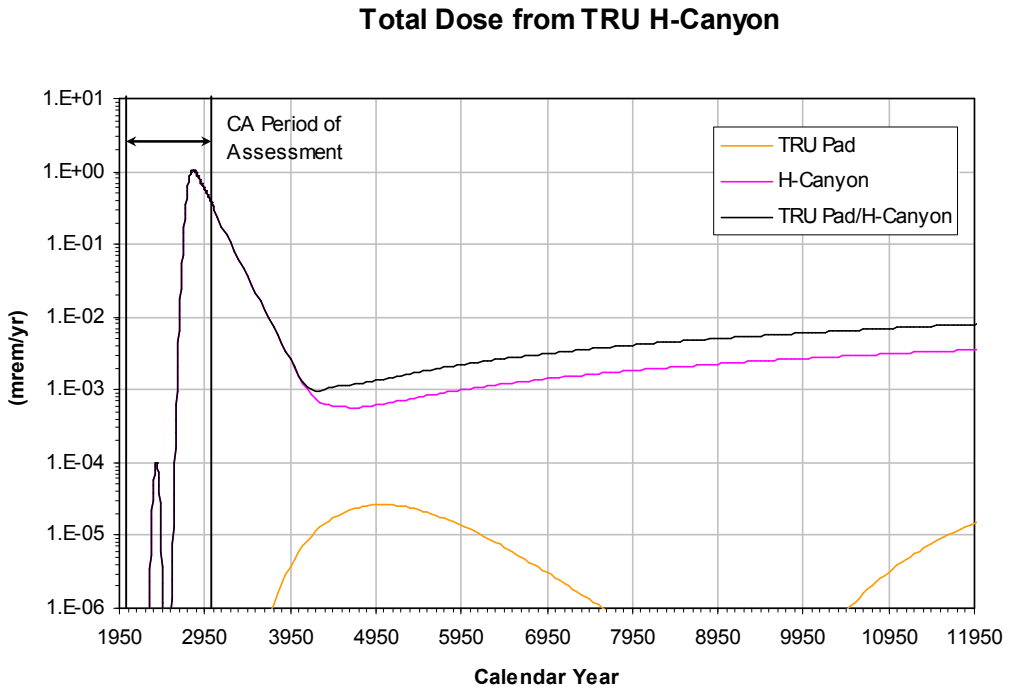


Figure 9-125. Total Dose from Alternative TRU Pad 1 Disposition Options

9.4.2.4.4 E-Area Low-level Waste Facility Completion Project

It has been proposed, as part of the ECP, to increase the E-Area low-activity radioactive waste disposal capacity by using several plots of land adjacent to the existing disposal units that are part of the disposal facilities but have not yet been developed (Jordan and Flach 2009). For purposes of this sensitivity analysis, the impact to the CA dose from using four of the proposed plots (Plots 6, 7, 7a and 8) was analyzed. Aquifer paths were generated for each of these sources as described by Hamm et al. (2009). It was assumed that the waste inventory in the proposed plots would be similar to what is currently being placed in the E-Area Central STs. The inventory in the plots was then estimated from the closure inventory for the Central STs (Hiergesell et al. 2008) by multiplying by the ratio of the adjacent plot area to that of the Central STs. Table 9-46 shows the areas and area ratios for these plots. Releases from the ECP plots were assumed to begin in the year 2025. Radionuclide releases from the ECP plots were modeled taking credit for clay caps over each plot which reduces infiltration through the waste material as described by (Smith et al. 2009a).

Table 9-46. ECP Plot Areas Used to Estimate Inventory

Plot Name	Area (sq. ft.)	Fraction of Central Slit Trench Area
Central Slit Trench	770476	1.00
Plot 6	1337870	1.74
Plot 7	203148	0.26
Plot 7a	226441	0.29
Plot 8	906662	1.18

Figure 9-126 shows the total dose from the four plots during the CA period of assessment and Figure 9-127 shows the total dose over the longer time period of 10,000 years. Figure 9-128 shows the contributions from individual isotopes to the dose from ECP Plot 8. Because the composition of the inventory is the same for all plots and the plots have similar aquifer transport paths, the radionuclides contributing to the dose follow the same pattern for all four plots. The dose peaks early in the transients at about year 2200 are caused by I129. The maximum peak doses at about year 2900 and the longer term dose increase beyond the CA assessment period are from Ra226 which is a product of uranium decay. The maximum combined dose from the four plots during the CA assessment period is 0.0115 mrem/yr. This dose is comparable to the base case dose of 0.0133 mrem/yr from the E-Area Central Slit Trench.

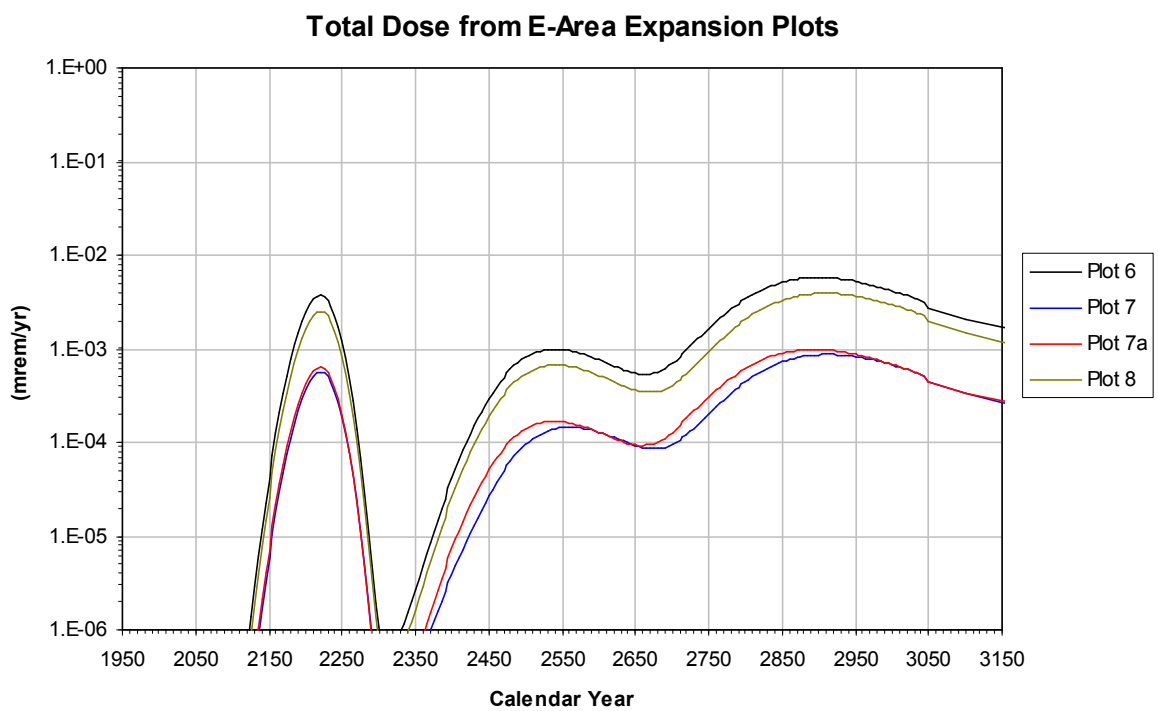


Figure 9-126. Total Doses from ECP Plots during CA Period of Assessment

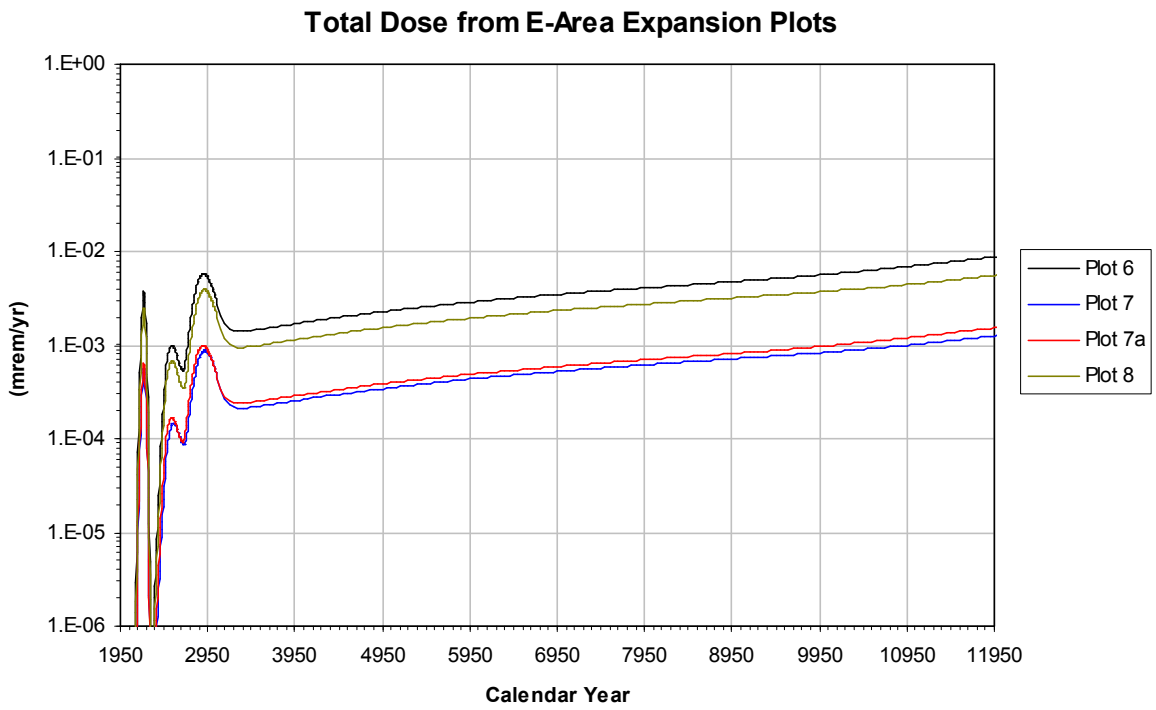


Figure 9-127. Total Doses from ECP Plots over 10,000 Years

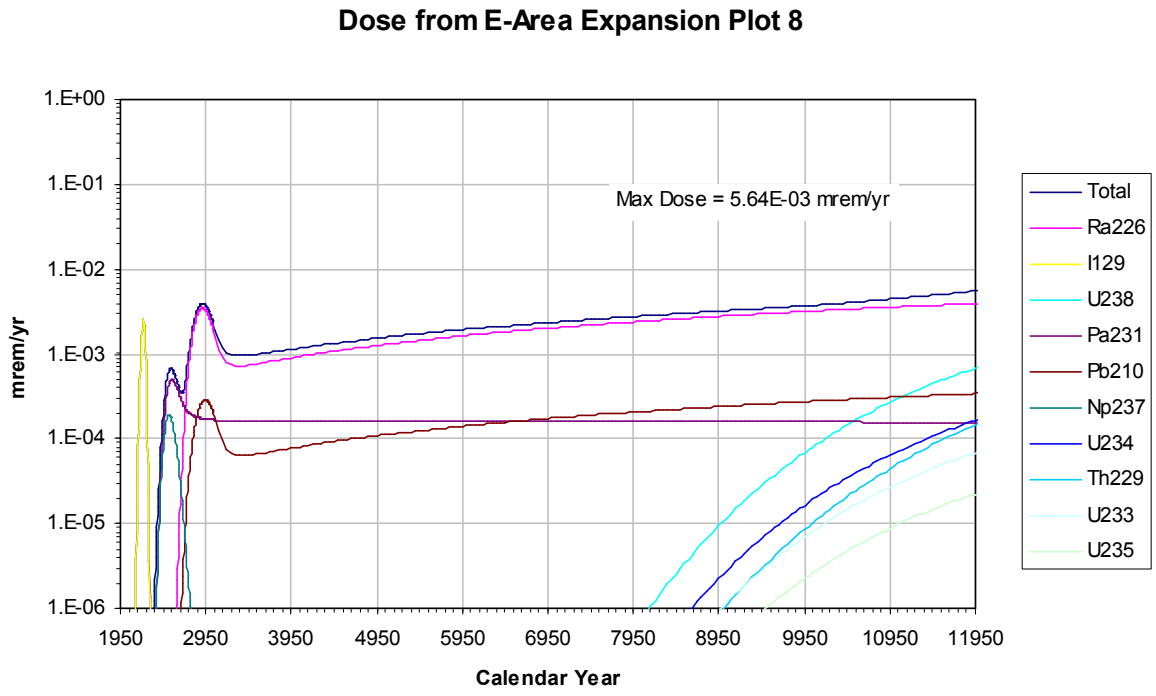


Figure 9-128. Radionuclide Contributions to Dose from ECP Plot 8

9.4.2.4.5 Inventory Sensitivity Summary

The inventory sensitivity studies conducted to evaluate the inventory multiplier that would be required to reach a maximum total dose of 30 mrem/yr resulted in 135 out of 152 sources with inventory multipliers greater than 1,000; 14 with inventory multipliers between 100 and 1,000; and only three sources with an inventory multiplier less than 100. The results of the sensitivity studies are summarized in Table 9-47. The H-Canyon, FMB IOU and LTR IOU had inventory multipliers of approximately 29, 15, and 10, respectively. The overall inventory multiplier required to produce a maximum total dose of 30 mrem/yr, for the combined sources associated with each POA were determined to be approximately 28, 14, 75, 10, and 950, respectively for the UTR, FMB, SC/PB, LTR, and Savannah River POA combined sources.

Table 9-47. Summary of Inventory Sensitivity Cases

POA Inventory Multiplier Sensitivity Case	
POA	**Multiplier on Total Inventory Required to Reach 30 mrem/yr**
Upper Three Runs	28
Fourmile Branch	14
Steel Creek/Pen Branch	75
Lower Three Runs	10
Savannah River	950

Other Significant Inventory Sensitivity Cases		
Source	**Sensitivity Case**	**Max Total Dose (mrem/yr)**
H-Canyon/HB-Line	1% of SAR Inventory (Base Case)	1.04
	10% of SAR Inventory	10.4
ECP Plots	No Plots (Base Case)	0
	Use of Plots 6, 7, 7a and 8	0.012

Inventory sensitivity studies were also conducted for selected facilities, for which some uncertainty currently exists relative to their end states and inventories. The studies resulted in the following conclusions:

- Increasing the H-Canyon/HB-Line inventory by a factor of 10 over the base case inventory increased its maximum total dose to approximately 10 mrem/yr, which is one third of the 30 mrem/yr constraint.
- The FAMS facility did not produce a significant CA dose even with 100% of the SAR inventory remaining at closure.
- Shipping the TRU Pad 1 TRU waste to an off-site disposal facility has very little impact upon the CA dose, because leaving it in place only results in a maximum total dose of 1.2E-12 mrem/yr during the 1,000-year CA assessment period.

- Transferring the TRU Pad 1 TRU waste to HB-Line has very little impact upon the H-Canyon/HB-Line maximum total dose during the 1,000-year CA assessment period.
- Use of ECP Plots 6, 7, 7a and 8 has very little impact upon the CA dose.

9.4.2.5 Groundwater Divide

Most of the high-impact radionuclide sources that are expected to remain when the SRS end state is achieved are located in the GSA, close to the center of the site. The GSA is situated on a relatively narrow strip of upland located between two site streams, UTR and FMB. A local groundwater divide runs parallel to these streams and is approximately half way between the streams. In the CA Base Case model, the best estimate position of the divide was used to determine which stream a surface disposal facility would eventually release its radionuclide inventory into or how much of its inventory would be discharged in one direction versus the other. Area Completion Projects has developed numerous water table maps of the GSA over the years as part of reporting requirements with the state regulator, the South Carolina Department of Health and Environmental Control (SCDHEC). These maps are of the configuration of the water table at a specific point in time (e.g., 3Q 1998 or 1Q 2002). The configurations of the water table and therefore the positions of the divide are never identical on these "snapshots" or representations that reflect one point in time. The system is dynamic and recharge occurs in discreet events that vary in intensity, duration and timing. The water table responds to recharge events, so the configuration will vary with respect to time. Because the estimated position of the groundwater divide has shifted slightly from year to year, a sensitivity scenario was devised to evaluate the worst outcome of such a change in conditions.

The long-term average flow rates associated with UTR and FMB are considerably different, 237 cfs versus 32 cfs, respectively. Dilution of contaminant fluxes at the POA on each of these streams is directly proportional to these average flow rates, hence the greater impact would be realized if more contaminant discharged to FMB, where far less dilution occurs. The strategy for this sensitivity scenario was to evaluate the bounding condition where the maximum plausible amount of disposed inventory might be diverted toward FMB. It is anticipated that this could only impact that part of the LLRDWF and MWMF, which discharges to UTR, and the FTF and HTF, which both straddle the groundwater divide. This analysis evaluates each of these situations separately by placing the entire inventory expected to discharge northward toward UTR into a groundwater pathway carrying it toward FMB.

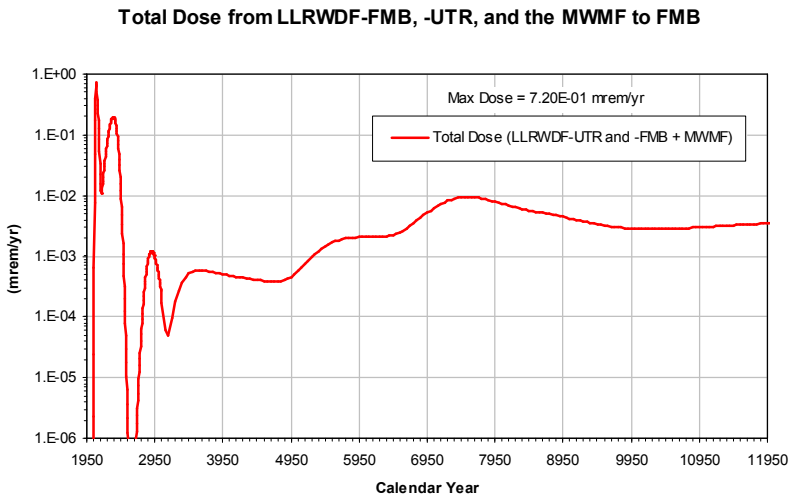
9.4.2.5.1 LLRWDF and MWMF

Base case modeling of the LLRWDF and MWMF was described in Section 9.3.7.1. The inventory from the LLRWDF UTR facility and the MWMF facility (all of which discharge to UTR) were introduced into that part of the LLRWDF that discharges toward FMB to evaluate this sensitivity scenario for E-Area. Because the radionuclide inventories from both the MWMF and the LLRWDF UTR were combined into a single simulation, the later inventory date of 1990 associated with LLRWDF was used for the entire inventory as opposed to separate simulations or using the earlier inventory date of 1979 associated with the MWMF. Using a single date for the combined inventories is expected to maximize the likelihood of peak impacts occurring simultaneously, thus it is expected to bias the results toward a higher impact.

The results of the GoldSim™ simulation of the release of the combined inventories are presented in Figure 9-129. The upper chart indicates the total dose that results from combining the radionuclide inventories from LLRWDF UTR and MWMF, along with the original inventory of LLRWDF FMB, and allowing them to migrate along the groundwater pathway defined for LLRWDF FMB. The maximum total dose realized from this simulation was 0.72 mrem/yr. When the maximum total dose from the original LLRWDF FMB simulation is subtracted from this, the net maximum total dose from diverting the LLRWDF UTR and MWMF inventories to FMB is 0.70 mrem/yr.

The lower chart indicates the total dose that results from combining the radionuclide inventories from LLRWDF UTR and MWMF and allowing them to migrate along the groundwater pathway defined for LLRWDF UTR, also using a single inventory date. The maximum total dose realized from that simulation was 0.11 mrem/yr.

Therefore, using the ratio of the maximum total doses, the increase in impact that results from diverting the radionuclide inventory associated with LLRWDF UTR and MWMF from UTR to FMB is a factor of 6.5 times higher. This result is largely proportional to the long-term average flow rate differences between UTR and FMB but is also influenced by the length and clay content differences of the groundwater flow paths.



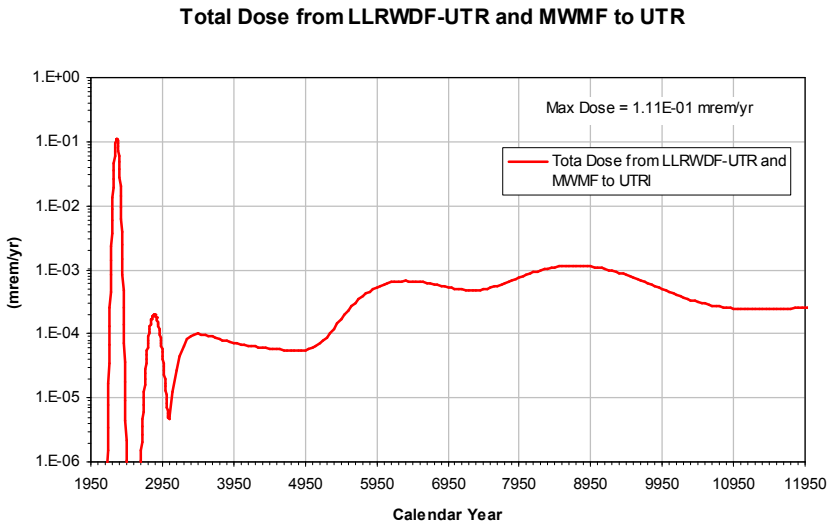


Figure 9-129. Total Dose Comparison for UTR and FMB Discharge Pathways for the Combined LLRWDF UTR and MWMF Inventories

9.4.2.5.2 F-Tank Farm

In the CA Base Case, there are six different groupings of F-Area tanks and equipment (items 44-49 in Table 9-24), each of which was simulated separately (Smith et al. 2009a). Five of these groupings discharge toward UTR and one discharges into FMB. A "bounding case" simulation of the collective inventory of these five tank groupings was made to establish the maximum dose that would occur if the groundwater divide were to shift so as to divert their discharge to FMB. The combined inventory from these units was introduced into the groundwater flow path associated with the F-Tank grouping that discharges toward FMB. To evaluate the impact of diverting the contaminant releases from the five F-Area Tank Farm groupings to FMB, the releases from these same units to UTR were combined such that the total dose realized in UTR could be compared to the total dose realized in FMB. This comparison is shown in Figure 9-130.

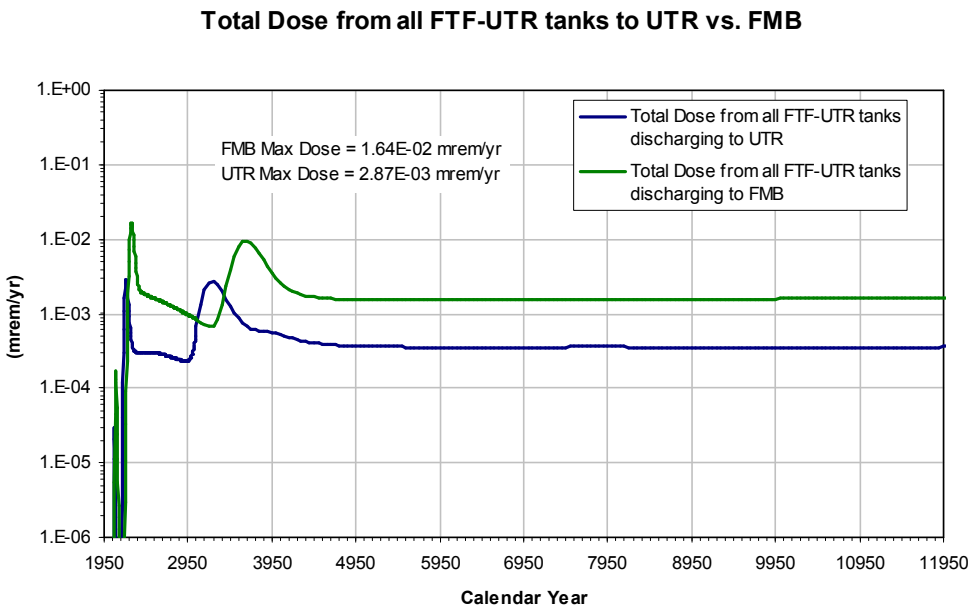


Figure 9-130. Total Dose from FTF-UTR Tanks Realized in UTR and in FMB

To enable the sensitivity analysis, simulations were made of each of the FTF UTR tank groupings individually as well as in a combined fashion. The tank groupings and the maximum total dose realized, both individually and collectively, are presented in Table 9-48.

Table 9-48. Summary of Maximum Doses of F-Tank Farm Groups of Tanks

Tank Grouping	Discharge Stream	Maximum Total Dose (mrem/yr)
FTF Type I	UTR	2.57E-04
FTF Type IV	UTR	6.31E-05
FTF Type III	UTR	1.43E-03
FTF Type IIIA-UTR	UTR	5.96E-06
FTF Type IIIA-FMB	FMB	1.04E-04
FTF Equipment	UTR	1.08E-05
Combined UTR Tanks to UTR	UTR	2.87E-03
Combined UTR Tanks to FMB	FMB	1.64E-02

The combined total doses calculated for each tank grouping that discharges to UTR produced a maximum total dose of 2.87E-03 mrem/yr. The maximum total dose realized when the inventory from these same tank groupings was diverted to FMB was 1.64E-02 mrem/yr. Hence the sensitivity, as estimated by calculating the ratio of the maximum total doses, is (1.64E-02/2.87E-03) = 5.73. This factor of 5.7 represents the maximum increase in the dose contributed by the UTR Tank groupings if the groundwater divide were to shift and cause them to discharge into FMB. This increase in dose is predominantly controlled by the decrease in dilution resulting in the lower average stream flow rate in FMB, as compared to UTR. Examination of Figure 9-130 also shows that discharging to FMB instead of UTR introduces a time delay in the dose peaks.

9.4.2.5.3 H-Tank Farm

There are 13 different groupings of H-Area tanks and equipment (items 53-65 in Table 9-24) each of which was simulated separately in the CA Base Case evaluation (Smith et al. 2009a). Four of these groupings discharge toward FMB and nine toward UTR. Each grouping was broken out based on tank source type and groundwater flow path length. To evaluate the "bounding case" which might result from a shifting in the position of the groundwater divide between UTR and FMB, a simulation whereby the collective inventory of the nine HTF-UTR tank groupings was diverted to FMB via the longest groundwater pathway of the four HTF-FMB tank groupings was conducted. Simulations of individual releases from the nine HTF-UTR groupings to UTR were also conducted so as to enable the comparison of their collective impact in UTR to their collective impact in FMB. This comparison is illustrated by the graphs presented in Figure 9-131 indicating the total dose realized in UTR from these sources and the total dose realized in FMB from those same sources. A summary of the individual HTF-UTR source groupings evaluated in the sensitivity analysis as well as the combined impact to both UTR and FMB, in terms of maximum total doses, is presented in Table 9-49.

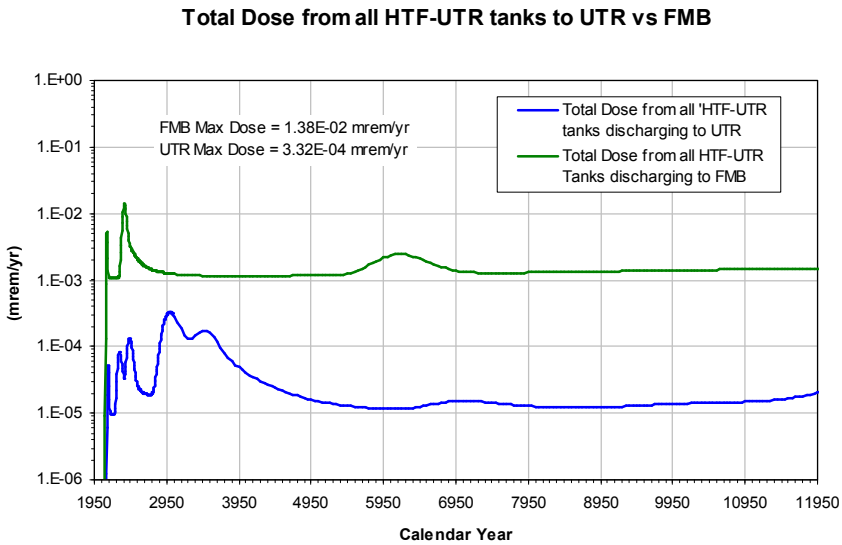


Figure 9-131. Total Dose from HTF-UTR Tanks Realized in UTR and in FMB

Table 9-49. Summary of Maximum Doses of HTF Groups of Tanks

Tank Grouping	Discharge Stream	Maximum Total Dose (mrem/yr)
HTF Type I UTR E	UTR	1.07E-04
HTF Type I UTR W	UTR	3.57E-05
HTF Type II	UTR	1.34E-04
HTF Type IV UTR	UTR	1.23E-05
HTF Type III UTR E	UTR	5.40E-05
HTF Type III UTR W	UTR	1.84E-06
HTF Type IIIA UTR E	UTR	2.67E-04
HTF Type IIIA UTR W	UTR	1.73E-05
HTF Equipment UTR	UTR	2.15E-05
Combined HTF Tanks-UTR to UTR	UTR	3.32E-04
Combined HTF Tanks-UTR to FMB	FMB	1.40E-02

The combined total doses calculated for each tank grouping that discharges to UTR produced a maximum total dose of 3.32E-04 mrem/yr. The maximum total dose realized when the combined inventory from these same tank groupings was diverted to FMB was 1.40E-02 mrem/yr. Hence the sensitivity, as estimated by calculating the ratio of the maximum total doses, is (1.40E-02/3.32E-04) = 42.2. This factor of 42.2 represents the increase in the maximum total dose contributed by the UTR tank groupings if the groundwater divide were to shift and cause them to discharge into FMB. This factor is considerably higher than can be explained just by the difference in stream flow rates of UTR and FMB (a factor of about seven times) and is also strongly influenced by the markedly shortened groundwater flow path length that would result should the groundwater divide shift toward UTR sufficiently to capture the contaminants from the FTF-UTR tank sources. The longest FMB groundwater path for an HTF source is 5,812 ft compared to the groundwater path lengths of the HTF-UTR Tank sources, which ranged from 5,494 to 17,495 ft and averaged 11,227 ft. This dose increase for the HTF-UTR tank sources represents a relatively large percentage increase to an original dose contribution that was very small. Even after evaluating the bounding case, the dose contribution is still less than the criterion established in the base case evaluation for determining if a source release model needed to be implemented beyond the generic evaluation.

9.4.2.5.4 Groundwater Divide Sensitivity Summary

The groundwater divide sensitivity analysis consisted of taking the LLRDWF/MWMF, FTF, and HTF, all of which straddle the GSA groundwater divide, and evaluating the dose increase that results from assuming that the groundwater divide changes from base case conditions, such that the entire facilities discharge to FMB. Table 9-50 provides a summary of the results of this evaluation. The dose from those portions of LLRWDF/MWMF and F-Tank Farm, which under base case conditions discharge to UTR, increased by a factor of approximately six, when discharge to FMB was assumed instead.

This increase was predominately due to the lower annual average flow rate in FMB relative to UTR. The dose from that portion of H-Tank Farm, which under base case conditions discharged to UTR, increased by a factor of approximately 42, when discharge to FMB was assumed instead. This increase was predominately due to both the lower annual average flow rate in FMB relative to UTR and the shorter aquifer flow path length assumed to FMB.

Overall, while the doses increased due to this assumed change in the groundwater divide, all the doses remained relatively low. It is also unlikely that such a drastic change in the groundwater divide location would occur, particularly in the direction toward UTR, since UTR has incised deeper and has a greater flow rate than FMB.

Table 9-50. Groundwater Divide Sensitivity Summary

Facility	Maximum Total Dose to UTR (mrem/yr)	Maximum Total Dose to FMB (mrem/yr)	Dose Increase Factor	Predominate Reason(s) for Increase
LLRWDF / MWMF	1.11E-01	7.02E-01	6.5	~7 times less flow in FMB than UTR
F-Tank Farm	2.87E-03	1.64E-02	5.7	~7 times less flow in FMB than UTR
H-Tank Farm	3.32E-04	1.40E-02	42.2	~7 times less flow in FMB than UTR and flow path almost half the original length

9.4.2.6 Alternative End-State Date

The base case end-state date of 2025 was determined by Phifer and Wilhite (2008). However, as also noted by Phifer and Wilhite, longer operating dates are projected for some SRS facilities. The current SRS liquid waste disposition plan (WSRC 2007f) projects that all FTF and HTF tanks will have been grouted, Defense Waste Processing Facility (DWPF) and SWPF will have been cleaned, and the Glass Waste Storage Buildings and the canister shipping facility will have been decommissioned and closed by September 2032. The SRS End State Vision (DOE 2005) also states, "The SRS Cleanup Project mission and goal is to complete the cleanup by 2025 and transition SRS to a site focused on national security. SRS will accommodate the ongoing National Nuclear Security Administration (NNSA) missions before and beyond 2025. SRS is not a DOE closure site." Since site operations are expected to continue past 2025, the sensitivity of CA dose estimates to varying the SRS end-state year from 2025 to 2050 was evaluated. No attempt was made to revise estimated closure inventories with the extended period of operation. Results from this sensitivity study are summarized in Table 9-51.

Table 9-51. Maximum Cumulative Doses at Each POA

Point of Assessment	Maximum Dose 2025 to 3025 mrem/yr	Maximum Dose 2050 to 3050 mrem/yr	Decrease mrem/yr
Upper Three Runs	1.06	1.06	0
Fourmile Branch	2.16	1.09	1.07
Steel Creek/Pen Branch	0.42	0.23	0.19
Lower Three Runs	2.97	1.58	1.39
Savannah River	0.14	0.14	0

Except for the UTR POA, the maximum doses all originated from Cs137 released from stream beds. These maxima occurred at the beginning of the CA period of assessment and declined thereafter. Therefore, shifting the start of the CA assessment period 25 years later results in lower doses at all of the POAs except for UTR and the Savannah River at the U.S. 301 Bridge. The dose at the U.S. 301 Bridge is calculated from an accumulation of the isotopic concentrations in the river from all upstream discharges, is very low to begin with, and does not change significantly.

Figure 9-132 through Figure 9-136 plot doses from direct discharges at each POA (i.e., they do not include accumulation of dose from upstream sources) and show the ten major sources contributing to the doses. Sources are listed in order from the source giving the highest dose to the one giving the lowest dose in the figure legends. The maximum dose at UTR is caused by Np237 released from H-Canyon/HB-Line which peaks in the year 2820. Even without the dose from the H-Canyon/HB-Line source, the contribution from the UTR IOU to total dose at this POA is relatively less than at the other POAs. The contributions from Cs137 released from streambed IOUs are linear on the semi-log plots, peak as soon as the inventory is established, and are declining when the CA period of assessment begins. The IOU sources typically cause the highest dose to occur at the start of the CA assessment period. Therefore, shifting the start of the assessment period from 2025 to 2050 decreases the maximum dose. The linear decrease in IOU doses on the semi-log plots results from the decay of Cs137 with a half-life of 30 years.

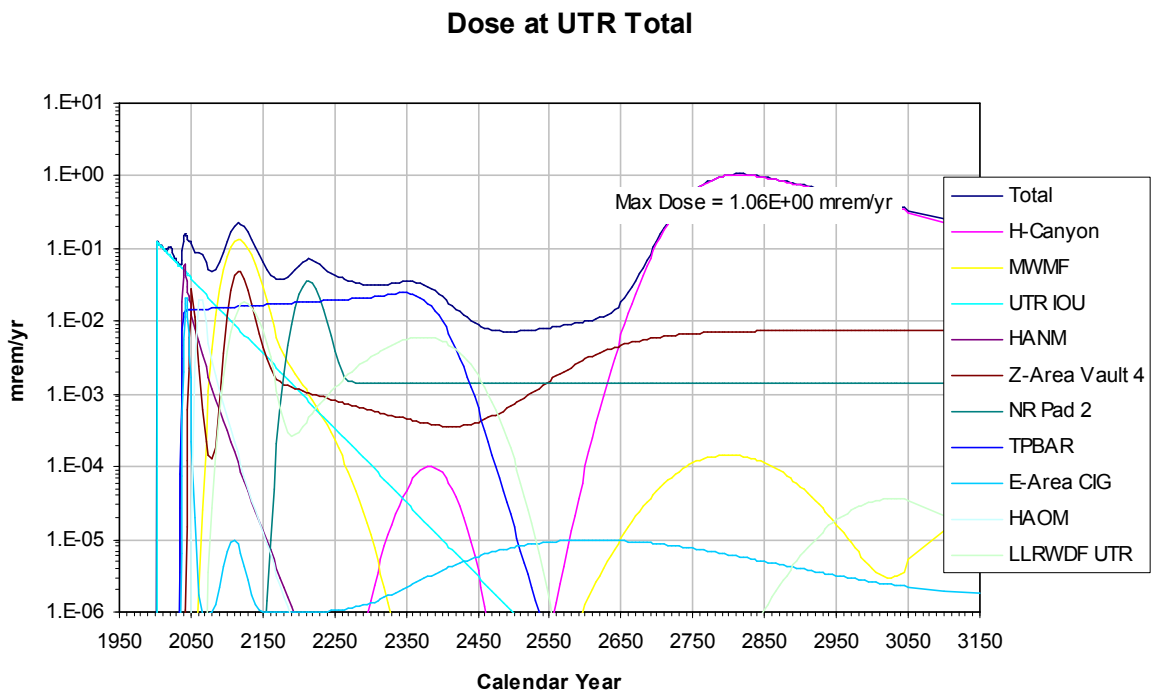


Figure 9-132. Total Direct Dose to Upper Three Runs POA and Contribution from Major Sources

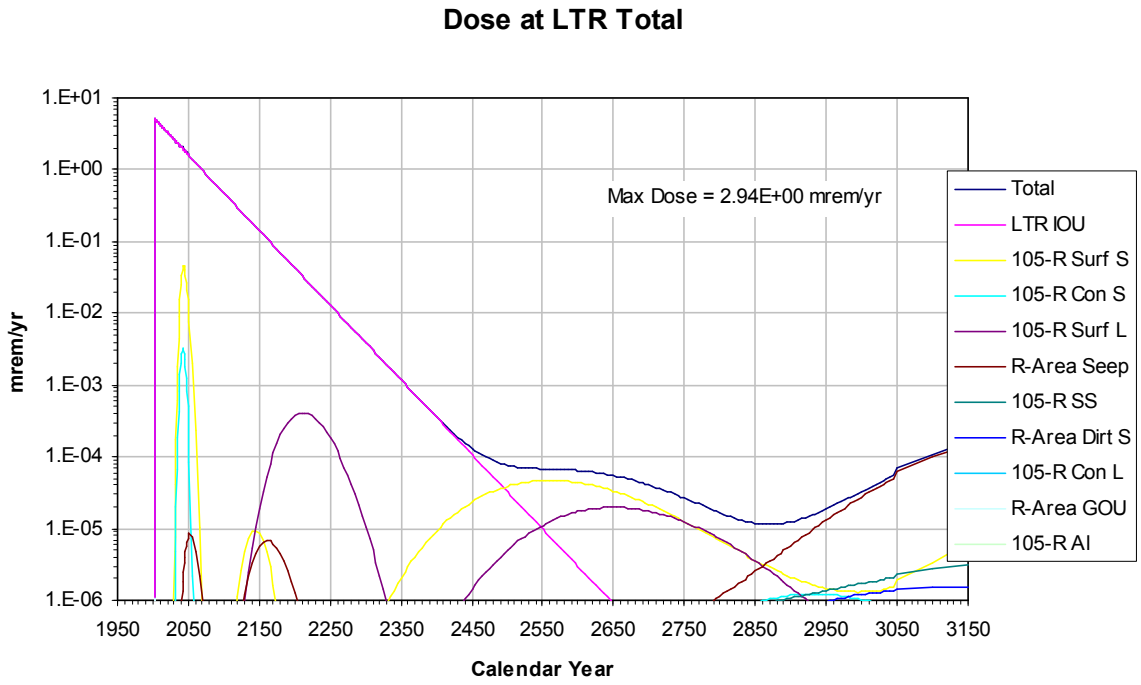


Figure 9-133. Total Direct Dose to Lower Three Runs POA and Contribution from Major Sources

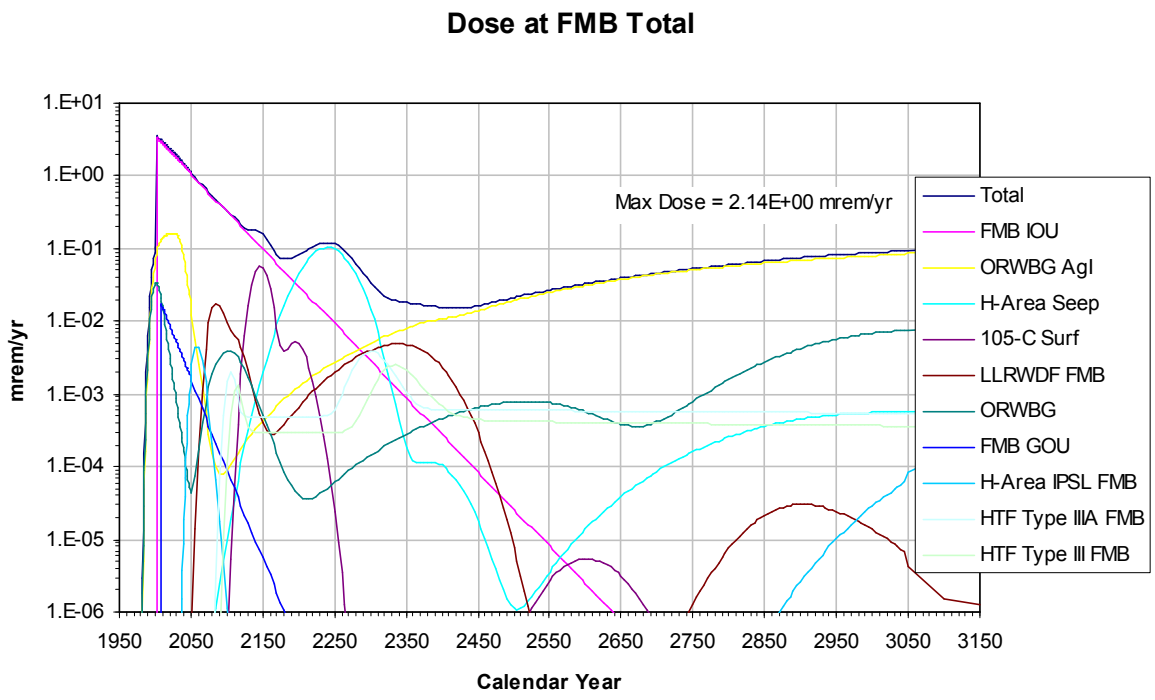


Figure 9-134. Total Direct Dose to Fourmile Branch POA and Contribution from Major Sources

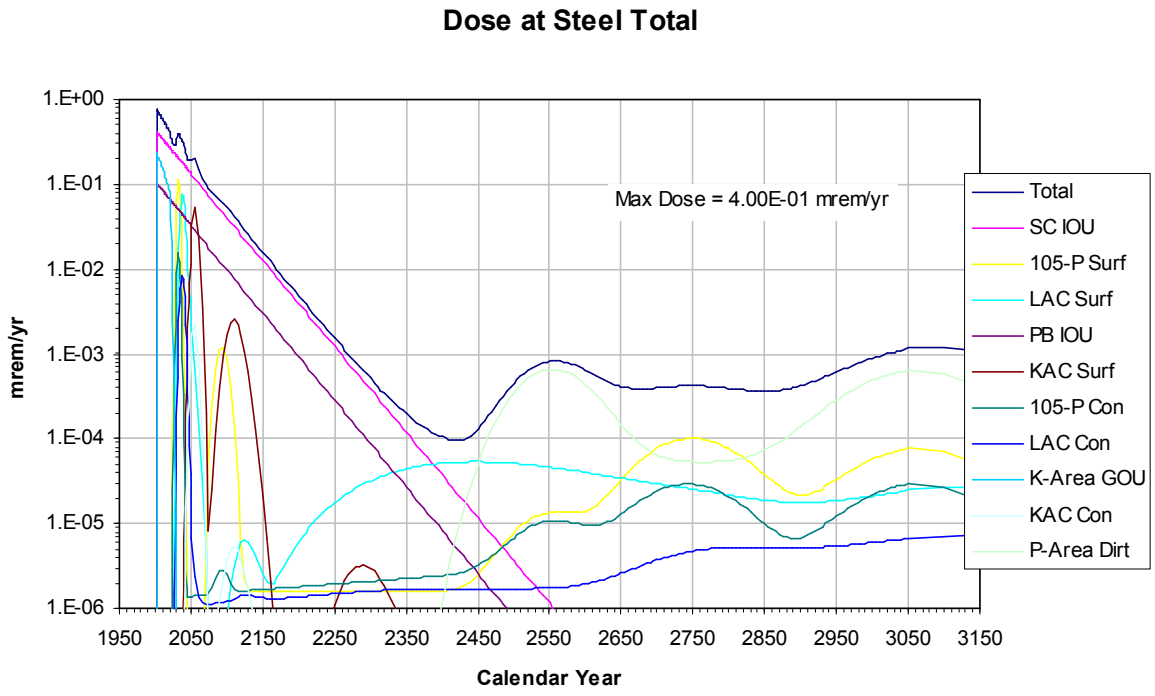


Figure 9-135. Total Direct Dose to Steel Creek/Pen Branch POA and Contribution from Major Sources

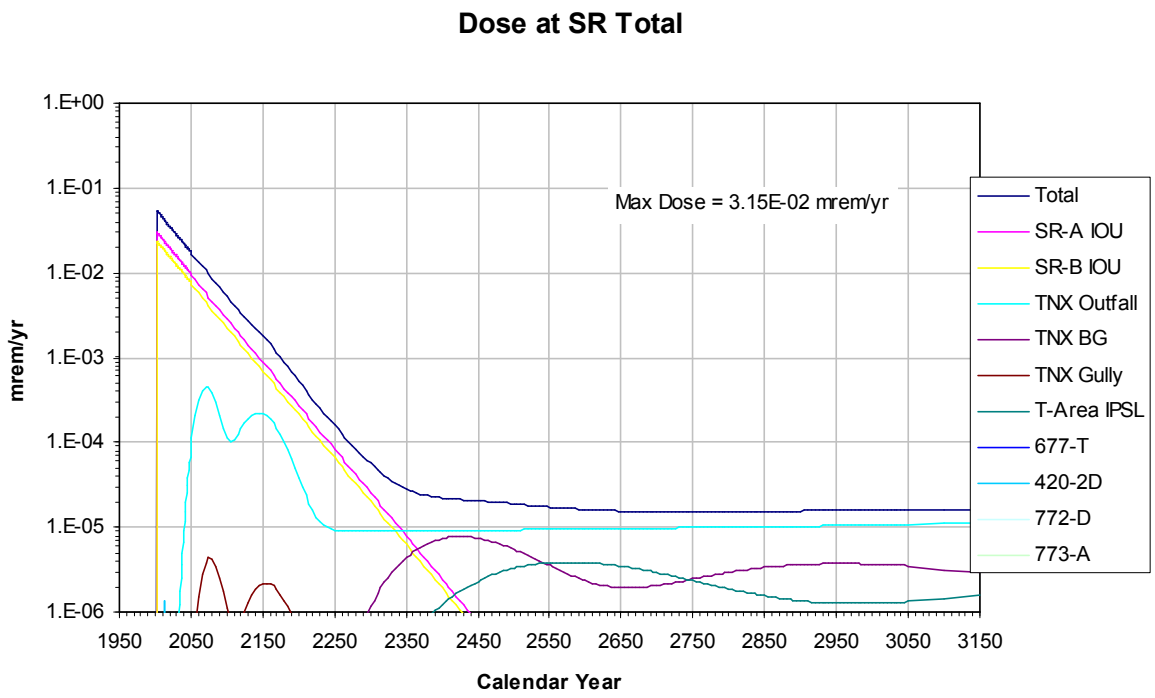


Figure 9-136. Total Direct Dose to Savannah River and Contribution from Major Sources

9.4.2.7 Bioaccumulation Factor

A BCF for C14 in fish of 3.0 L/kg was used for the deterministic base case CA calculations based on the study by Hinton et al. (2009). The reference study developed a detailed ecosystem model of carbon dynamics in FMB using data collected at SRS that was used to calculate the C14 BCF. This Water-to-Fish BCF takes into account the abundance of stable carbon within FMB and the resulting reduction in the Water-to-Fish BCF. Previously a C14 BCF of 4,500 L/kg (NCRP 1985) was applied in dose calculations and a value of 50,000 L/kg has recently been proposed by Lee and Coffield (2008) based on Staven et al. (2003). The study by Hinton et al. (2009) arrives at the much smaller value by accounting for, among other effects, the dilution of C14 by naturally occurring C12 in the environment. The deterministic calculations showed that ingestion of fish was the pathway giving the highest CA dose to a member of the public. The significant difference between various estimates of the bioaccumulation factor for C14 in fish clearly indicates that the sensitivity of CA doses to the C14 bioaccumulation factor must be established.

C14 was present in 79 of the 152 sources analyzed in the CA deterministic calculations. In these 79 sources, the amount of C14 ranged from 4,000 Ci in the E-Area ORWBG to 1.3E-08 Ci in the S-Area sand filter. The 39 sources having greater than 0.1 Ci of C14 were chosen for sensitivity analysis and model calculations were performed with the C14 BCF increased to 4,500 and 50,000 L/kg for comparison to results using the nominal value of 3 L/kg.

Results of the sensitivity analysis showed that maximum doses during the 1,000-year CA assessment period from 2025 to 3025 did not change when the higher C14 BCFs were used. Four of the eight sources having C14 inventories greater than 100 Ci were the stainless steel components in the reactor vessels. Isotopic release from these components was modeled as originating from slow corrosion of the metal (Smith et al. 2009a) which resulted in small contributions to overall doses. The contributions to total dose from C14 for the other four sources having C14 inventories greater than 100 Ci over a 10,000-year period are shown in Figure 9-137 through Figure 9-140.

As noted above, the largest C14 inventory (4,000 Ci) was in the ORWBG. As shown in Figure 9-137, assuming a BCF of 50,000 L/kg, the maximum dose from C14 at the ORWBG is about 45 mrem/yr. This dose would occur at the FMB POA in the year 4500. As shown in Figure 9-138, the next highest dose from C14, using the largest estimate of the BCF, is about 3 mrem/yr from the 2,830 Ci of C14 in the MWMF. This dose would occur at UTR POA in the year 6400. The total C14 inventory in all 152 sources was 9,340 Ci so the ORWBG and MWMF account for 73% of the total C14 inventory at SRS. Figure 9-137 through Figure 9-140 show that the contributions from C14 to total doses calculated during the CA assessment period are negligible.

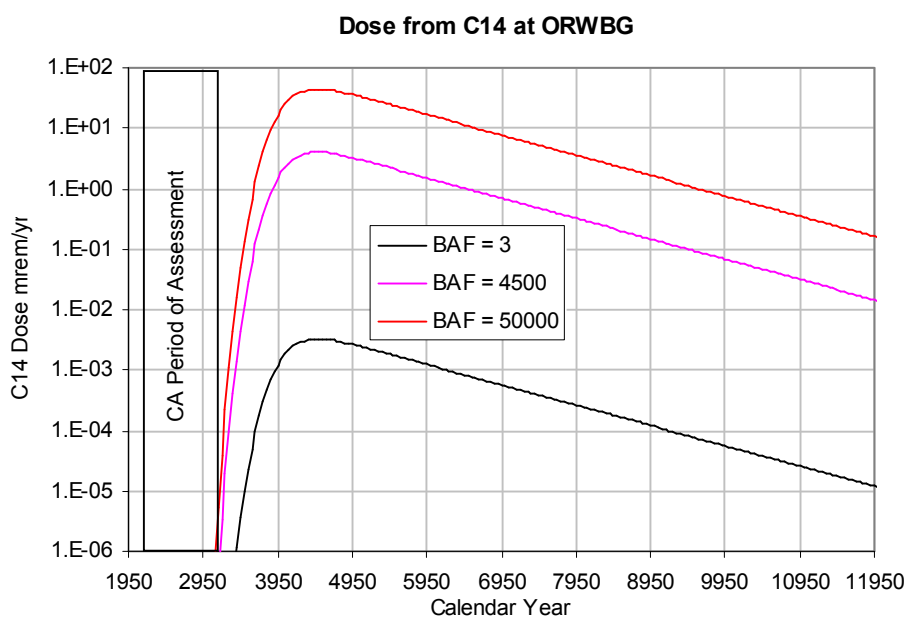


Figure 9-137. Dose from C14 for Old Radioactive Waste Burial Ground

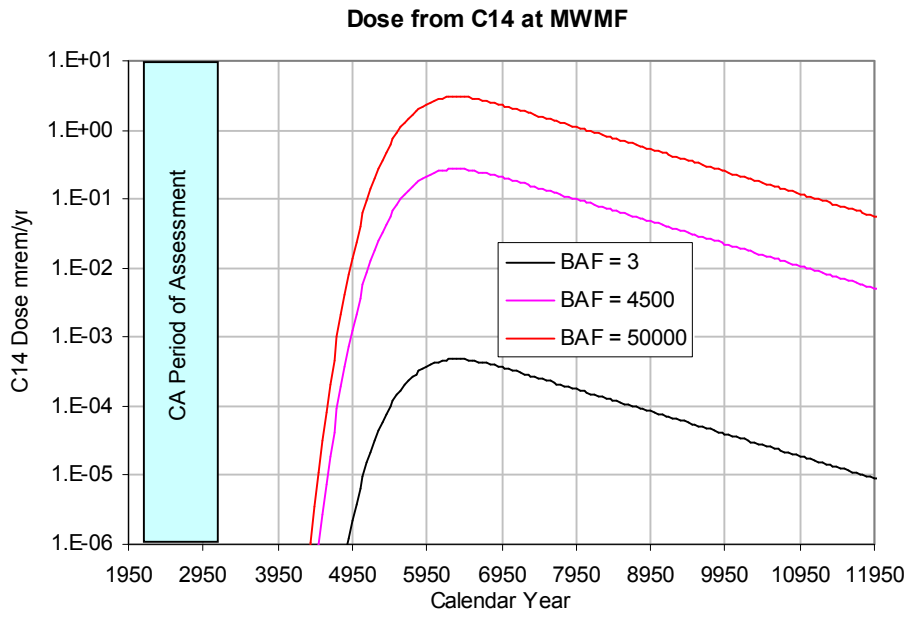


Figure 9-138. Dose from C14 for Mixed Waste Management Facility

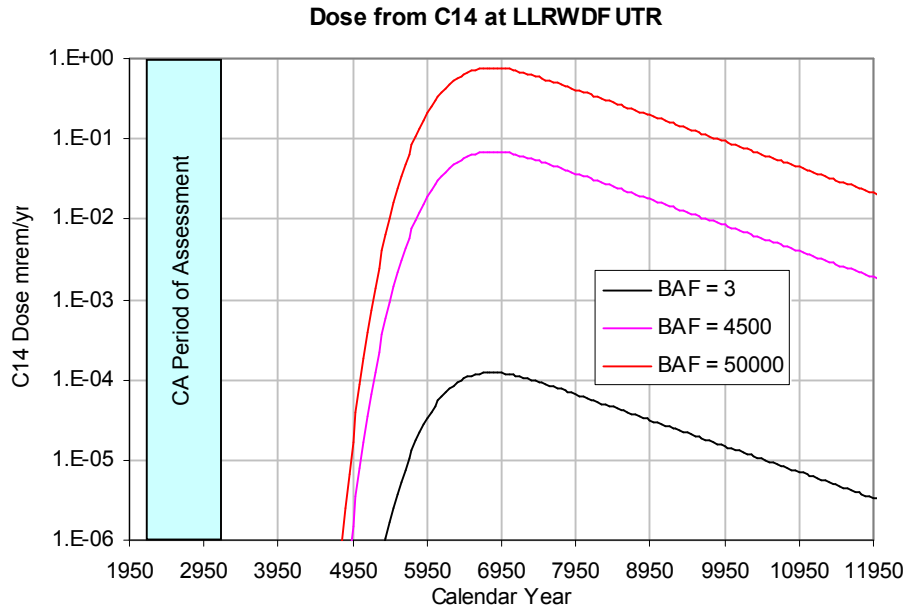


Figure 9-139. Dose from C14 for Low Level Radioactive Waste Disposal Facility at Upper Three Runs

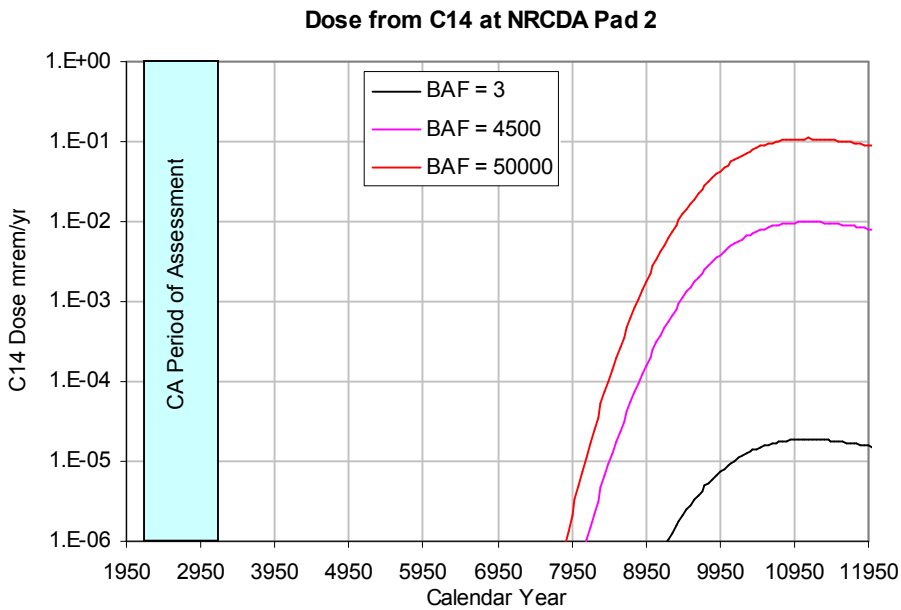


Figure 9-140. Dose from C14 for Naval Reactor Pad 2

9.4.2.8 Aquifer Clay

Clay regions in the aquifer will tend to increase radionuclide transport times because K_d values are typically greater in clayey soil which reduces the rate of migration in groundwater simulations. Delaying the transport of radionuclides with half-lives on the order of the transport time will decrease peak doses at the POA through decay. Longer transport times will also most likely lead to greater dispersion of the radionuclides, again decreasing peak doses. Because accounting for clay in the aquifer would tend to give lower peak doses, sensitivity runs were made for selected sources with and without clay present.

The 28 sources having maximum doses greater than 0.001 mrem/yr during the CA period of assessment that had any clay in the aquifer path, excluding three sources that were H3 only which has zero K_d in both sandy and clayey soil, were run replacing the clay with saturated sandy soil. Results for the maximum doses obtained with this sensitivity analysis are presented in Table 9-52 where the sources are arranged in order of decreasing clay length. The columns in Table 9-52 give the abbreviated name for the source, the point of assessment, the length of the clay region in the aquifer (Hamm et al. 2009), the water travel time through the clay, the dose obtained with clay present, the dose obtained with the clay replaced with sandy soil, and the ratio of the dose without clay to that with clay. Water travel time represents the unretarded time of passage through the clay.

On average, the maximum dose increased 37% (±48% at 1 σ) when clay in the aquifer was replaced with sand. The greatest increase was for the H-Canyon/HB-Line source where the maximum dose increased from 1.04 to 2.79 mrem/yr. Figure 9-141 and Figure 9-142 show the dose profiles over time for the H-Canyon/HB-Line source with and without clay, respectively. Without clay, the Np237 peak is much sharper leading to a higher maximum dose. The K_d for Np237 increases from 0.6 ml/g in sandy soil to 35 ml/g in clayey soil. With the exception of I129, which has a K_d of zero in sandy soil, Np237 has the largest relative difference in soil K_d (K_{dclay}/K_{dsand} = 58) of any radionuclide.

The next largest increase in maximum dose from replacing aquifer clay with sand occurred for the ORWBG where the maximum dose increased from 7.28E-03 to 1.81E-02 mrem/yr. Figure 9-143 and Figure 9-144 show the dose profiles over time for the ORWBG source with and without clay, respectively. Interestingly, the maximum dose during the CA assessment period is again caused by Np237 and the same peak sharpening in the absence of clay was found as observed for the H-Canyon/HB-Line Np237 source. In fact, careful observation showed that this same sharpening of the Np237 profile in the absence of clay also accounts for the increase in doses from the FTF Type III Tanks and 221-F. Therefore, the behavior of Np237 is responsible for the three increases in maximum dose greater than a factor of two and for the four largest increases in dose found in the clay sensitivity analysis. Table 9-53 shows the impact on the maximum dose at each POA from using dose profiles calculated for the 28 sources shown in Table 9-52 without clay in the aquifer. The maximum dose at the UTR POA is impacted the most by neglecting clay in the aquifer which reflects the change in the dose from H-Canyon/HB-Line. The maximum dose in the Savannah River is also impacted but remains relatively low. Neglecting clay has little impact on maximum doses at the other three POAs.

Table 9-52. Maximum CA Doses from Clay Sensitivity Study

Source	POA	Clay Length (Feet)	Water Travel Time (Years)	Dose with Clay (mrem/yr)	Dose without Clay (mrem/yr)	Dose Ratio
105-C Surf	FMB	1861.51	19.74	5.75E-02	5.70E-02	0.99
KAC Surf	SC	1615.87	19.11	5.30E-02	5.42E-02	1.02
KAC Con	SC	1615.87	19.11	2.17E-03	2.16E-03	1.00
105-R Surf S	LTR	386.14	3.16	4.68E-02	5.24E-02	1.12
105-R Con S	LTR	386.14	3.16	3.26E-03	3.06E-03	0.94
E-Area CIG	UTR	117.20	2.19	2.09E-02	2.47E-02	1.18
E-Area ST Cent	UTR	115.08	2.31	1.33E-02	2.11E-02	1.59
H-Area IPSL FMB	FMB	94.10	3.70	4.36E-03	6.82E-03	1.56
F-Area Seep	FMB	74.88	3.83	2.24E-03	2.28E-03	1.02
LAC Surf	SC	71.15	2.92	7.57E-02	8.95E-02	1.18
LAC Con	SC	71.15	2.92	8.24E-03	9.97E-03	1.21
H-Area IPSL UTR	UTR	52.75	2.09	3.69E-03	4.53E-03	1.23
HAOM	UTR	46.71	1.53	2.02E-02	2.51E-02	1.24
281-3F	UTR	37.14	2.76	8.34E-03	1.43E-02	1.72
HTF Type III FMB	FMB	31.84	2.44	2.49E-03	3.98E-03	1.60
H-Canyon	UTR	28.08	1.28	1.04E+00	2.90E+00	2.79
ORWBG AgI	FMB	27.19	1.84	1.63E-01	1.64E-01	1.01
ORWBG	FMB	27.19	1.84	7.28E-03	1.81E-02	2.48
MWMF	UTR	26.64	1.93	1.35E-01	1.45E-01	1.08
LLRWDF UTR	UTR	26.64	1.93	1.80E-02	1.95E-02	1.08
FTF Type III	UTR	17.21	2.20	1.43E-03	3.22E-03	2.26
Z-Area Vault 4	UTR	15.02	0.88	4.93E-02	6.96E-02	1.41
LLRWDF FMB	FMB	14.88	2.60	1.79E-02	2.37E-02	1.33
221-F	UTR	14.47	0.98	2.75E-03	5.13E-03	1.86
HTF Type IIIA FMB	FMB	12.08	1.23	3.89E-03	4.60E-03	1.18
Z-Area Vault 1	UTR	9.36	0.53	7.71E-03	1.02E-02	1.33
NR Pad 2	UTR	7.90	1.88	3.65E-02	3.67E-02	1.01
F-Area Old Seep	UTR	1.51	0.06	1.16E-02	1.08E-02	0.93
maximum		1861.51	19.74	1.040	2.903	2.79
minimum		1.51	0.06	0.001	0.002	0.93
average		243.06	3.93	0.065	0.135	1.37
standard deviation		523.35	5.50	0.195	0.544	0.48

Table 9-53. Impact of Clay on Dose at POA

Point of Assessment	Maximum Cumulative Dose with Clay mrem/yr	Maximum Cumulative Dose without Clay mrem/yr
Upper Three Runs	1.06	2.92
Fourmile Branch	2.16	2.16
Steel Creek/Pen Branch	0.42	0.46
Lower Three Runs	2.97	2.97
Savannah River	0.14	0.38

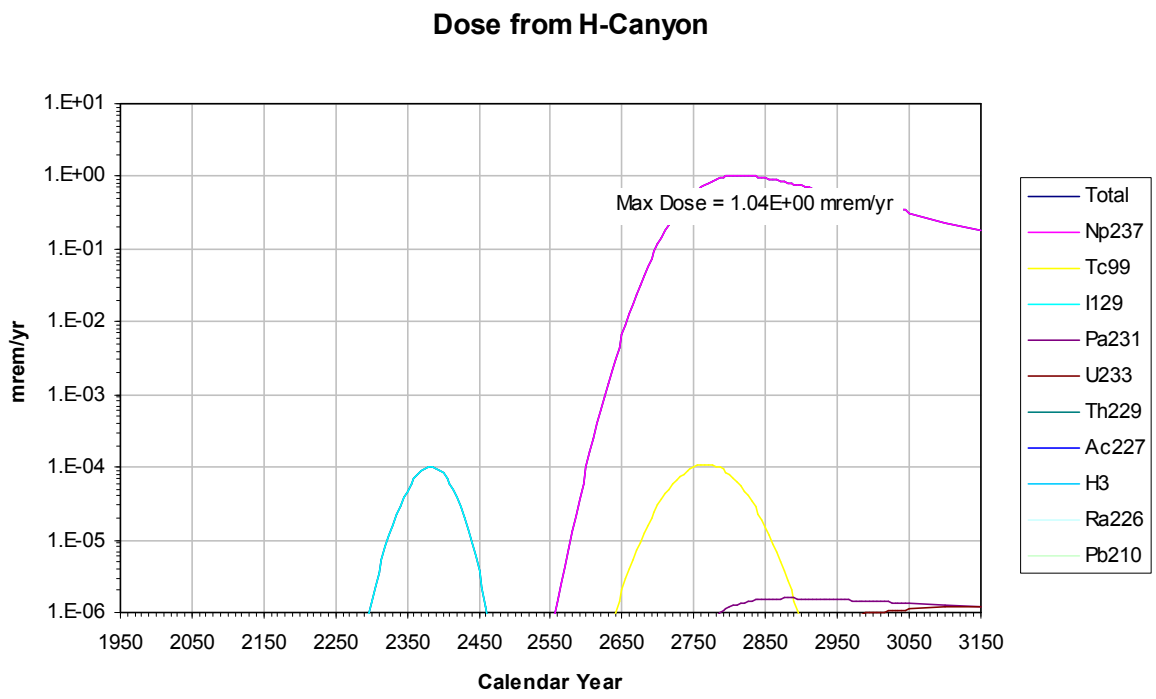


Figure 9-141. H-Canyon/HB-Line Dose Profile with Clay in Aquifer Path

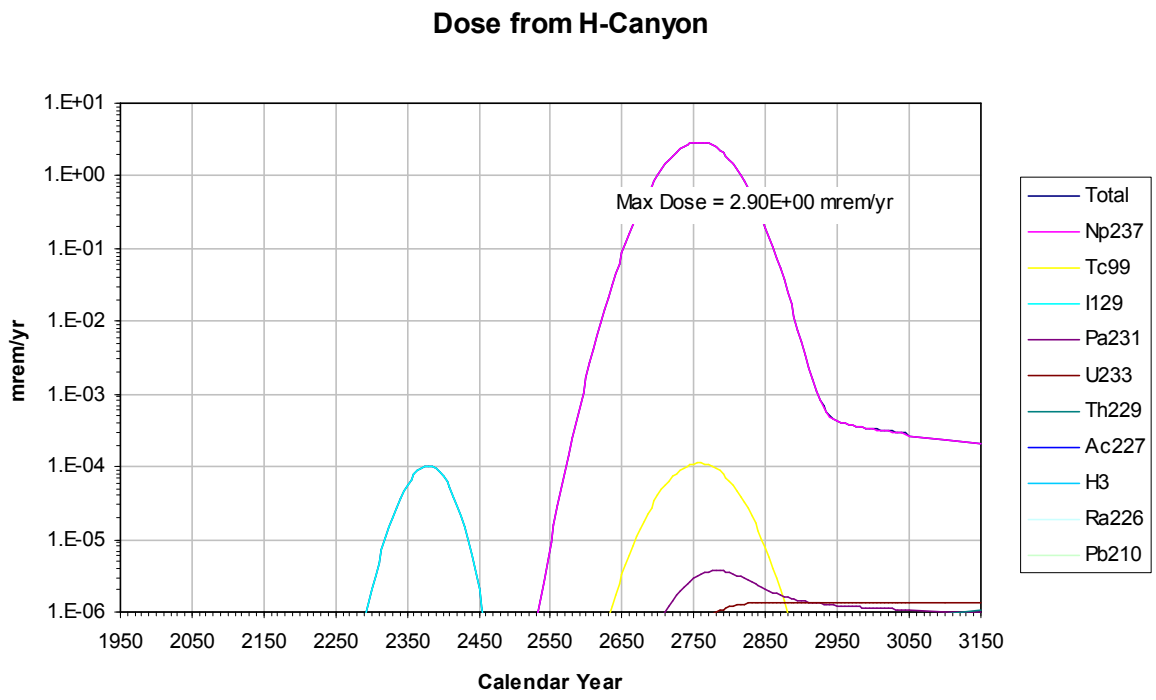


Figure 9-142. H-Canyon/HB-Line Dose with No Clay in Aquifer Path

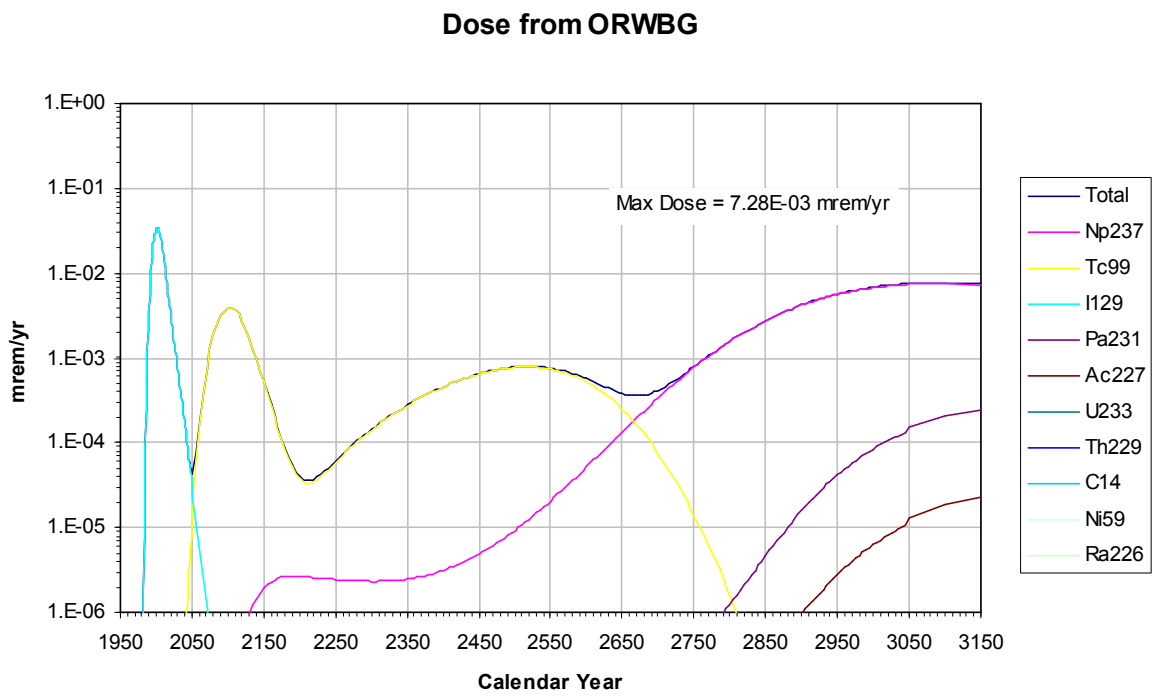


Figure 9-143. ORWBG Dose Profile with Clay in Aquifer Path

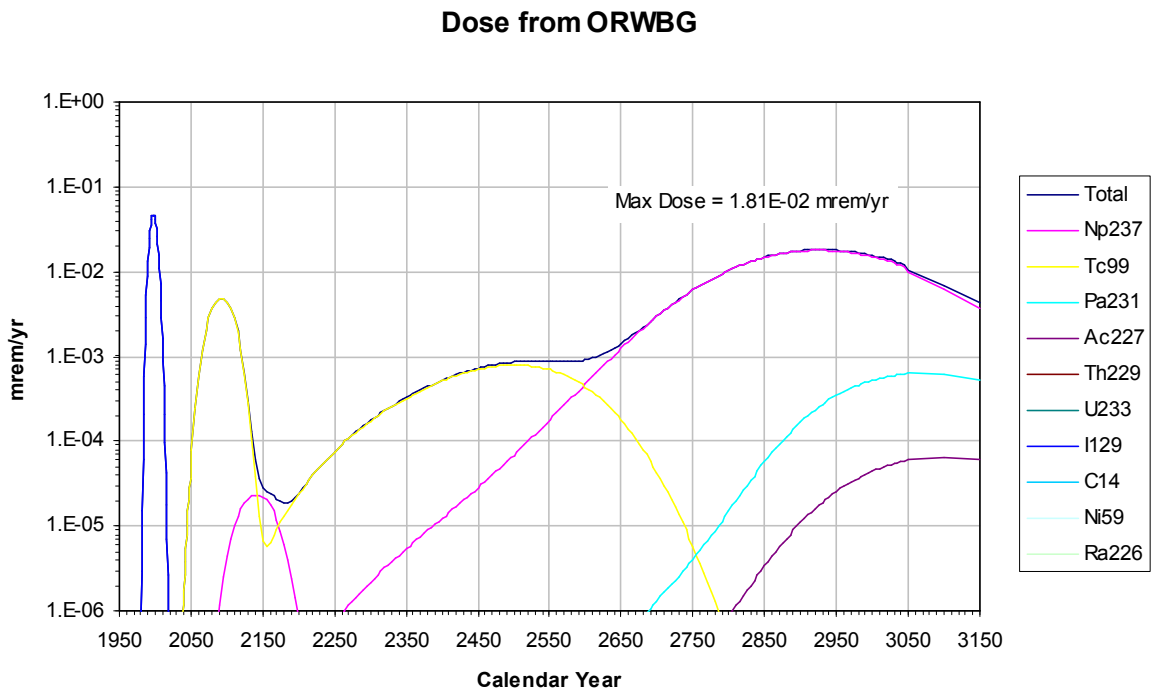


Figure 9-144. ORWBG Dose Profile with No Clay in Aquifer Path

9.4.2.9 Maximum Doses over 100,000 Years

Deterministic results from the base case CA calculations (Smith et al. 2009a) typically showed dose behavior from the start of SRS operations to just past the end of the 1,000-year CA assessment period at 3025. However, doses were actually calculated for the 10,100-year period from 1950 to 12050. The 41 CA sources having maximum doses greater than 0.001 mrem/yr during the 10,025-year time period from year 2025 to year 12,050 were further run to year 102,050 to determine if maximum doses had been seen. The list of 41 sources (out of all 152 sources) chosen for this sensitivity analysis excluded sources that were entirely H3 from the tritium processing buildings and GOUs or Cs137 (IOUs) and I129 from the ORWBG, all of which had clearly peaked during the CA assessment time. Results for the maximum doses obtained from this sensitivity analysis are presented in Table 9-54 where maximum doses over the three time periods: 2025 to 3025, 2025 to 12050, and 2025 to 102050 are listed. The fifth and sixth columns in Table 9-54 indicate the 12 sources where the maximum dose during the second (approximately 10,000 year) time period exceeded the maximum base case dose during the CA period of assessment and the 10 sources where the maximum dose during the third (approximately 100,000 year) time period exceeded the dose during the second. Sixteen of the sources (the unique sources identified in the 5th and 6th columns of Table 9-54) have maximum doses outside the CA period of assessment. Doses from the six sources indicated in the last column of Table 9-54 are increasing at the end of the simulation and, therefore, possibly have maximum doses beyond year 102,050.

Figures E-137 through E-158 in Appendix E show plots of the dose over approximately 100,000 years for the 16 sources where the maximum dose occurred outside the CA period of assessment and for the six other sources (indicated in the last column of Table 9-54) where the dose was increasing at the end of the 100,000-year period. Maximum doses printed on the figures are for the entire 100,000-year time period. The 10 radionuclides giving the highest doses are plotted and listed in the figure legends in order from the one giving the highest dose to the one giving the lowest dose. The ten sources that were found to have peak doses very near or beyond 100,000 years are:

- 105-C Surface Contamination (Figure E-138)
- 105-P Concrete (Figure E-139)
- 221-F (Figure E-141)
- LLRWDF (both aquifer paths, Figures E-150 and E-151)
- MWMF (Figure E-152)
- Zircaloy in the NRCDA (metal parts 5 and 6, Figures E-153 and E-154)
- NRCDA Pad 2 (Figure E-155)
- ORWBG (Figure E-156)

From Figures E-139, E-141 and E-156, it appears that the total doses from 105-P concrete, 221-F, and the ORWBG are peaking or have actually peaked very near 100,000 years. The Ra226 dose from 105-C surface contamination is still less than 1.0E-04 mrem/yr at 100,000 years. The dose from Nb93m from NRCDA Pad 2 is increasing very slowly at 100,000 years. For the MWMF and LLRWDF, the isotopes causing increasing doses at 100,000 years are Ra226 and Pb210 and for the NRCDA Zircaloy it is Nb94. The radionuclides Ra226, Pb210 and Nb93m, all of which have a short half-life compared to the 100,000 year evaluation period, appear because they are daughters of long-lived radionuclides. The maximum dose over 100,000 years from any of the 41 sources examined in this sensitivity study is the 1.04 mrem/yr from Np237 originating from H-Canyon/HB-Line, which has already been included in the CA dose analysis (Smith et al. 2009a). No doses found over the extended time period used in this sensitivity study would significantly impact CA doses.

Table 9-54. Maximum Doses over Three Time Intervals

Unit Name	Maximum Base Case Dose mrem/yr 2025 – 3025	Maximum Dose mrem/yr 2025 - 12050	Maximum Dose mrem/yr 2025 - 102025	10,000 yr Dose > 1,000 yr Dose (Base Case)	100,000 yr Dose > 10,000 yr Dose	Dose Increasing at 100,000 yrs
105-C SS	8.42E-08	6.41E-03	6.41E-03	✓		
105-C Surf	5.75E-02	5.75E-02	5.75E-02			✓
E-Area CIG	2.09E-02	2.09E-02	2.09E-02			
E-Area ILV	2.66E-05	1.81E-02	1.81E-02	✓		
NR Pad 1 Part 5	1.20E-03	1.69E-02	1.59E-01	✓	✓	
NR Pad 1 Part 6	3.66E-03	4.95E-02	3.57E-01	✓	✓	
NR Pad 2	3.65E-02	3.65E-02	3.65E-02			✓
E-Area ST Cent	1.33E-02	1.33E-02	2.02E-02		✓	
E-Area ST West	1.36E-04	2.99E-03	3.41E-03	✓	✓	
LLRWDF FMB	1.79E-02	1.79E-02	1.79E-02			✓
LLRWDF UTR	1.80E-02	1.80E-02	1.80E-02			✓
MWMF	1.35E-01	1.35E-01	1.35E-01			✓
ORWBG	7.28E-03	3.21E-02	3.46E-01	✓	✓	
221-F	2.75E-03	2.75E-03	2.86E-03		✓	
281-3F	8.34E-03	8.34E-03	8.34E-03			
F-Area Seep	2.24E-03	3.48E-03	1.55E-02	✓	✓	
FTF Type III	1.43E-03	1.43E-03	1.43E-03			
F-Area Old Seep	1.16E-02	1.16E-02	2.52E-02		✓	
HTF Type III FMB	2.49E-03	2.49E-03	2.49E-03			
HTF Type IIIA FMB	3.89E-03	3.89E-03	3.89E-03			
H-Area IPSL UTR	3.69E-03	3.69E-03	3.69E-03			
H-Area IPSL FMB	4.36E-03	4.36E-03	4.36E-03			
H-Area Seep	1.52E-02	1.52E-02	1.03E-01		✓	
H-Canyon	1.04E+00	1.04E+00	1.04E+00			
HAOM	2.02E-02	2.02E-02	2.02E-02			
TPBAR	2.44E-02	2.44E-02	2.44E-02			
KAC SS	4.16E-06	4.45E-03	4.45E-03	✓		
KAC Con	2.17E-03	2.17E-03	2.17E-03			
KAC Surf	5.30E-02	5.30E-02	5.30E-02			
LAC SS	1.42E-05	1.10E-03	1.10E-03	✓		
LAC Con	8.24E-03	8.24E-03	8.24E-03			
LAC Surf	7.57E-02	7.57E-02	7.57E-02			
105-P SS	1.59E-05	1.14E-03	1.14E-03	✓		
105-P Con	1.54E-02	1.54E-02	1.54E-02			✓
105-P Surf	1.14E-01	1.14E-01	1.14E-01			
105-R Con S	3.26E-03	3.26E-03	3.26E-03			
105-R Surf S	4.68E-02	4.68E-02	4.68E-02			
R-Area Seep	3.84E-05	1.74E-03	1.74E-03	✓		
TNX Outfall	4.61E-04	1.30E-03	1.83E-03	✓	✓	
Z-Area Vault 1	7.71E-03	7.71E-03	7.71E-03			
Z-Area Vault 4	4.93E-02	4.93E-02	4.93E-02			

9.4.3 Probabilistic Uncertainty Analysis

9.4.3.1 Uncertainty Analysis Approach

An estimate of the uncertainty in the dose calculated in the base case analysis presented by Smith et al. (2009a) was obtained by further analyzing the most significant sources at each POA. These sources, defined as those giving a maximum dose of 0.05 mrem/yr or greater in the deterministic base case, were identified by Smith et al. (2009a). The 17 sources meeting this criterion are listed in Table 9-55 and were used for the CA uncertainty analysis. The block in the lower right hand corner of Table 9-55 shows the percent of the total sum of maximum doses at each POA represented by the sources used for the uncertainty analysis. This measure shows that the uncertainty analysis includes the most significant dose sources.

Table 9-55. Major Dose Sources at Each POA Identified in CA

Source Name	Max Dose mrem/yr	Source Name	Max Dose mrem/yr
105-C Surf	5.75E-02	HANM	6.07E-02
H-Area Seep	1.03E-01	UTR IOU	7.19E-02
ORWBG AgI	1.63E-01	MWMF	1.35E-01
FMB IOU	1.97E+00	H-Canyon	1.04E+00
Fourmile Branch	2.29E+00	Upper Three Runs	1.31E+00
105-R Surf S	4.68E-02	SR-B IOU	1.38E-02
LTR IOU	2.94E+00	SR-A IOU	1.76E-02
Lower Three Runs	2.99E+00	Savannah River	3.14E-02
KAC Surf	5.30E-02	POA	% Total
PB IOU	6.00E-02	Fourmile Branch	98%
LAC Surf	7.57E-02	Lower Three Runs	99%
105-P Surf	1.14E-01	Steel Creek/Pen Branch	95%
SC IOU	2.43E-01	Upper Three Runs	83%
Steel Creek/Pen Branch	5.46E-01	Savannah River	98%

To facilitate the uncertainty analysis, a special version of the CA GoldSim™ model was created that allowed up to five sources to be run simultaneously. Each source has independent definitions of parameters such as inventory, source release mechanism, vadose zone, and aquifer path. However, each source is connected to the same POA so only one dose calculation is made from the combined stream and river radionuclide concentrations. Figure 9-145 shows how this model was constructed within GoldSim™. This version of the model was run for each POA using the sources listed in Table 9-55. If less than five sources were needed, the unused sources remained connected in the model with nominal flow paths but no inventory was specified so there was no contribution to the dose.

An initial version of this multi-source model, allowing up to ten units to be connected, was tested by comparing the results obtained with ten combined sources to that obtained by running the sources separately and then summing the doses. Figure 9-146 shows the total dose and doses from individual radionuclides for the combined model while Figure 9-147 shows the same doses obtained by summing the results from running the ten sources individually. A comparison of the figures shows that the doses are identical thereby verifying the combined model. The ten sources used in this test were: E-Area Components in Grout, Engineered Trench, Intermediate Level Vault, Low Activity Waste Vault, Slit Trench East, Slit Trench Center, and Slit Trench West, F-Area Old Seepage Basin, H-Area Seepage Basin, and TPBAR. This model was later reduced to five connected units to reduce run time and storage requirements because, as shown in Table 9-55, five units were the most that were required for the CA uncertainty analysis.

Using this model, a probabilistic uncertainty assessment was performed at each POA where the total dose (mrem/yr) was computed for a simulation time period of 2075 years (i.e., from calendar year 1950 to 4025). Time steps of 0.5 years and 2.0 years were employed for the time periods 0-75 years and 75-2075 years, respectively. Total dose by all pathways at a POA was chosen for our probabilistic assessment measure. For a specific POA, multiple simulations (i.e., Monte-Carlo realizations) were performed by GoldSim™ where a subset of CA Units discharging to each POA was chosen. For each POA, as shown in Table 9-55, the largest contributors to total dose based on the best estimate (i.e., deterministic) dose were used for the uncertainty analysis. A minimum dose threshold value of 0.05 mrem/yr was used to limit the total number of units that had to be considered.

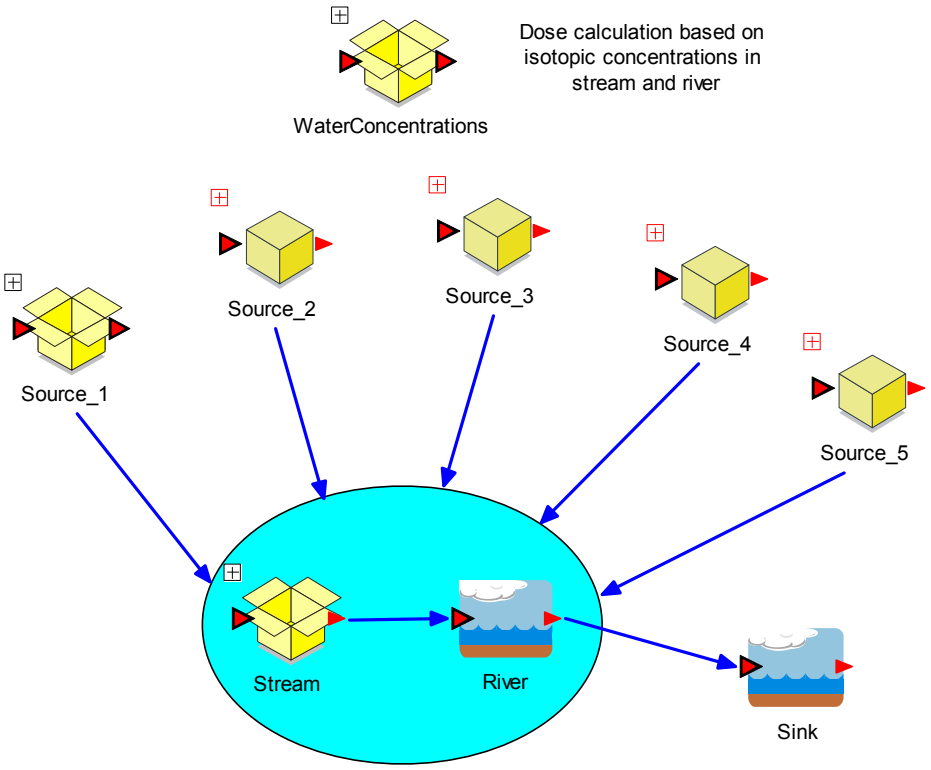


Figure 9-145. Five Source GoldSim™ Model Used for CA Uncertainty Analysis

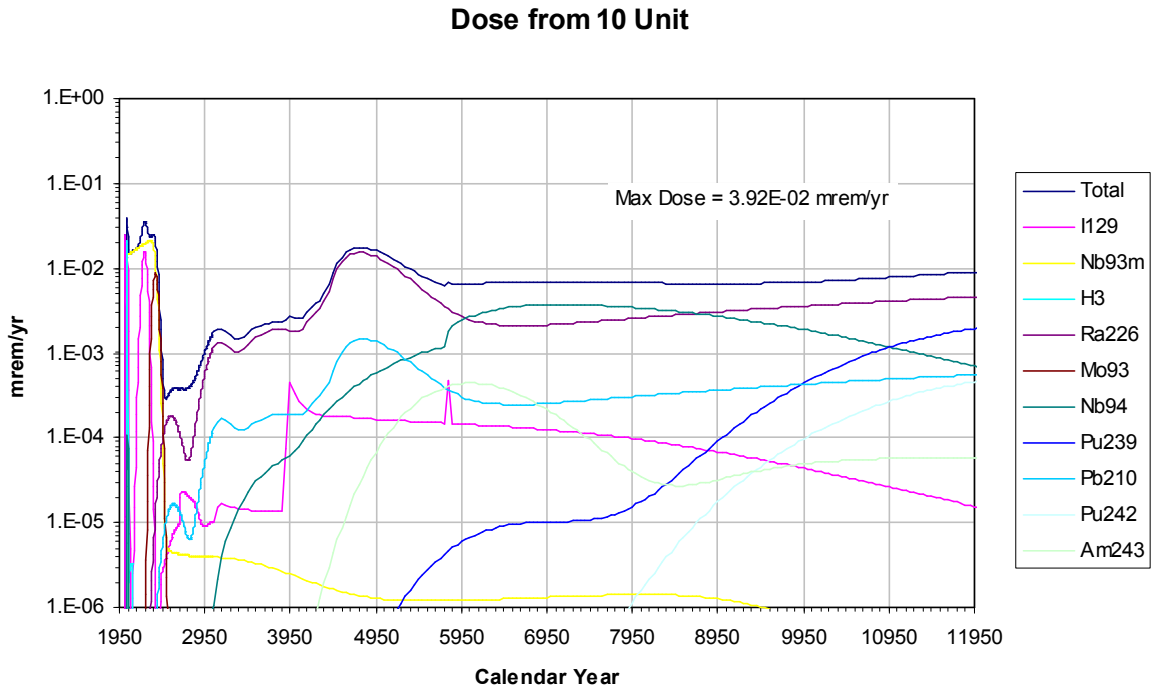


Figure 9-146. Dose from Radionuclides Obtained with 10 Connected Sources

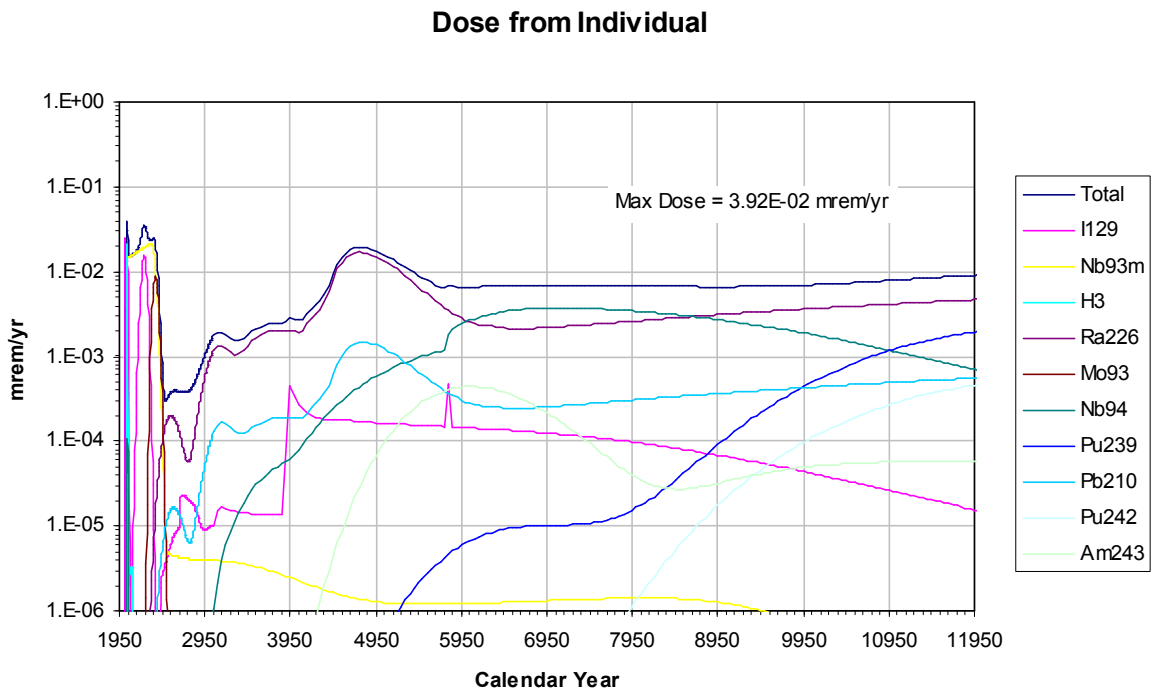


Figure 9-147. Dose from Radionuclides Obtained by Summing 10 Separate Calculations

9.4.3.2 Parameter Distributions

Parameters in the CA model that were assigned stochastic distributions are listed in Tables C-10, C-18, and C-19 in Appendix C. For probabilistic Monte-Carlo calculations, GoldSim™ randomly chooses a value for each of these parameters from the associated distribution. Stochastic distributions that appear in the dose module (Phifer et al. 2009) are given in Table C-18, distributions for material property values (Phifer and Dixon 2009) are listed in Table C-10, and distributions for flow parameters and concrete aging (Phifer and Dixon 2009 and Hamm et al. 2009) are listed in Table C-19. In all cases, the mean or, for triangular distributions, the most likely value was the value used for base case deterministic calculations (Smith et al. 2009a). The cap infiltration distributions listed in Table C-19 are applied as multipliers to the nominal cap infiltration flow calculated as a function of time (Smith et al. 2009a).

Similarly, the aquifer sand and clay velocity "ratio" distributions shown in Table C-19 are factors applied to the nominal Darcy velocities. Results obtained by performing the aquifer flow parameters analysis (Hamm et al. 2009) with three GSA Flow Model solutions at high, low and nominal water table levels were used to address the uncertainty associated with aquifer travel times. For deterministic GoldSim™ calculations, all of the aquifer parameters were set to their nominal values. To address travel time uncertainty, all of the variability was assigned to the pore velocities keeping the aquifer travel lengths constant. Normalized travel time ratios through sand and clay were defined as:

$$\eta_s = t_s / t_s^N \quad \text{and} \quad \eta_c = t_c / t_c^N$$

where the superscript N indicates values obtained from the "nominal" GSA flow solution. Assuming no variability in travel length, variability in the pore velocity is expressed through the use of these travel time ratios as:

$$u_s = u_s^N / \eta_s \quad \text{and} \quad u_c = u_c^N / \eta_c \ .$$

Given probability density functions for the travel time ratios, these equations give pore velocity distributions that address the overall uncertainty associated with travel time through the aquifer (i.e., both uncertainties in pore velocity as well as travel length) for each material type. For use in GoldSim™, a log-normal distribution function was chosen to reflect the sand and clay travel time uncertainties. To reduce pore velocities to their nominal values when deterministic calculations were performed, the log-normal parameters were constrained to have mean values of unity. This also produced distributions that were skewed to lower values which is consistent with the aquifer modeling results. More details of this analysis are provided in Appendix E of Hamm et al. (2009).

The only other distributions used in the CA model were applied to K_d values. Following McDowell-Boyer and Kaplan (2009), it was assumed that K_d values have log-normal distributions and that for sandy soil and cement the standard deviation is 75% of the mean value ("best value" in Table C-12) and for clayey soil the standard deviation is 50% of the mean value (also "best value" in Table C-12).

Probabilistic simulations were performed with GoldSim™ using the Monte-Carlo algorithm under the following options:

- Stochastic variables were chosen using a Latin Hypercube sampling algorithm.
- Sampling sequences are repeatable with the initial random seed set to 1.

Simulations were also performed using the uncertainty GoldSim™ model with the stochastic variables set to their deterministic values. The majority of the deterministic value settings were set to the mean of the stochastic variable distribution. However, for some of the stochastic variables associated with the dose calculation, their deterministic (most likely) values were set to the minimum or maximum extreme values of their distributions. This approach caused the mean of the probabilistic calculations to differ from the deterministic result.

In summary a total of 75 parameters were assigned distributions. Thirty-four of the parameters were related to the two exposure scenarios used in the dose calculations, (e.g., consumption rates of various foodstuffs and times spent performing various activities), twenty-six of the parameters were physical properties of the materials needed to conduct transport calculations (e.g., porosity and density), twelve parameters defined the flow rates in aquifers and surface streams in the model, and three parameters were used to define the changing properties of concrete barriers through time.

9.4.3.3 Number of Realizations Required

For each POA, a set of multiple simulations typically using a large number of realizations must be run for GoldSim™ to compute, as a function of time, the following statistical parameters for total dose: mean value, lower bound, 5th Percentile, 25th Percentile, 50th Percentile (median value), 75th Percentile, 95th Percentile, and upper bound. Best estimate values can also be computed using GoldSim™ in its deterministic mode. These statistical parameter values will vary as the number of realizations is increased and will ultimately converge to stationary values at some number of realizations. To establish an acceptable number of realizations to use for CA probabilistic GoldSim™ runs, a large number of realizations (1,000) were performed for a single CA source. The CA source from the E-Area MWMF discharging into UTR was chosen for this purpose. Outside of GoldSim™, the results from these 1,000 realizations were processed to compute the statistical parameters listed above as a function of the number of realizations considered from 1-50 realizations to 1-1,000 realizations in increments of 50. This type of sensitivity analysis provides guidance as to the number of realizations required to produce acceptable statistical parameter values.

The time histories for computed upper bound and median total doses are shown in Figure 9-148 for three sets of realizations: (1) the first 50 realizations (1-50); (2) the first 400 realizations (1-400); and (3) all 1,000 realizations (1-1,000). The peak value for each statistical parameter as a function of the number of realizations performed is shown in Figure 9-149. As illustrated in Figure 9-149, acceptable estimates of the statistical parameters were achieved with about 400 realizations. Based on this analysis, 500 realizations within GoldSim™ were used for the POA probabilistic assessments presented in the next section.

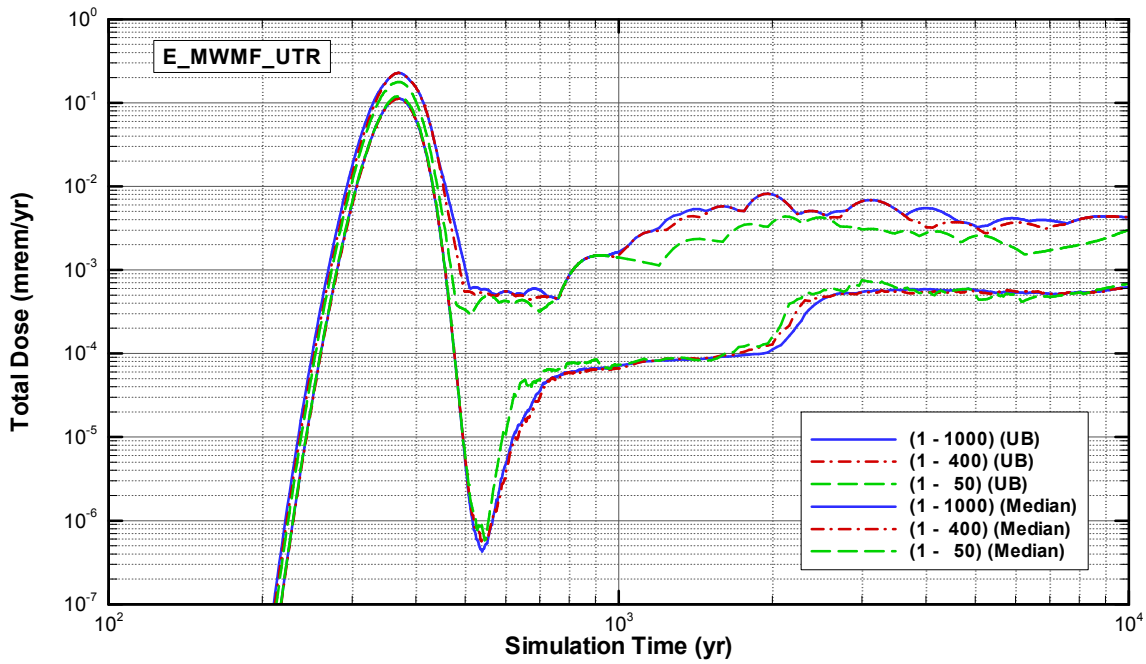


Figure 9-148. Upper Bound and Median Estimates of Total Dose at UTR from E-Area MWMF as a Function of the Number of Realizations

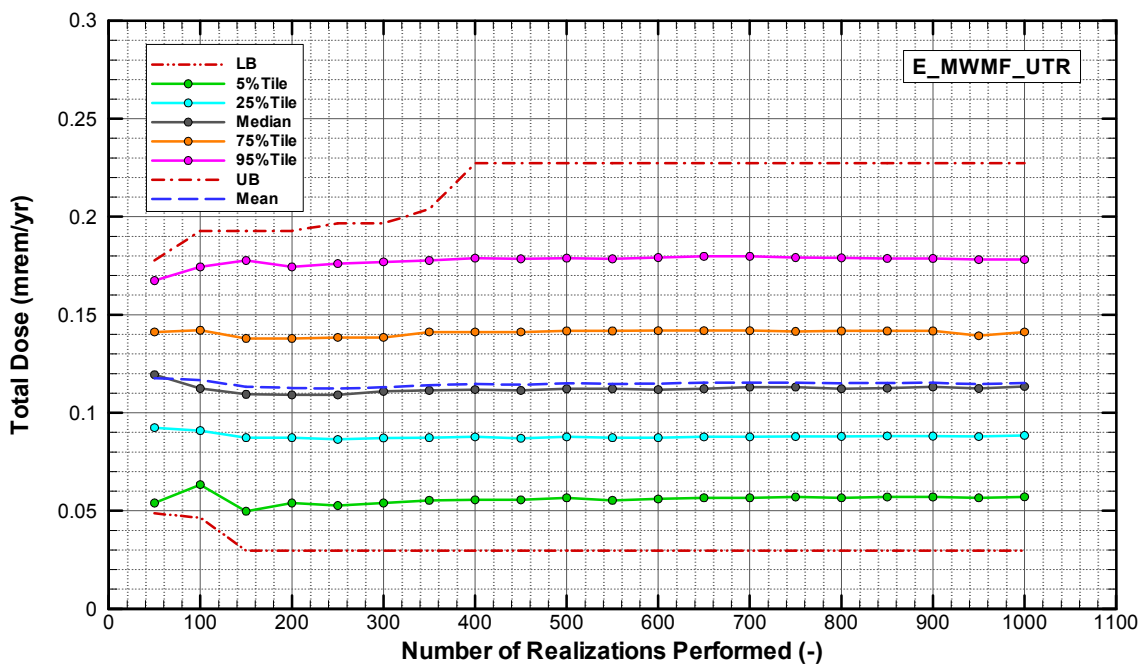


Figure 9-149. Statistical Parameters for Total Dose at UTR from E-Area MWMF as a Function of the Number of Realizations

9.4.3.4 Uncertainty Results at each POA

Probabilistic assessments using 500 realizations within GoldSim™ were performed for the five POAs and associated sources listed in Table 9-55. Deterministic runs were also performed using consistent simulation settings for a direct comparison to the probabilistic results. Peak values of statistical parameters for total dose from direct discharges to each POA over the CA assessment period from year 2025 to year 3025 and over the longer 2075-year simulation period from 1950 to 4025 are provided in Table 9-56 and Table 9-57 respectively. The higher values in Table 9-57 for LTR, FMB, SC/PB and the Savannah River are caused by higher doses from Cs137 releases from stream sediments that occur prior to the start of the CA period of assessment in 2025.

Results from the uncertainty analysis for each POA are plotted in Figure 9-150 through Figure 9-159. Two graphs are provided for each POA. The first graph shows the statistical distribution of the probabilistic results over time and the second graph compares mean, median and deterministic results. As can be seen in Figure 9-150 through Figure 9-159, the mean and median curves are different because some of the parameter distributions are asymmetric. For some of the dose parameter distributions, the selected "Best Estimate" values used in the deterministic simulations do not coincide with either the mean or median values of their distributions. Therefore, as observed in Figure 9-151, Figure 9-153, Figure 9-155, Figure 9-157, and Figure 9-159, the best estimate curves differ from both the mean and median curves. The abrupt drop in dose for the FMB POA in about year 3250 seen in Figure 9-154 and Figure 9-155 is caused by depletion of soluble I129 from the ORWBG.

Figure 9-160 compares median values (50th Percentile) of total dose obtained from the probabilistic calculations for all five POAs over the entire 2075 years of simulation. In Figure 9-161, a comparison of the 95th Percentile values for all five POAs is shown focusing in on the 1,000-year CA assessment period from 2025 to 3025. Doses are plotted in linear coordinates to better compare their values to the 30 mrem/yr dose criterion. The estimated doses prior to the start of the assessment period are also provided to better illustrate the dose behavior near their peak values. The exponential drop off in total dose observed in Figure 9-160 and Figure 9-161 is due to the decay of cesium-137 released from the stream beds. At the 95% level, only the maximum doses of 11.5 mrem/yr at LTR and 8.2 mrem/yr at FMB are greater than 10% of the 30 mrem/yr dose constraint.

All of the probabilistic uncertainty calculations are for the dose resulting from direct discharges to each POA. The residential component of the total dose calculated at a POA is not based on a cumulative concentration at the POA that considers upstream contributions to radionuclide concentrations.

Table 9-56. Peak Values of Statistical Parameters for Total Dose at Each POA over the CA Assessment Period

Statistical Parameter (mrem/yr)	Upper Three Runs	Lower Three Runs	Fourmile Branch	Steel Creek/Pen Branch	Savannah River
Best Estimate (Deterministic)	1.04	2.95	2.14	0.46	0.03
Mean	0.60	4.64	3.25	0.72	0.05
Lower Bound*	0.01	0.43	0.33	0.06	0.01
5th Percentile	0.04	1.08	0.78	0.17	0.02
25th Percentile	0.08	2.33	1.62	0.35	0.03
50th Percentile (Median)	0.37	3.58	2.48	0.56	0.04
75th Percentile	0.92	6.02	4.14	0.91	0.06
95th Percentile	2.43	11.6	8.16	1.75	0.10
Upper Bound*	5.59	24.8	18.0	3.71	0.20

*The upper bound is the envelope of maximum values at each point in time that bounds all of the 500 realizations. Similarly, the lower bound is the envelope of minimum values at each point in time that bounds all of the 500 realizations. All of the doses calculated in the uncertainty analysis fall within the upper and lower bounds.

Table 9-57. Peak Values of Statistical Parameters for Total Dose at Each POA over the 2075 Years Simulated

Statistical Parameter (mrem/yr)	Upper Three Runs	Lower Three Runs	Fourmile Branch	Steel Creek/Pen Branch	Savannah River
Best Estimate (Deterministic)	1.04	2.95	2.14	0.46	0.03
Mean	0.60	8.21	5.59	1.14	0.08
Lower Bound*	0.02	0.73	0.46	0.10	0.02
5th Percentile	0.05	1.86	1.31	0.26	0.04
25th Percentile	0.10	4.03	2.73	0.56	0.05
50th Percentile (Median)	0.37	6.17	4.26	0.86	0.07
75th Percentile	0.92	10.7	7.23	1.50	0.10
95th Percentile	2.43	20.9	14.2	2.92	0.17
Upper Bound*	5.59	49.0	33.1	6.63	0.36

*The upper bound is the envelope of maximum values at each point in time that bounds all of the 500 realizations. Similarly, the lower bound is the envelope of minimum values at each point in time that bounds all of the 500 realizations. All of the doses calculated in the uncertainty analysis fall within the upper and lower bounds.

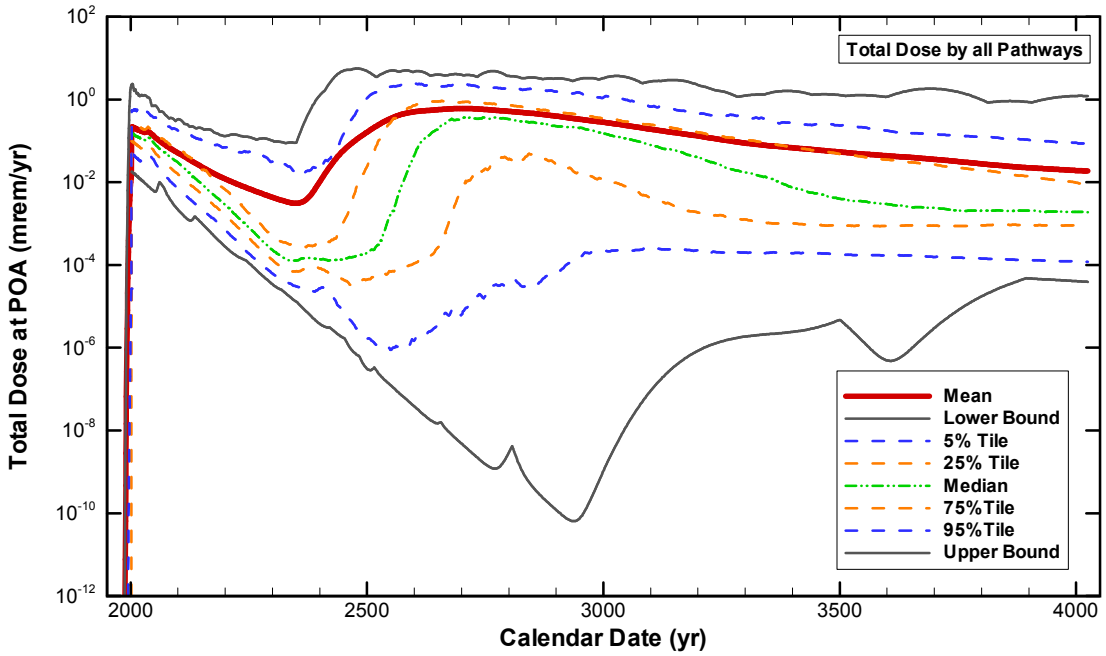


Figure 9-150. Confidence Levels in Computing Total Dose by All Pathways at Upper Three Runs POA

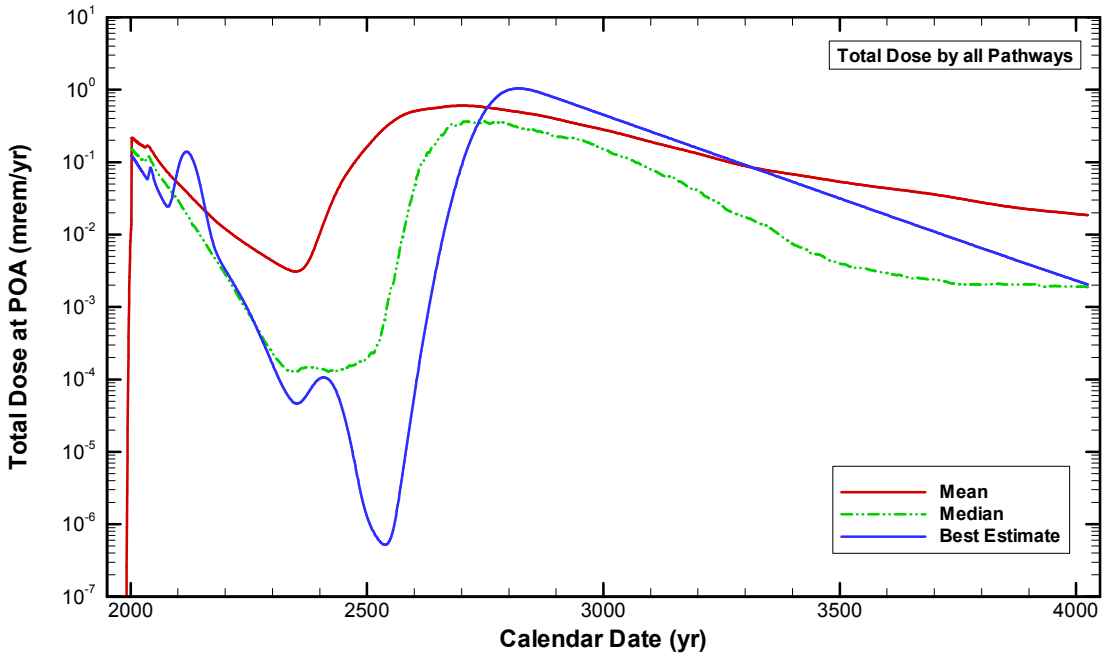


Figure 9-151. Comparison of Best Estimate, Mean, and Median Values in Dose by All Pathways at Upper Three Runs POA

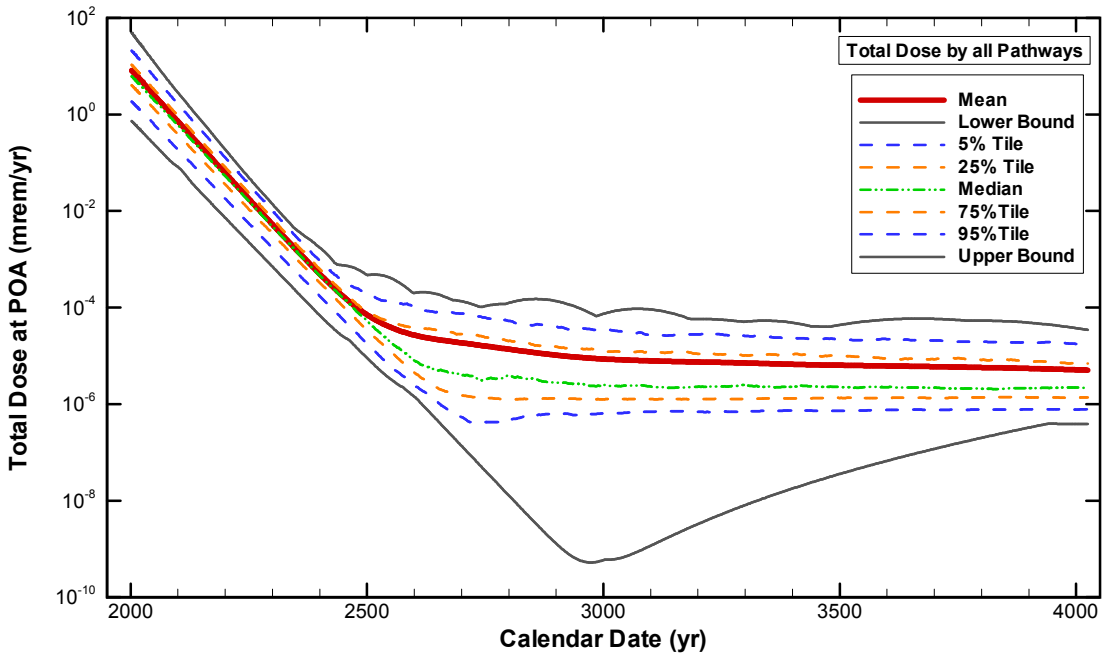


Figure 9-152. Confidence Levels in Computing Total Dose by All Pathways at Lower Three Runs POA

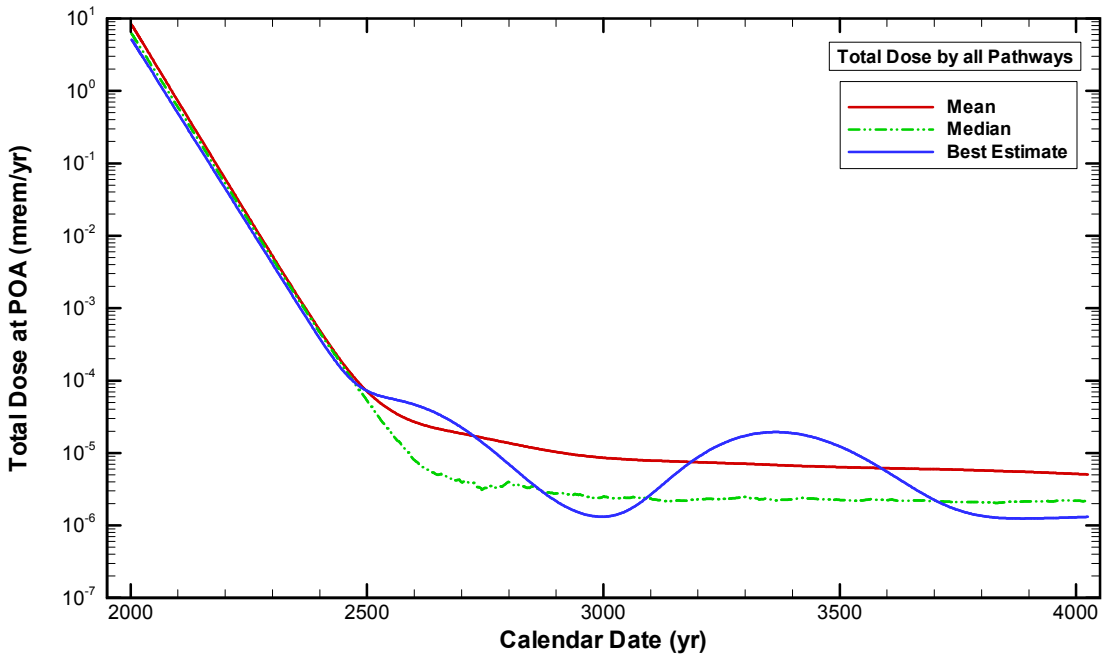


Figure 9-153. Comparison of Best Estimate, Mean, and Median Values in Dose by All Pathways at Lower Three Runs POA

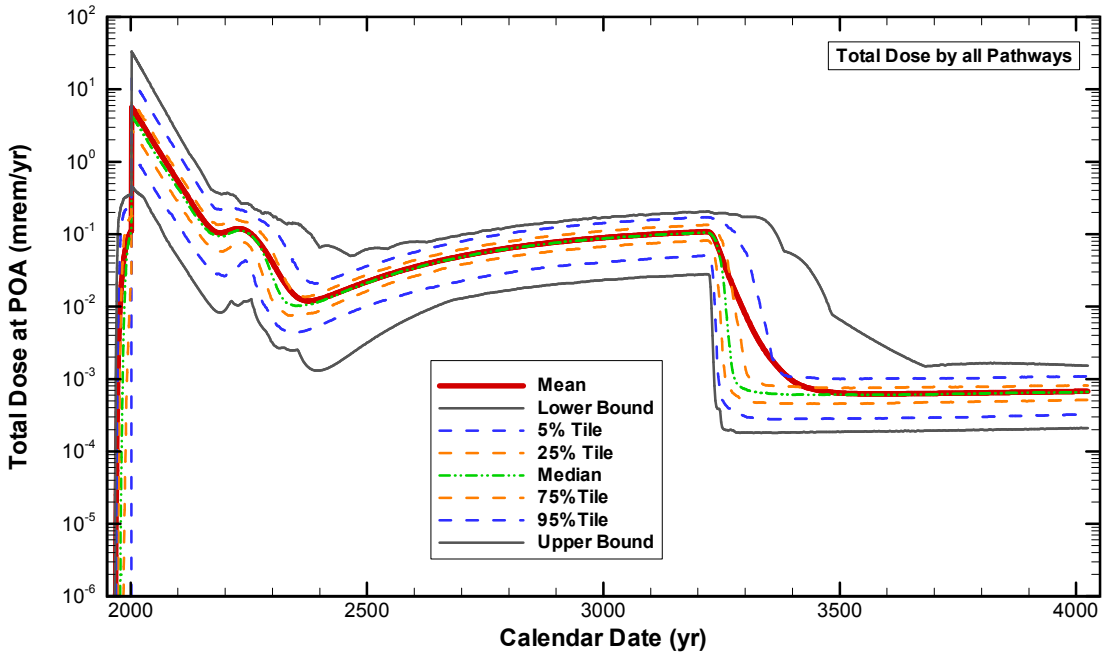


Figure 9-154. Confidence Levels in Computing Total Dose by All Pathways at Fourmile Branch POA

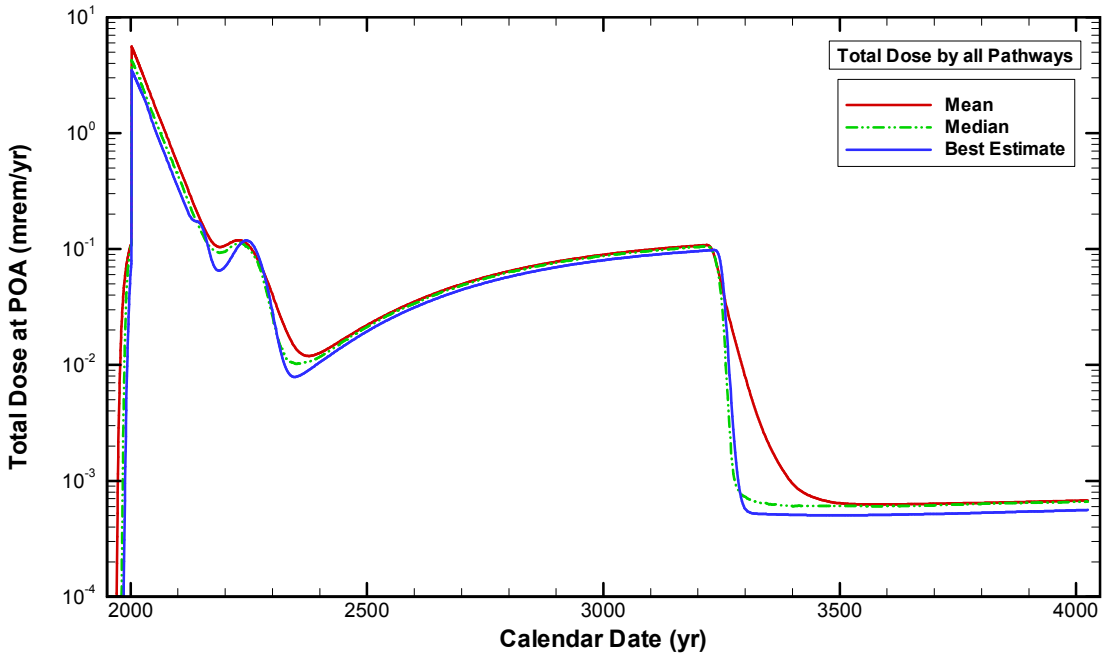


Figure 9-155. Comparison of Best Estimate, Mean, and Median Values in Dose by All Pathways at Fourmile Branch POA

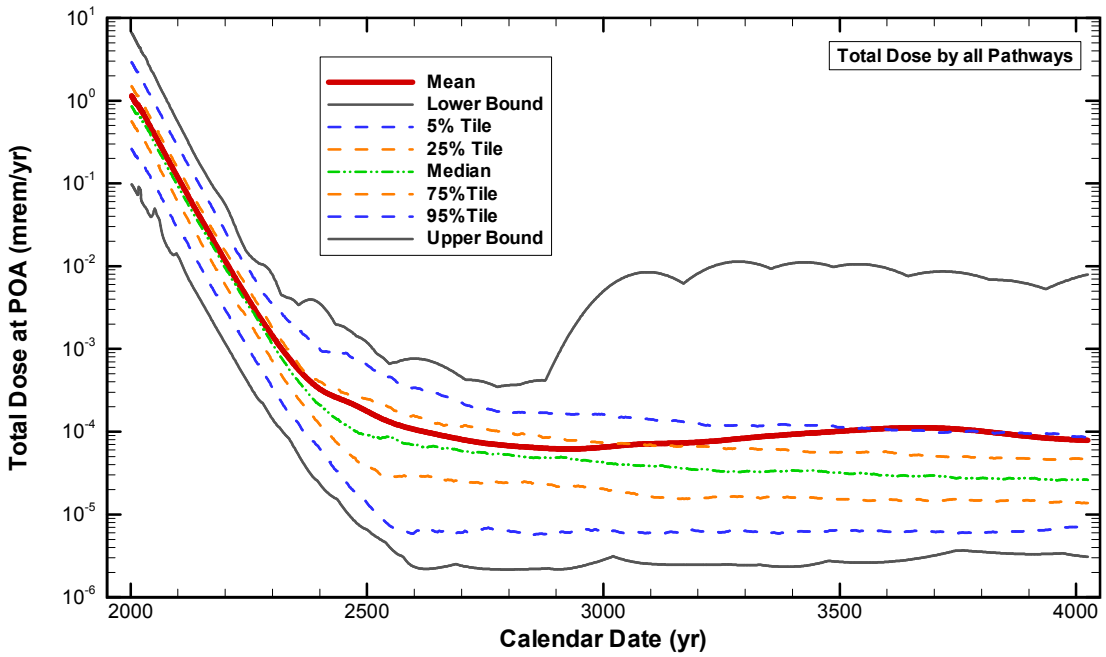


Figure 9-156. Confidence Levels in Computing Total Dose by All Pathways at Steel Creek/Pen Branch POA

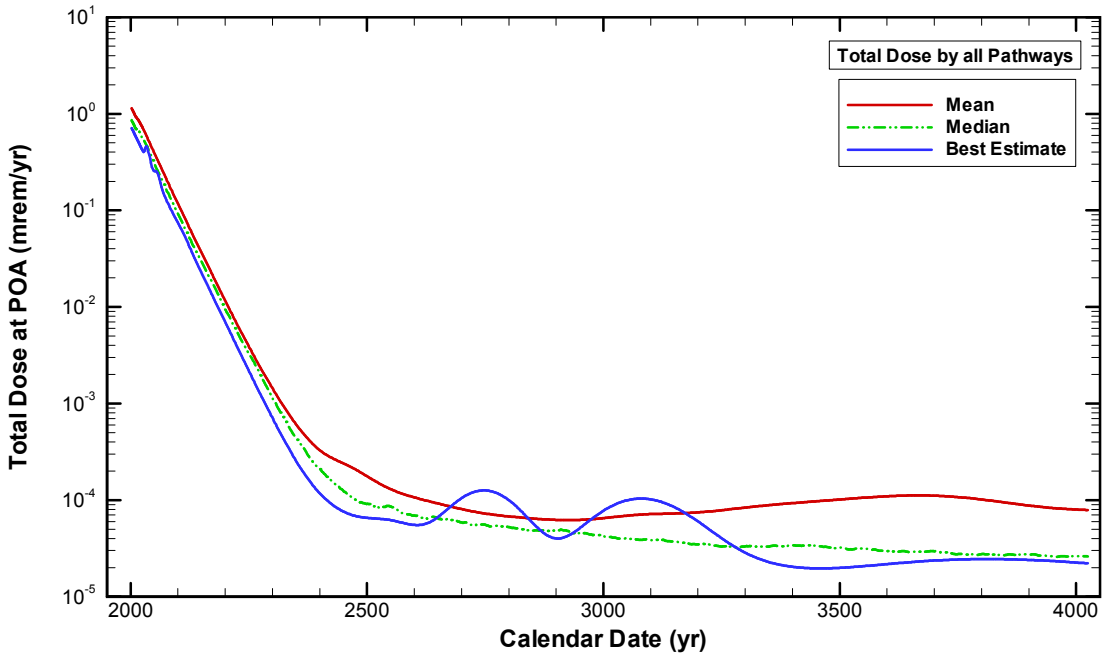


Figure 9-157. Comparison of Best Estimate, Mean, and Median Values in Dose by All Pathways at Steel Creek/Pen Branch POA

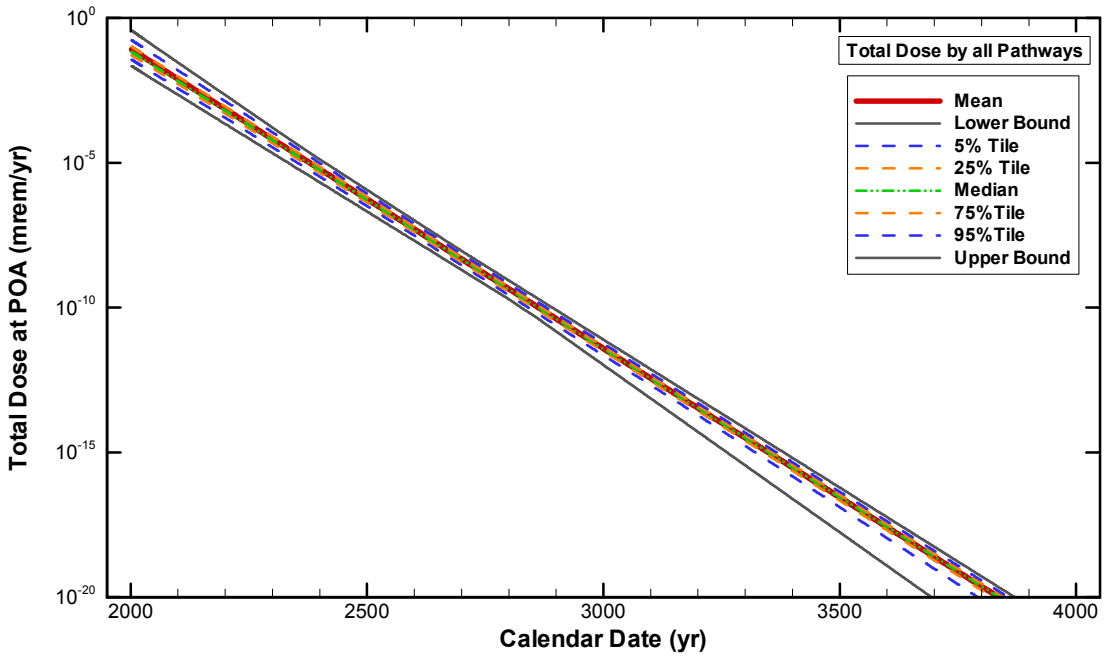


Figure 9-158. Confidence Levels in Computing Total Dose by All Pathways at Savannah River POA

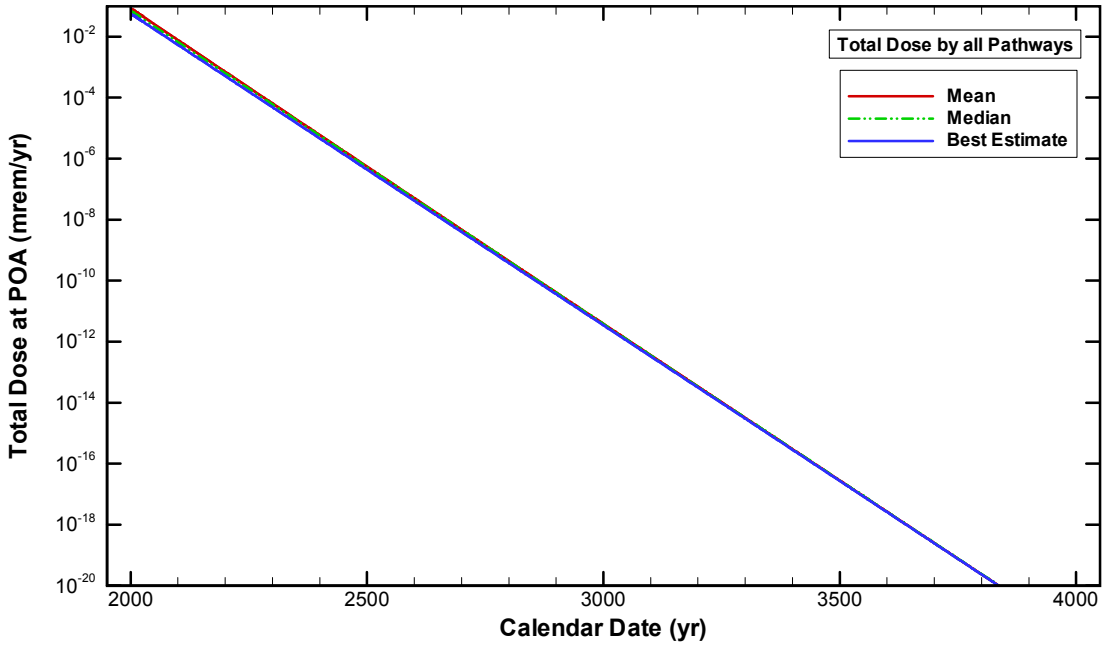


Figure 9-159. Comparison of Best Estimate, Mean, and Median Values in Dose by All Pathways at Savannah River POA

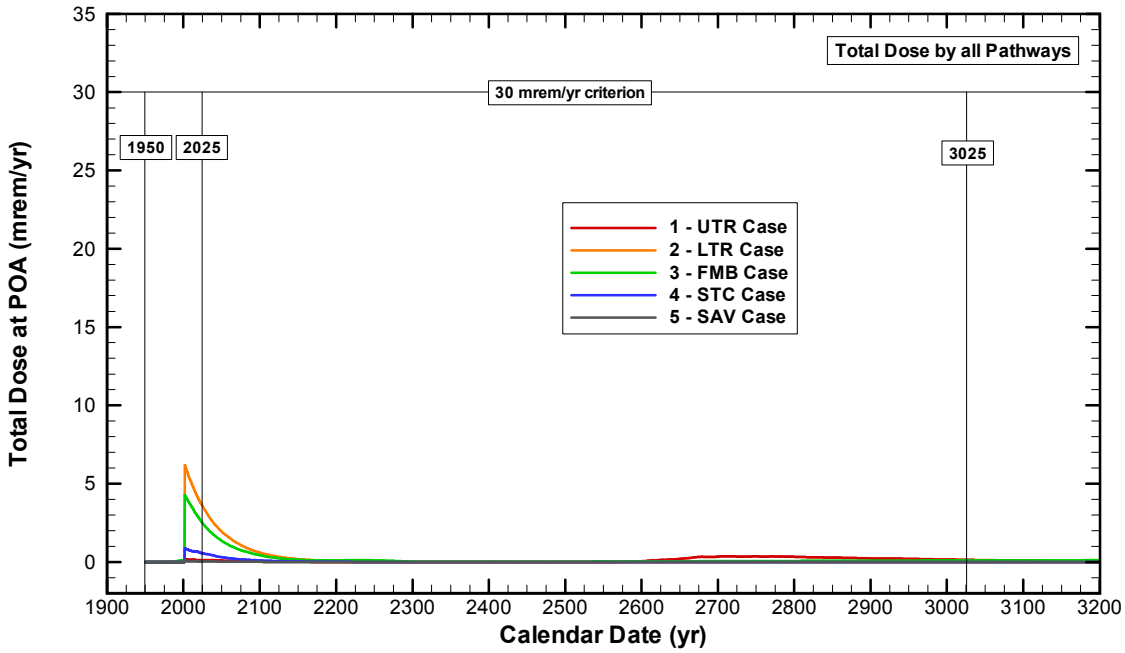


Figure 9-160. Comparison of Median Values in Dose by All Pathways for Five POAs during CA Period of Assessment

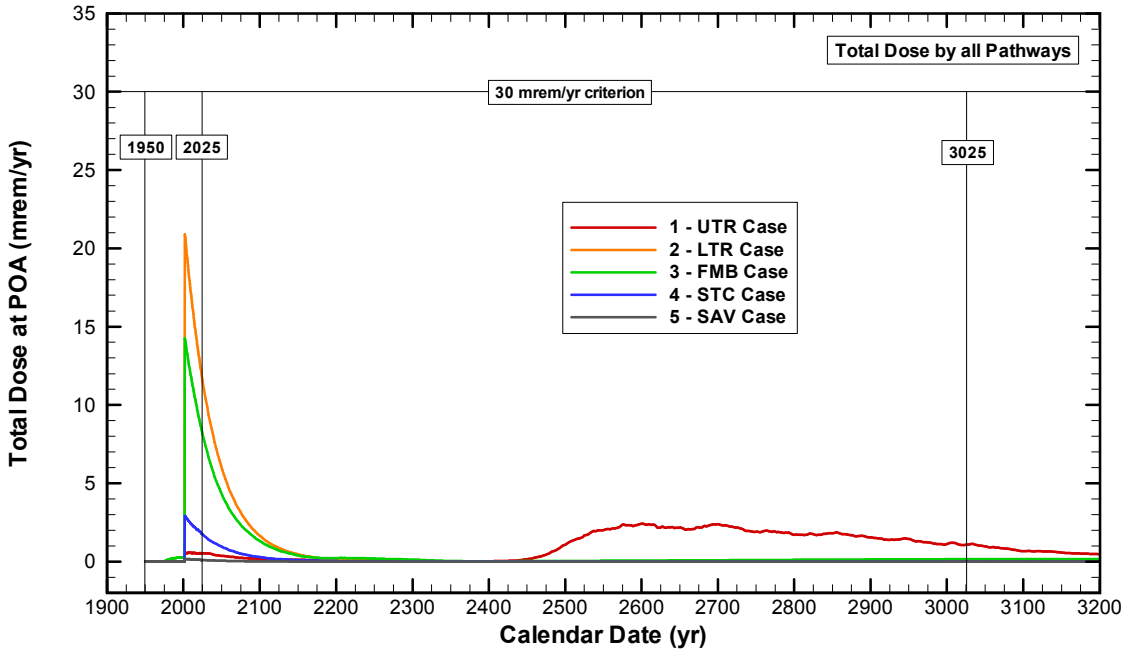


Figure 9-161. Comparison of 95th Percentile Values in Dose by All Pathways for Five POAs During CA Assessment Period

9.4.3.4.1 Confidence Levels in Peak Total Dose at each POA

For the five POAs listed in Table 9-55, the peak total dose for each of the 500 realizations was determined within the CA assessment period of 1,000 years (calendar year 2025 to calendar year 3025). Histograms for these peak total doses were computed and are shown in Figure 9-162 through Figure 9-166. Adequate distribution shapes were achieved when 40 histogram bins were chosen for this analysis. The lower bound, 5th Percentile, 25th Percentile, 50th Percentile (median), 75th Percentile, 95th Percentile, and upper bound confidence levels are listed in Table 9-56. For comparison purposes, the best estimate (deterministic) and 95th Percentile values are shown in Figure 9-162 through Figure 9-166 as red and blue bars, respectively.

These total dose estimations are lower bound estimates of the true total doses for each POA. As indicated by Table 9-55, only those units with deterministic maximum doses exceeding 0.05 mrem/yr were considered in the probabilistic analyses. However, it is expected that these lower bound estimates are reasonably close predictors of the total dose values that would be obtained if all CA sources discharging to each POA are accounted for. It is expected that including additional CA sources will only marginally increase the statistical parameter values.

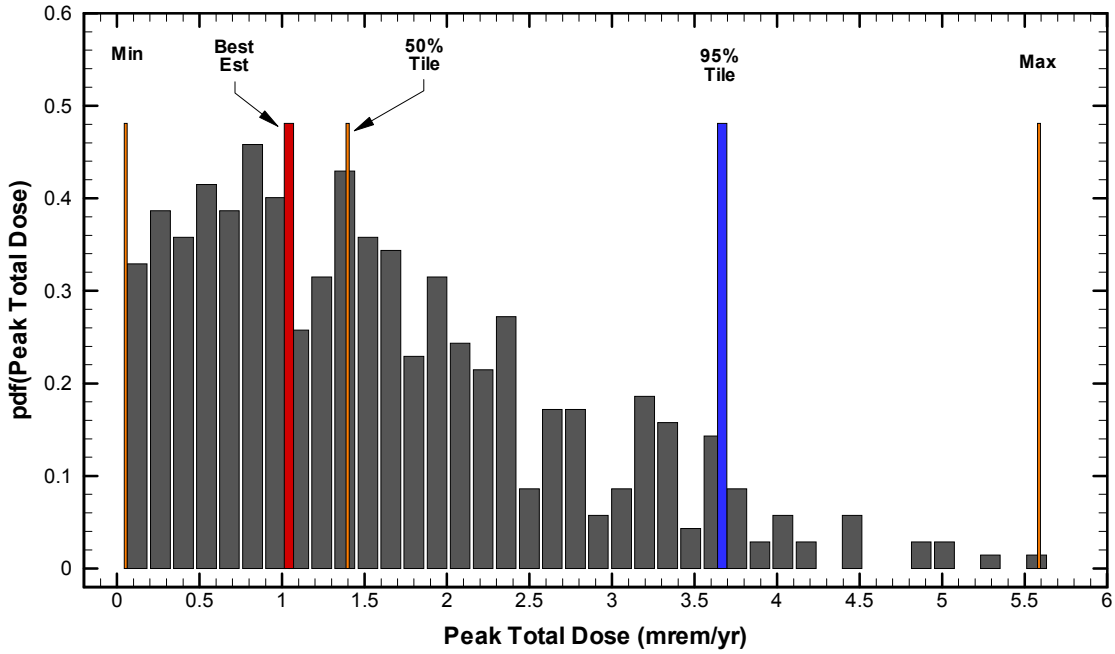


Figure 9-162. Distribution of Peak Total Dose by All Pathways at the Upper Three Runs POA for CA Assessment Period

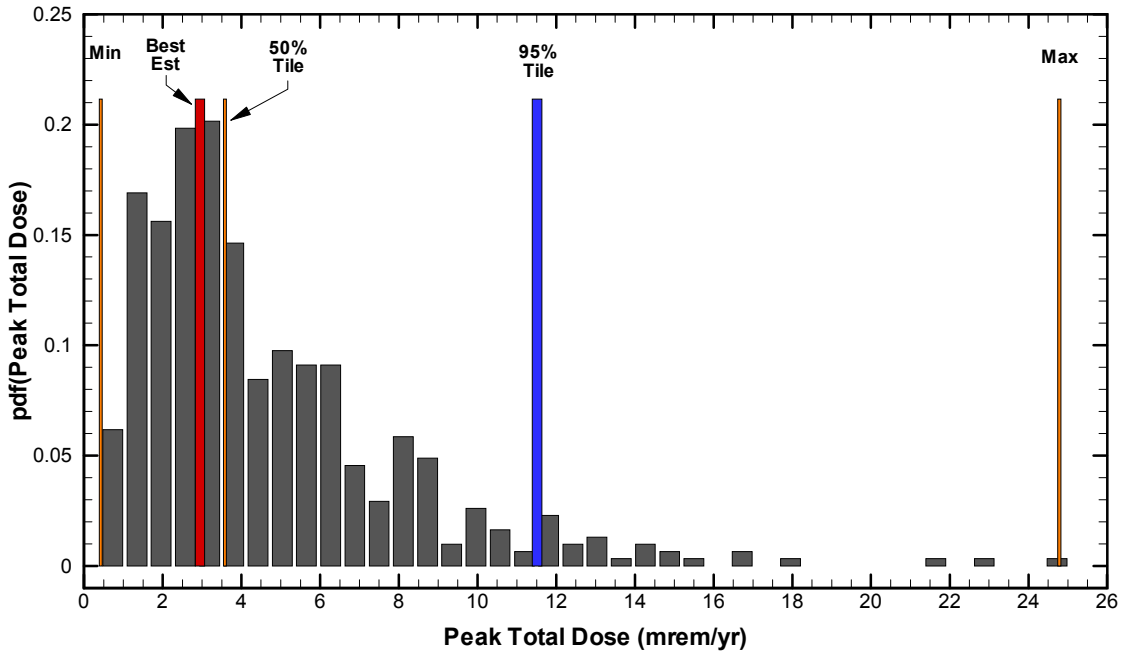


Figure 9-163. Distribution of Peak Total Dose by All Pathways at the Lower Three Runs POA for CA Assessment Period

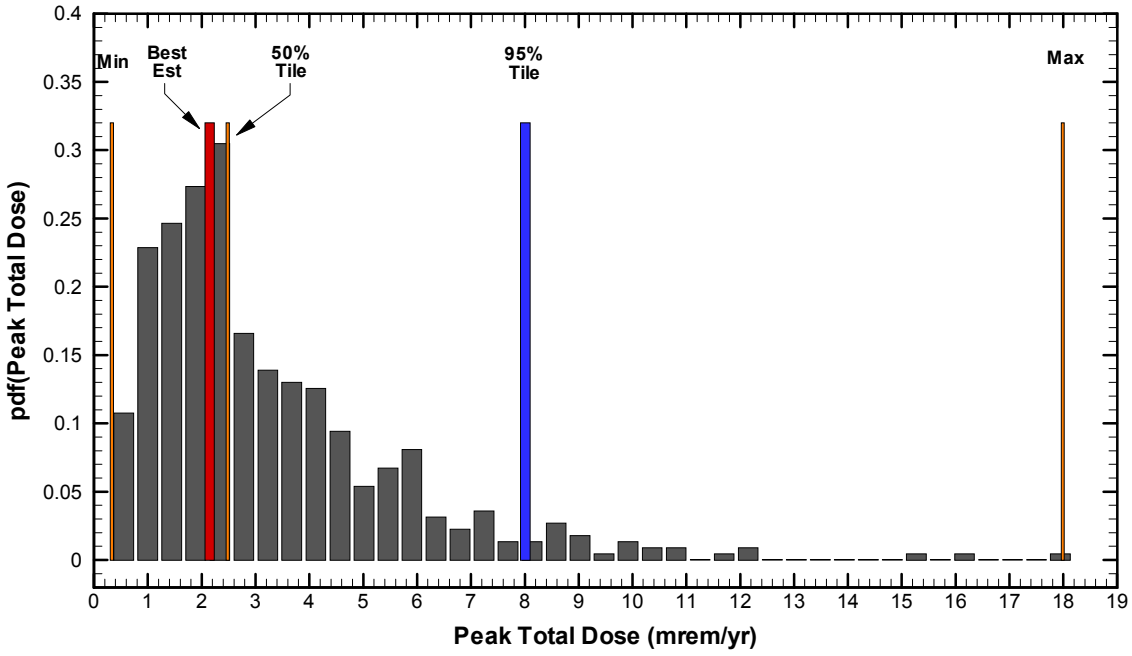


Figure 9-164. Distribution of Peak Total Dose by All Pathways at the Fourmile Branch POA for CA Assessment Period

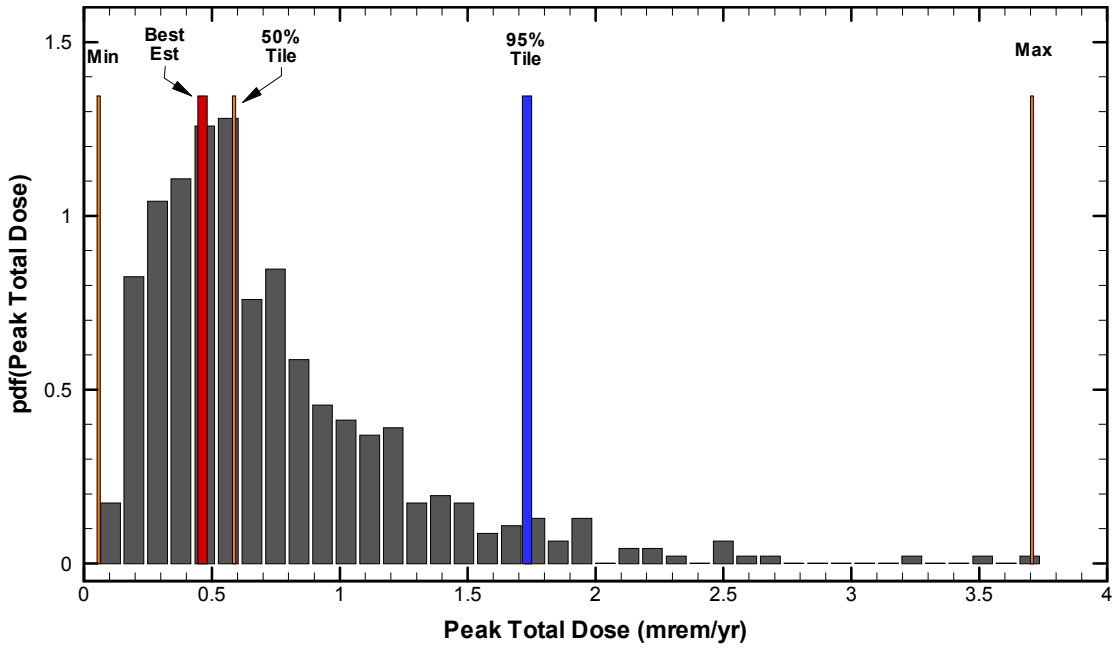


Figure 9-165. Distribution of Peak Total Dose by All Pathways at the Steel Creek/Pen Branch POA for CA Assessment Period

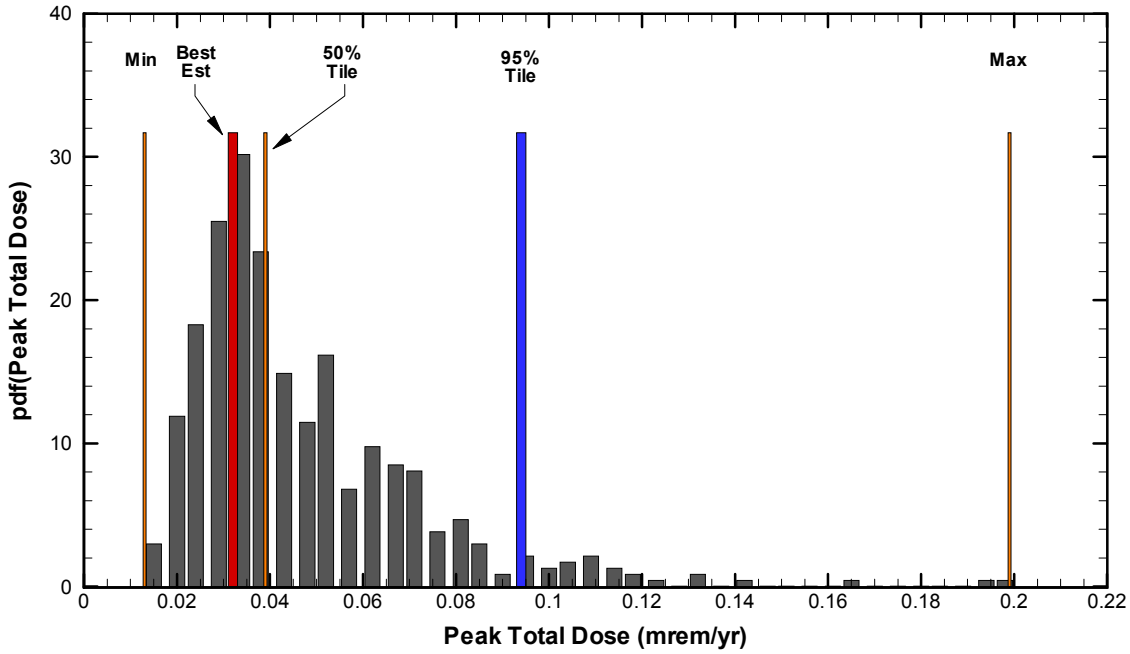


Figure 9-166. Distribution of Peak Total Dose by All Pathways at the Savannah River POA for CA Assessment Period

9.4.4 Sensitivity and Uncertainty Analysis Summary

This CA section presents results from sensitivity and probabilistic calculations designed to provide a measure of the uncertainty in the deterministic dose results. The sensitivity cases were chosen to address the requirements in the LFRG Manual (DOE 2008) to consider factors such as release rates, radionuclide inventories, alternative points of assessment, groundwater divides, stream flow variation, and alternative disposal actions in sensitivity and uncertainty analysis (Phifer et al. 2008a). The principle results are shown for each of the eight sensitivity cases in Table 9-58 below.

Table 9-58. Principle Results of Sensitivity Cases

Sensitivity Case	Principle Result	
Source Release Modeling	Sum of Peak Dose with Source Release Modeling of Surface Sources = 2 mrem/yr	Sum of Peak Dose without Source Release Modeling of Surface Sources = 37.5 mrem/yr
Stream and River Flow	Maximum at 7Q10 (Low) Flow = 10.1 mrem/yr (LTR POA)	
Alternative POA (subset of Stream and River Flow Sensitivity Cases)	Maximum dose of 9.7 mrem/year results for the alternative Fourmile Branch POA at the edge of the SRS industrial core	
Inventory	Minimum Multiplier to reach 30 mrem/yr = 10 (LTR POA)	
Groundwater Divide	Move in groundwater divide toward UTR results in maximum dose increase multiplication factor of 42.2 for H-Tank Farm (although the dose increases, it still remains relatively low)	
Alternative End State Date	Changing the End State Date from 2025 to 2050 results in either no change or a reduction in dose at each POA	
C14 Bioaccumulation Factor	No effect during 1,000-yr CA assessment period	
Aquifer Clay	Flow path clay elimination results in an average dose increase multiplication factor of 1.4	
Maximum Dose over 100,000 years	Maximum Total Dose Attributed to any Individual Unit Source = 1.04 mrem/yr (H-Canyon)	

The following paragraphs provide a brief summary of each of the eight major sensitivity categories and the results:

- Source Release Modeling Sensitivity Case: Modeling source release mechanisms that account for waste forms, barriers, and engineered caps for 19 CA surface sources gave a sum of the peak dose of 2.0 mrem/yr (Smith et al. 2009a). By comparison, the sum of the peak doses during the CA assessment period from these same 19 surface sources without source release modeling, where the inventory was released directly onto the ground surface, gave a total dose of 37.5 mrem/yr (Smith et al. 2009a). These 19 CA surface sources, listed in Table 9-23 in Section 9.3.7, were the only surface sources where source release modeling was employed, thereby introducing some conservatism into the base case results.
- Stream and River Flow Sensitivity Cases: Maximum total doses over the 1,000-year assessment period for the stream and river flow sensitivity cases that were performed are shown in Table 9-42. These results demonstrate that the sustained reductions in flow from current annual average rates that are required to give doses approaching the 30 mrem/yr dose constraint would be so great as to not be credible.
- Alternative POA: The alternative POA sensitivity is a subset of the stream and river flow sensitivity cases. This sensitivity case was run assuming that the extent of controlled land decreased to encompass only the central, industrial core of the site. This put the POA at the boundary of the site industrial zone for both the residential and recreational exposure scenarios. As shown in Table 9-42, this results in a maximum total dose of 9.7 mrem/year at the alternative Fourmile Branch POA during the CA assessment period.
- Inventory Sensitivity Cases: The primary inventory sensitivity case determined the inventory multiplier for each individual source in each POA that would be required to reach a cumulative POA dose of 30 mrem/yr over the 1,000-year assessment period, keeping all of the other sources for that POA at their base case inventory. Only three sources (H-Canyon, FMB IOU, and LTR IOU) were found to have an inventory multiplier less than 100. The multiplier on overall inventory required to produce a maximum total dose of 30 mrem/yr for the combined sources associated with each POA was also determined and was presented in Table 9-47. The lowest overall inventory multiplier is 10 for the LTR POA. This sensitivity demonstrates that the total inventory would have to be at least an order of magnitude greater than the best estimate inventory utilized in the CA Base Case deterministic evaluation (Smith et al. 2009a) to result in doses of any significant concern. Table 9-47 also presented results from other selected inventory sensitivity cases. Other inventory sensitivity cases were basically insignificant except for that of the H-Canyon/HB-Line. This sensitivity indicates that the level of radionuclide removal during deactivation and decommissioning (D&D) of the H-Canyon/HB-Line is a significant consideration from a CA perspective.

- Groundwater Divide Sensitivity Case: This sensitivity analysis assumed that the groundwater divide between UTR and FMB moved such that the entire LLRDWF/MWMF, FTF, and HTF discharged solely to FMB rather than to both UTR and FMB. Due to the lower flow in FMB this assumption results in an increased dose at the POA. Table 9-50 summarizes the results of this study. The doses associated with FTF and HTF discharging solely to FMB were both less than 0.02 mrem/yr. The dose associated with LLRDWF/MWMF increased from 0.1 mrem/yr to 0.7 mrem/yr when discharging solely to FMB. It is unlikely that such a drastic change in the groundwater divide location would occur, particularly in the direction of UTR, because UTR has incised deeper and has a greater flow rate than FMB.
- Alternative End-State Date Sensitivity Case: As the end-state date moves out from 2025 to 2050, maximum total doses over the 1,000-year CA assessment period decrease by approximately 50% at the FMB, LTR and SC/PB POAs and remained essentially unchanged at the UTR and SR POAs. The results of this sensitivity case are summarized in Table 9-51. The decrease in dose was caused by decay of Cs137 released from the stream beds which caused the highest doses at the FMB, LTR and SC/PB POAs. These maximum doses occur at the start of the CA assessment period.
- C14 BCF Sensitivity Case: The C14 bioaccumulation factor was varied from the base case site specific value of 3 L/kg (Hinton et al. 2009) to 4,500 L/kg (NCRP 1985) and 50,000 L/kg (Staven et al. 2003). Maximum doses during the CA assessment period did not change when the higher C14 BCFs were used. The largest C14 inventory (4,000 Ci) was in the ORWBG. Assuming a BCF of 50,000 L/kg, the maximum dose from C14 originating from the ORWBG was about 45 mrem/yr at the FMB POA in the year 4500, which is beyond the CA period of assessment. The base case C14 BCF of 3 L/kg is based on data from FMB and takes into account the ecosystem carbon dynamics that is dominated by C12 rather than C14.
- Aquifer Clay Sensitivity Case: The 28 CA sources with clay in the aquifer path that produced a maximum dose greater than 0.001 mrem/yr during the CA assessment period, were run replacing the clay with saturated sandy soil. The dose at UTR more than doubled with the elimination of clay primarily due to the H-Canyon/HB-Line source where the maximum dose increased from 1.0 to 2.8 mrem/yr. Results of this sensitivity case are summarized in Table 9-53. Sharpening of the Np237 peak concentration in the absence of clay caused the increase in dose from H-Canyon/HB-Line and similar proportional increases in small doses from the F-Area Tank Farm Type III Tanks and 221-F.
- Maximum Doses over 100,000 Years Sensitivity Case: The 41 CA sources having base case maximum doses greater than 0.001 mrem/yr during the 10,025-year time period from year 2025 to year 12,050 were further run to year 102,050 to determine if maximum doses had been seen. This sensitivity case determined that 16 of the CA sources have maximum doses outside the 1,000-year CA assessment period. However, no doses found over the extended time period approached the 30 mrem/yr CA dose criterion.

Probabilistic calculations were performed to assess the uncertainty in doses calculated from direct discharge of radionuclides to each POA. A total of 75 parameters were assigned distributions. Thirty-four of the parameters were related to the two exposure scenarios used in the dose calculations, (e.g., consumption rates of various foodstuffs and times spent performing various activities), twenty-six of the parameters were physical properties of the materials needed to conduct transport calculations (e.g., porosity and density), twelve parameters defined the flow rates in aquifers and surface streams in the model, and three parameters were used to define the changing properties of concrete barriers through time.

Table 9-56 in Section 9.4.3.4 presents a summary of these results. These calculations were performed for each POA considering all sources that contributed greater than 0.05 mrem/yr to the total maximum dose. At the 95th Percentile, the maximum doses from direct discharge at each POA are: 2.4 mrem/yr at UTR, 11.5 mrem/yr at LTR, 8.2 mrem/yr at FMB, 1.8 mrem/yr at SC/PB and 0.1 mrem/yr in the Savannah River. These probabilistic calculations provide a high level of confidence that the 30 mrem/yr dose constraint will not be exceeded.

9.5 RELATIONSHIP TO PERFORMANCE ASSESSMENT MODELING

9.5.1 Summary

The conceptual models developed for the various Savannah River Site (SRS) Performance Assessments (PA) were considered during development of the conceptual model for the Composite Analysis (CA). PA modeling has been conducted for the E-Area Low-Level Waste Facility (ELLWF), the F-Tank Farm (FTF), and the Saltstone Disposal Facility (SDF) as outlined in Section 3.4.1, Section 3.4.2, and Section 3.4.4, respectively. Because the CA does not need to assess the use of contaminated groundwater (Sections 4.2.5 and 7.1.2), and the CA points of assessment are more distant from the sources of residual radioactive material, the CA modeling is simpler than that utilized in the PA. Input data (e.g., inventory, hydrogeological data) for the CA was drawn from the same sources as that used in the PAs with the exception of the stream and river flows, which were not used in the PAs. For the most significant PA disposal units (i.e., those giving doses greater than 0.1 mrem/yr with the generic surface release mechanism), the CA used the flux to the water table calculated in the PAs. Figure 9-167 illustrates the relationship between this CA and the SRS PAs.

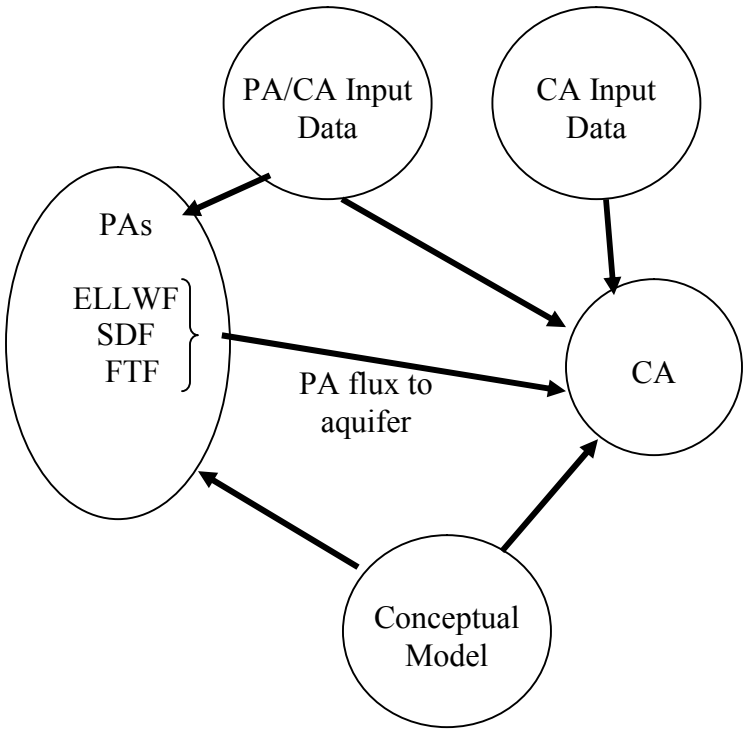


Figure 9-167. Relationship between CA and SRS PAs

9.5.2 CA Review Criteria

Criteria from the Low-Level Waste Disposal Facility Federal Review Group Manual (DOE 2008) pertinent to assuring consistency between the CA and PAs are provided below. Only those criteria or portions of criteria that apply to the relationship between the CA and PAs are given here (Appendix B contains all of the CA review criteria and where this CA has met them). The criteria are shown in italics and a description of the way in which the relationship between the two types of analyses is demonstrated in this CA follows.

3.3.2.4 Source terms and flow and transport models in the CA are commensurate with the available data consistent with the PA, incorporate the important characteristics identified in the PA, and provide outputs consistent with the PA.

The source term utilized in the CA for PA facilities has been defined in one of three ways:

- As the PA inventory released directly to the top of the vadose zone (i.e., generic release) with background infiltration and K_d-mediated transport through the vadose and saturated zones to the streams
- As the intermediate PA flux to the water table results (i.e., source release) with K_d-mediated transport through the saturated zone to the streams
- As a CA-calculated flux to the water table (i.e., source release) based on PA inventory with K_d-mediated transport through the saturated zone to the streams

The PA flux to the water table was utilized as the source term for those PA facilities whose generic release results resulted in a maximum dose at the Point of Assessment (POA) greater than or equal to 0.1 mrem/yr (see Section 9.3.6). The PA facilities for which the PA flux to the water table was used as the CA source term were the E-Area Intermediate Level (IL) Vault, the E-Area Low Activity Waste (LAW) Vault, and the SDF Vault 2 units.

The CA-calculated flux to the water table was utilized for the NRCDA 643-7E only, using PA-consistent corrosion parameters. All other PA source terms were represented as the PA inventory released directly to the top of the vadose zone.

The same flow model (Flach 2004) utilized to develop the flow fields for each of the PA models was also used to define the flow properties utilized in the CA model. This process is described in Section 9.2.1 and Section 9.3.4.4 and in Hamm et al. 2009 in more detail.

3.3.5.1 Assumptions incorporated into the analysis, including those related to the radionuclides to be considered, the inventories of radionuclides, the source term evaluation and the transport of radionuclides, are identified, justified, and consistent with the conceptual model of site behavior presented in the PA conducted on the LLW disposal facility.

The CA radionuclide screening outlined within Section 7.2.2 considered the same initial list of 826 radionuclides screened in the ELLWF and FTF PAs, with an additional 23 radionuclides added to the list, and ensured that the key radionuclides identified within these PAs were also included for modeling within the CA. The SDF PA was not available at the time that the CA radionuclide screening was conducted, but it was not expected that the SDF would require the addition of radionuclides to the initial list. Radionuclide half-lives, molecular weights, decay chains, and branching fractions were consistent in both the PA and CA modeling.

The inventories utilized within the CA for the PA facilities were taken from the PAs themselves or from the same source documents utilized by the PAs (see Sections 8.2.3.1, 8.2.4.7, and 8.2.10 respectively, for the inventories for the ELLWF, FTF, and SDF facilities). As indicated above, the source term utilized in the CA for PA facilities has been defined as either the PA inventory released directly to the top of the vadose zone, as the PA flux to the water table results, or as a CA-calculated flux reliant on the PA inventory. Transport within the vadose and saturated zones for both the PAs and the CA is defined as K_d-mediated transport. A brief discussion of the ELLWF, FTF, and SDF PA conceptual models can be found in Sections 3.4.1.5, 3.4.2.5, and 3.4.4.5, respectively, for comparison to that of the CA, which is discussed within Section 9.1.3.

Additionally, vadose zone thicknesses used in the CA modeling of PA facilities were taken from the PAs. Both modeling efforts used the same rates of background infiltration and infiltration through clay caps (ELLWF) and geosynthetic caps (FTF). Material properties for sandy and clayey soil and for concrete were the same in both CA and PA models.

In the case of the Naval Reactor Components Disposal Area (NRCDA), discussed in Section 9.3.1.6.4.2, the GoldSim™ model of metal corrosion used for the CA calculations was, in its essential features, identical to the GoldSim™ ELLWF PA model.

3.3.6.3 The models used in the CA provide calculated results that are representative of the results calculated in the PA for similar wastes in similar disposal facilities.

The PA-modeled fluxes of each CA radionuclide into the aquifer at each time step for the E-Area IL Vault, the E-Area LAW Vault, and the SDF Vault 2 units were collected and used as direct input into the CA model and therefore represent the PA results exactly to that point. The mechanics of how this was accomplished are explained in Section 9.3.1.6.6. Using PA fluxes in the CA calculations implicitly accounts for source release mechanisms from vaults, tanks, and waste forms considered in the PA calculation. As outlined above, generic release results were used within the CA for PA facilities whose generic release, maximum dose at the Point of Assessment (POA) was less than 0.1 mrem/yr.

3.3.6.5 The input data to the models are based on field data from the site, laboratory data interpreted for field applications, referenced literature sources which are applicable to the site, or related analyses performed for the PA. Any assumptions used to formulate input data are justified and have a defensible technical basis.

Input data for the CA model were taken from referenced sources based on site-specific field and laboratory studies and literature sources. Most of the data for facilities for which a PA has been done were taken from the same sources as those used for that PA or were compiled from PA input data sets. CA data categories, references and usage relative to the PAs are given in Table 9-59.

Table 9-59. Major CA Data Types and References

Data Type	Reference	Used in a PA?
Facility Inventory	Hiergesell et al. 2008	Y
Stream flow rates	Jones 2009	N
Material properties, infiltration rates, and saturation estimates and distributions	Phifer and Dixon, 2009 Shine 2008	Y
Corrosion rate of casks and activated metal	Smith et al. 2009	Y
Vadose zone properties	Noffsinger et al. 2009	Y
Aquifer flow properties	Hamm et al. 2009	Y
K_d and CDP values	McDowell-Boyer and Kaplan 2009	Y
PA model flux to the water table results input to CA model	Smith et al. 2009	Y
Radionuclide data	Phifer et al. 2009	Y
Transfer factors, dose conversion factors, exposure parameters, and consumption rates	Phifer et al. 2009	Y

3.3.6.6 Intermediate calculations are performed, and the results are presented to demonstrate the CA calculations are representative of the site and are consistent with results presented in the PA for similar situations.

With the exception of the NRCDA 643-7E, isotope fluxes to the aquifer taken directly from the PA calculations were used in the CA model for PA facilities requiring source release modeling. This method ensures that the CA calculation is consistent with the PA as much as possible. Use of the PA flux accounts for source release factors in the CA without specific modeling, because the source release has already been taken into account when calculating the PA fluxes. Time histories of species fluxes from the PA calculation are entered into the CA model where contaminant species enter the aquifer. This method was used for the E-Area ILV, E-Area LAWV and Z-Area Vault 2 units.

3.3.6.7 The conceptual model used for the CA is consistent with the representation of the conceptual model used in the PA, and includes the major mechanisms affecting the transport of radionuclides at the DOE site. The components of the conceptual model for the CA are reasonably represented in the analysis of the LLW disposal facility and other contributing sources.

A brief discussion of the ELLWF, FTF, and SDF PA conceptual models can be found in Sections 3.4.1.5, 3.4.2.5, and 3.4.4.5, respectively, for comparison to that of the CA discussed within Section 9.1.3. These descriptions include the radionuclides analyzed, release mechanisms, the basis for the vadose zone and aquifer conceptual models, and the groundwater discharge point (i.e., either Upper Three Runs or Fourmile Branch). The conceptual models for each of the facilities for which a PA has been completed have been appropriately considered within the CA model. Transport within the vadose and saturated zones for both the PAs and the CA is defined as K_d transport controlled.

3.3.7.1 The exposure pathways identified in the CA should be consistent with the exposure pathways in the PA.

The exposure pathways identified in the CA were selected using a screening process described in Wilhite and Phifer 2008. The CA and the PAs both use contaminated water as an exposure medium. In the CA surface water is the source, while the PA uses groundwater. However, some of the ways in which the water is used, such as drinking and irrigation, are the same in the two analyses. The resulting exposures due to direct radiation, ingestion and inhalation are calculated using the same equations using parameters appropriate to the different activities examined in CA and PAs (Phifer et al. 2009).

3.3.7.2 The dose analysis performed for the CA is consistent with that performed for the PA for similar exposure pathways and similar exposure scenarios.

The dose module for the CA was developed based upon the dose modules utilized within the ELLWF PA (WSRC 2008a), FTF PA (WSRC 2008b), and the CA radionuclide screening (Section 7.2). In fact the dose module (Phifer et al. 2009) developed for use in the CA has also been produced to support of a variety of PA efforts involving multiple facilities at the SRS. This approach assures consistency in dose calculations between the PA and CA programs.

3.3.10.2 The analysis and results of the CA are consistent with comparable results of the PA and provide a defensible and complete basis for an acceptable decision by DOE.

The point at which the PA and CA results could most effectively be compared would be at the top of the aquifer, which is a location in both models that values can be readily obtained. In this CA, however, those values from the PA models are used as input to the CA model for all sources potentially contributing more than 0.1 mrem/yr (with the exception of the NRCDA 643-7E), thus assuring consistency between the two efforts. For the NRCDA 643-7E, the CA modeling is based on the PA inventory and on source release calculations consistent with the PA.

All of the facilities at SRS for which PAs have been done are in the GSA and will eventually have radionuclides outcrop into either UTR or FMB. Figure 9-168 and Figure 9-169 show the total dose to each of these streams together with the total dose due to PA facilities alone. The scenarios used to calculate dose in the CA are most similar to the All Pathways scenario in the PAs, which have a dose limit of 25 mrem/yr. The two figures show that the calculated CA dose is orders of magnitude less than the PA limit, which is to be expected given that the CA POA is in a relatively distant stream rather than a relatively close 100 m well and includes dilution of the radionuclides in the stream. This is further evidence of consistency between the two efforts.

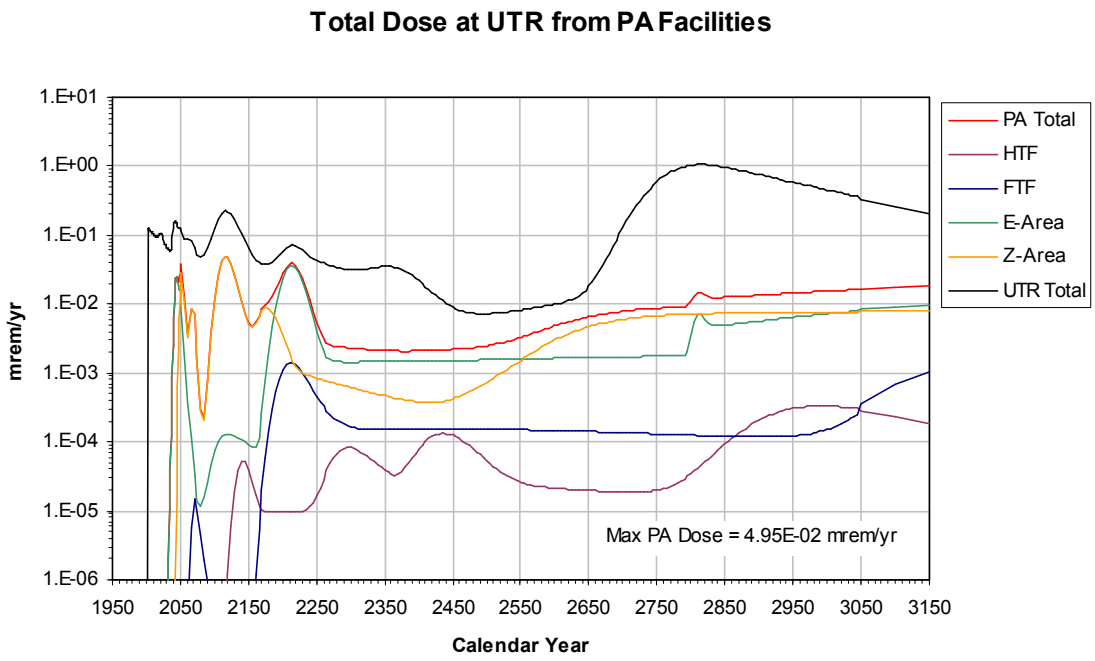


Figure 9-168. Contribution from PA Facilities to CA Dose at Upper Three Runs POA

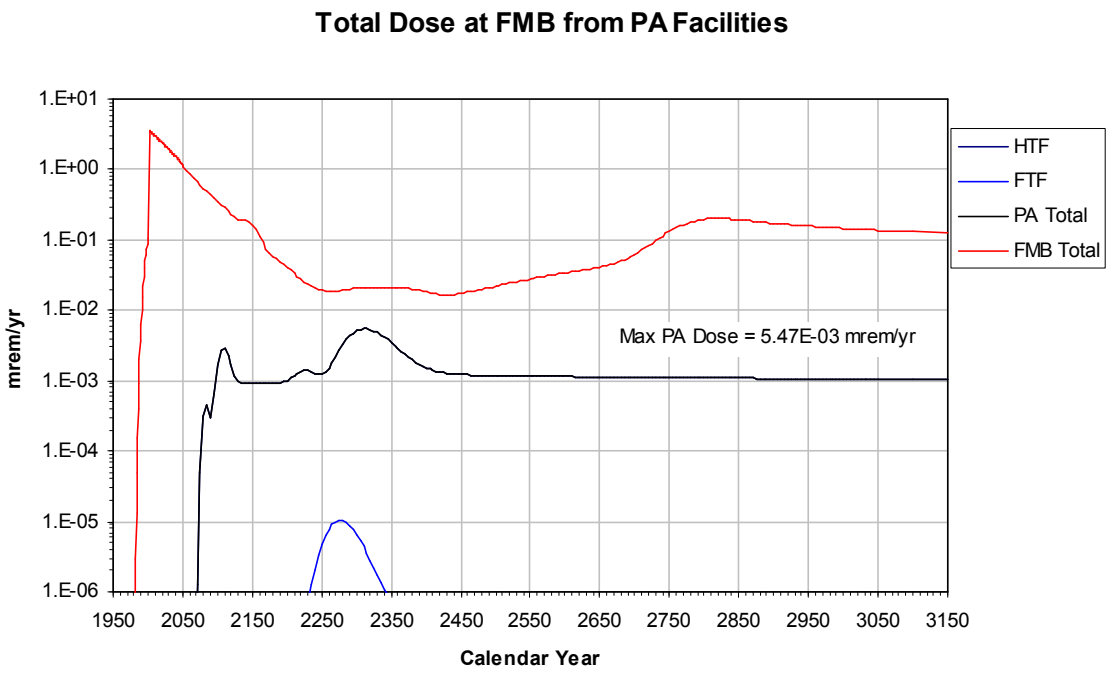


Figure 9-169. Contribution from PA Facilities to CA Dose at Fourmile Branch POA

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10.0 PERFORMANCE EVALUATION AND RESULTS INTEGRATION

In this section, the Savannah River Site (SRS) Composite Analysis (CA) doses presented in Section 9.0 are compared with performance objectives. Implications of the CA results for SRS land-use planning are also discussed. Performance objectives in DOE M 435.1-1 are stated in terms of a radiological dose limit to a hypothetical member of the general public. This person is assumed to reside and recreate at publicly accessible locations. The specific performance objectives are

- The Department of Energy (DOE) primary dose limit of 100 mrem/year
- The DOE dose constraint of 30 mrem/year, which is essentially a threshold dose above which it is necessary to conduct an options analysis to consider potential means to reduce the dose

In summary, the analysis results provide great confidence that neither the primary dose limit nor the dose constraint will be exceeded.

10.1 DETERMINISTIC BASE CASE

The results of the deterministic base case, presented in detail in Section 9.3.8 and Section 9.3.9 are summarized in Table 10-1 and in Figure 10-1 and Figure 10-2.

Table 10-1. Maximum Cumulative Doses at each Point of Assessment (POA) during CA Period of Assessment

Point of Assessment	Maximum Cumulative [1] Dose mrem/yr 2025 to 3025	Maximum Cumulative [1, 5] Dose mrem/yr 3025 to 12050	Major Contributing Source [2]	Major Contributing Radionuclide	Major Exposure Scenario/ Pathway
Upper Three Runs	1.06	0.40	H-Canyon	Np237	Recreational/ Fish Ingestion
Fourmile Branch	2.16	0.14	FMB IOU [3]	Cs137	Recreational/ Fish Ingestion
Steel Creek/Pen Branch	0.42	0.05	SC IOU	Cs137	Recreational/ Fish Ingestion
Lower Three Runs	2.97	0.05	LTR IOU	Cs137	Recreational/ Fish Ingestion
Savannah River	0.17 [4]	0.05 [4]	LTR IOU	Cs137	Residential/ Vegetable Ingestion

1 Sum of doses from the residential and recreational exposure scenarios, using the respective stream flow rate for recreational dose and the Augusta, GA, river flow rate, unless otherwise noted, for residential dose.
2 See Table C-1 for Source Identification corresponding to abbreviations given below.
3 IOU stands for Integrator Operable Unit, which are the stream and river beds.
4 Both residential and recreational doses are cumulative from all sources; the U.S. Highway 301 bridge flow was used.
5 In all cases, the maximum dose in the 9,000 years beyond the 1,000 year assessment period occurred in year 3025.

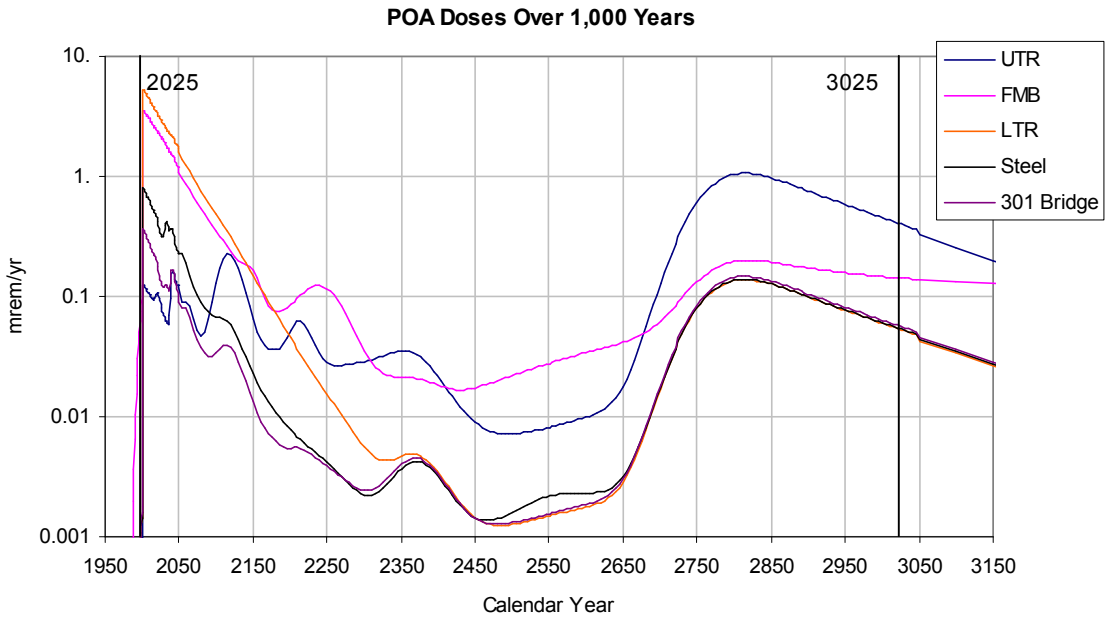

Figure 10-1. Total Cumulative Dose at POAs for 1,000 Year Assessment Period

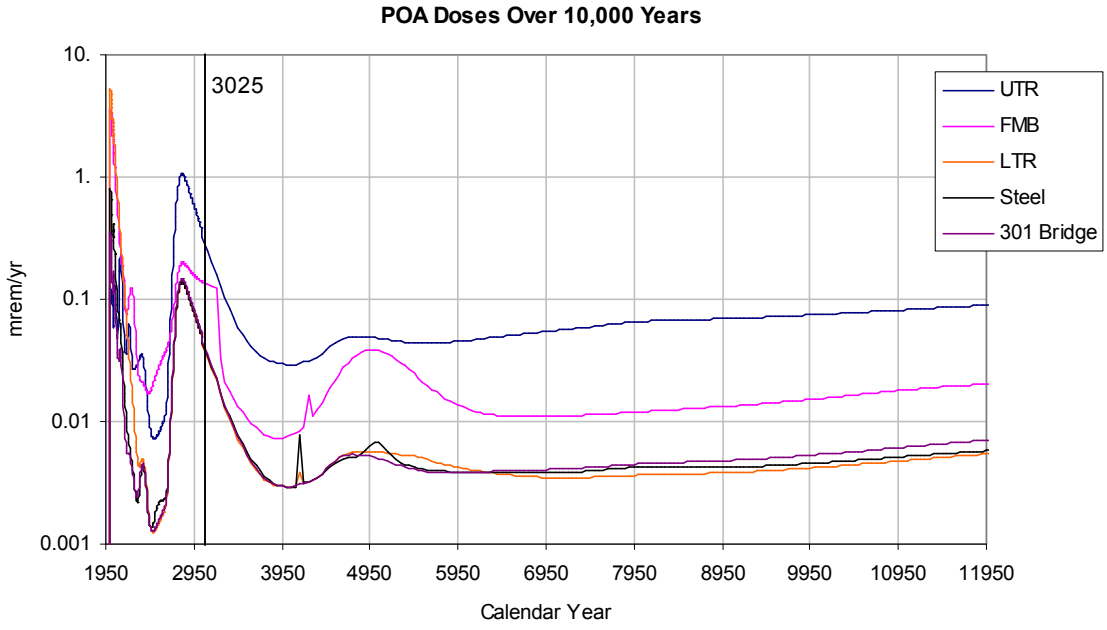

Figure 10-2. Total Cumulative Dose at POAs for Extended 10,000 Year Analysis Period

Table 10-1 shows the maximum dose calculated at each POA during the 1,000-year CA period of assessment starting at the projected site end-state date of 2025 and for an extended 10,000-year analysis period. Base case doses are all well below the primary dose limit of 100 mrem/yr, and the dose constraint of 30 mrem/yr (i.e., at least a factor of ten less than the dose constraint). Extending the assessment period to 10,000 years does not increase the maximum dose. The doses in Table 10-1 are cumulative doses at each POA. That is, the dose includes the contribution from direct discharges at the POA and from upstream sources.

For the stream POAs, the recreational dose ranges from 87% of the total dose for Upper Three Runs to 99.6% of the dose for Fourmile Branch. The recreational dose is dominated by fish consumption while the residential dose is dominated by consumption of vegetables. For all of the stream mouth POAs, recreational use based on stream concentrations dominated the calculated dose. Residential use of river water was less of a factor in the total dose at these POAs because of the large Savannah River flow. The largest contributor to the dose at Upper Three Runs was from Np237 released from H-Canyon. The largest contributor to the doses at the other stream locations was Cs137 released from stream sediment.

For the Savannah River POA both the recreational and residential doses were calculated using all sources with the annual average flow rate of the Savannah River at the U.S. Highway 301 Bridge. The residential dose represents 68% of the total maximum cumulative dose of 0.17 mrem/year (Table 10-1). This dose is dominated by Cs137 from the Lower Three Runs IOU.

Figure 10-1 and Figure 10-2 show the total cumulative dose at each POA as a function of time. The maximum dose from all the POAs, except for Upper Three Runs occurs at the beginning of the period of assessment and decreases due to the decay of Cs137. The maximum dose for the Upper Three Runs POA occurs at about year 2800. Figure 10-2 shows that the maximum dose, in all cases, is achieved within the 1,000-year CA period of assessment.

10.2 SENSITIVITY AND UNCERTAINTY RESULTS

Both deterministic sensitivity and stochastic uncertainty analyses were performed (Smith et al 2009). Results are summarized in this section.

10.2.1 Sensitivity Analyses

Several types of sensitivity cases were run to assess parameters that could have a significant impact upon the dose. These sensitivity cases were chosen to address the requirements in the Low-Level Waste Disposal Facility Federal Review Group (LFRG) Manual (DOE 2008) to consider factors such as radionuclide inventories, alternative points of assessment, groundwater divides, stream flow variation, and alternative disposal actions in sensitivity and uncertainty analysis (Phifer et al. 2008a). The results of the sensitivity analyses are summarized below.

10.2.1.1 Sensitivity to Period of Assessment

The deterministic calculations were carried out beyond the 1,000-year period of assessment. All of the sources were calculated for 10,100 years and the sources whose dose exceeded 0.001 mrem/year during the 10,100 year period were run further to year 102,050 (i.e., a 100,100-year assessment time), except for those sources that had clearly peaked during the 1,000-year CA assessment time. For twelve of the sources, the maximum dose between 1,000 and 10,000 years exceeded the maximum during the 1,000-year CA assessment period. For ten of the sources, the maximum dose between 10,000 and 100,100 years exceeded the maximum dose during the 1,000-year CA period of assessment. The dose from six sources was still increasing after 100,100 years. However, the magnitude of the doses associated with these six sources is small compared to the doses during the 1,000-year CA assessment period. The maximum dose observed after the 1,000-year CA assessment period is 1.04E+00 mrem/year from H-Canyon; this dose is the same as the maximum during the CA period of assessment. Therefore, even if the CA assessment period was extended to 10,000 years or longer, the maximum dose would not be expected to increase.

10.2.1.2 Sensitivity to Source Inventory

Sensitivity to source inventory was investigated by determining the factor by which each individual source in each POA would have to be multiplied for the cumulative POA dose to reach 30 mrem/year during the CA assessment period. Of the 152 sources assessed, 135 had inventory multipliers greater than 1,000, fourteen had inventory multipliers between 100 and 1,000, and only three had multipliers less than 100 (i.e., H-Canyon and the Fourmile Branch and Lower Three Runs IOUs). The overall inventory multiplier required for the combined sources to cause each POA to exceed 30 mrem/year was determined to range from 950 for the Savannah River POA to 10 for the Lower Three Runs POA. Other inventory sensitivity cases were run; the results are insignificant except for the H-Canyon/HB-Line, which shows that the extent of radionuclide removal during deactivation and decommissioning (D&D) of this facility is significant from a CA perspective. The CA base case assumed that only 1% of the Safety Analysis Report (SAR) inventory would remain; this resulted in a dose of 1.04 mrem/year. If 10% of the SAR inventory were to remain, the dose would be 10.4 mrem/year, which is a significant fraction of the CA dose constraint of 30 mrem/year. These results show that source inventory would have to be at least an order of magnitude greater than the best estimate inventory used in the base case for doses to approach the 30 mrem/year dose constraint.

10.2.1.3 Sensitivity to Location of the Point of Assessment

Although the SRS land use planning envisions no unrestricted use of any of the current SRS property, a sensitivity case was run assuming that the extent of controlled land decreased to encompass only the central, industrial core of the site. This would put the point of assessment at the boundary of the site industrial zone for both the residential and recreational exposure scenarios. At this boundary, the maximum total dose at the Upper Three Runs and Fourmile Branch POAs was 9.7 mrem/year for the Fourmile Branch POA during the CA assessment period. Doses at these POAs were also calculated for a 10,000-year assessment period. In no case did the maximum total dose reach the 30 mrem/year dose constraint. These results show that even if the extent of controlled land at SRS was greatly reduced and the assessment period was increased to 10,000 years, the CA dose constraint would not be exceeded due to radionuclide migration to the surface streams (i.e., impacts from groundwater use or intrusion into waste sites in the land outside the industrial zone were not assessed).

10.2.1.4 Sensitivity to Stream and River Flow

Sensitivity to stream and river flow rates was assessed in multiple ways. One way that it was assessed was to determine how low the annual average flow in each of the SRS streams would have to be to cause the recreational dose at the stream mouth POA to reach 30 mrem/year. This is a pertinent statistic, because the base case dose at all of the SRS stream mouth POAs is dominated by the recreational dose. The results were 3.1%, 7.1%, 1.3%, and 9.7% of annual average flow for Upper Three Runs (UTR), Fourmile Branch (FMB), Pen Branch/Steel Creek (PB/SC), and Lower Three Runs (LTR), respectively. The annual average flow would have to be reduced by at least a factor of ten before the recreational dose in these SRS streams would exceed 30 mrem/year.

Another way that sensitivity to flow rates was assessed was to determine how low the flow in the Savannah River at the U.S. Highway 301 Bridge would have to be to cause the combined recreational and residential dose to reach 30 mrem/year. The Savannah River flow rate would have to be reduced to 56.8 cfs before the combined recreational and residential dose would reach 30 mrem/year. The Savannah River annual average flow at the 301 Bridge is 10,175 cfs. The annual average flow in the Savannah River at the 301 Bridge would have to be reduced by approximately 180 times before the combined recreational and residential dose would reach 30 mrem/year.

These results show that no credible reduction in flow will cause the CA dose to approach the 30 mrem/year dose constraint.

10.2.1.5 Sensitivity to Groundwater Divide

Most of the high-impact radionuclide sources that are expected to remain at the SRS end state are located in the General Separations Area (GSA), close to the center of the site. The GSA is situated on a relatively narrow strip of upland located between two site streams, Upper Three Runs and Fourmile Branch. A local groundwater divide runs parallel to these streams and is approximately half way between them. In the CA Base Case model, the best estimate position of the divide was used to determine which stream a source would eventually release its radionuclide inventory into or how much of its inventory would be discharged in one direction versus the other. Because the estimated position of the groundwater divide has shifted slightly from year to year, a sensitivity scenario was devised to evaluate the worst outcome of such a change in conditions. The groundwater divide sensitivity analysis consisted of taking the Low-Level Radioactive Waste Disposal Facility (LLRDWF)/Mixed Waste Management Facility (MWMF), F-Tank Farm, and H-Tank Farm, all of which straddle the GSA groundwater divide, and evaluating the dose increase that results from assuming that the groundwater divide changes from base case conditions, such that the entire facilities discharge to Fourmile Branch, which has a significantly lower flow than Upper Three Runs. Although the dose increases by up to a factor of 42 when discharge to Fourmile Branch instead of Upper Three Runs is assumed, all the doses remained relatively low (i.e., the maximum dose is 0.7 mrem/year for the LLRWDF/MWMF source to Fourmile Branch). It is also unlikely that such a drastic change in the groundwater divide location would occur.

10.2.1.6 Sensitivity to End-State Date

The SRS end-state date is conservatively assumed to occur in 2025 in this CA; however, it is likely that the SRS end-state date will be later. Therefore a sensitivity case was conducted assuming that the SRS end-state date is 2050 rather than 2025. However, no attempt was made to revise estimated closure inventories with the extended period of operation. The total maximum doses associated with each POA over the 1,000-year assessment period either decrease or remain the same as the end-state date moves out from 2025 to 2050. The primary cause of the decrease in dose is a reduction in the dose from Cs137 released from streambed sediments. The peak dose from Cs137 occurs at the start of the CA assessment period and declines thereafter because of the rapid decay of Cs137. Therefore increasing the end-state date will not adversely impact CA results.

10.2.1.7 Sensitivity to Aquifer Clay

The 28 CA sources with any clay in their aquifer path that had a maximum dose greater than 0.001 mrem/yr, during the CA assessment period, were run replacing the clay with saturated sandy soil. The only POA for which the dose increased significantly was Upper Three Runs, where the maximum dose increased from 1.06 to 2.92 mrem/year due to sharpening of the Np237 peak concentration in the absence of clay. This study shows the importance of clay in the aquifer flow path, but even if there were no clay present in any of the flow paths, the CA dose would not approach the 30 mrem/year dose constraint.

10.2.1.8 Carbon-14 Bioaccumulation Factor

The C14 bioaccumulation factor was varied from the base case value of 3 L/kg to 4,500 L/kg and 50,000 L/kg. Results of the sensitivity analysis showed that maximum doses during the 1000-year CA assessment period did not change when the higher C14 bioaccumulation factors were used. The largest C14 inventory (4,000 Ci) was in the Old Radioactive Waste Burial Ground (ORWBG). Assuming a bioaccumulation factor of 50,000 L/kg, the maximum dose from C14 at the ORWBG is about 45 mrem/yr at the Fourmile Branch POA in the year 4500. However, such a large dose can only be produced by using a non-site specific C14 bioaccumulation factor that does not take into account the ecosystem carbon dynamics that are dominated by C12 rather than C14.

10.2.2 Uncertainty Analysis

Probabilistic uncertainty calculations were also performed for each POA with the sources that contributed greater than 0.05 mrem/yr to the total maximum dose. A total of 75 parameters were assigned distributions which were sampled using the Monte Carlo technique to provide input values for 500 simulations. Thirty-four of the parameters were related to the two exposure scenarios used in the dose calculations, (e.g., consumption rates of various foodstuffs and times spent performing various activities), twenty-six of the parameters were physical properties of the materials needed to conduct transport calculations (e.g., porosity and density), twelve parameters defined the flow rates in aquifers and surface streams in the model, and three parameters were used to define the changing properties of concrete barriers through time.

The results of the 500 runs at each POA were analyzed statistically. At the 95th percentile, the maximum doses are 11.5 mrem/yr at Lower Three Runs and 8.2 mrem/yr at Fourmile Branch. The other POAs had substantially smaller 95th percentile doses (i.e., UTR=2.4 mrem/year, SC/PB=1.8 mrem/year, and Savannah River=0.10 mrem/year[1]). The probabilistic calculations provide a high level of confidence that the 30 mrem/yr dose constraint will not be exceeded.

[1] Because the stochastic analyses only calculated doses for each POA and its sources, the 95th percentile dose for the Savannah River POA is smaller than the cumulative base case results.

10.3 OPTIONS ANALYSIS

Consistent with international and national recommendations, the DOE radiation protection system encompasses two principal elements— dose limits and optimization:

- Dose limits constitute allowable or tolerable doses that are not to be exceeded under normal conditions. The 100 mrem/year all-pathways dose is the primary dose limit for protection of the public from all sources and pathways. DOE also employs dose constraints in the implementation of the radiation protection system. The dose constraint of 30 mrem/year, which is set at a fraction of the primary dose limit, is established to ensure that no single source, practice, or pathway uses an extraordinary portion of the primary dose limit.
- Optimization effectively reduces public doses to levels as far below dose limits or constraints as is practicable, giving due consideration to collective impacts, costs, and other factors, using the as low as reasonably achievable (ALARA) process.

The CA process incorporates the elements of the radiation protection system as benchmarks to aid environmental management. The CA uses long-term projections of potential doses to support systematic environmental management of waste management and restoration sites. Two decision criteria, based on whether results exceed the dose constraint or the primary dose limit, are used in considering the implications of the CA results:

- The first decision criterion evaluates if the potential exists for the total dose from the CA to be greater than the dose limit of 100 mrem/year. If so, the potential future problem must be corrected or mitigated before it occurs. In this case, an options analysis must be conducted to identify alternatives for reducing future doses (before they occur) to tolerable levels.
- If no potential exists for the total dose from the CA to be greater than the dose limit of 100 mrem/year, then the CA results must be reviewed to determine if the potential exists for exceeding the DOE dose constraint of 30 mrem/year. If so, the options analysis must be conducted and the alternatives considered for determining what actions are reasonable to reduce potential future public doses.

The difference between exceeding the 100-mrem/year dose criterion and the 30-mrem/year options analysis criterion is that, in the first case, mitigating measures must be taken before the dose limit is exceeded, while in the second case, an action could be taken but may be determined to not be warranted.

DOE guidance for the Composite Analysis (DOE 1999) requires that impacts of commercial nuclear operations be considered, if warranted. Consistent with requirements in DOE Order 5400.5, doses from non-DOE activities need be considered only when (1) the dose to individual members of the public from DOE activities exceeds 30 mrem in a year and (2) the dose from the non-DOE activities also exceeds 30 mrem in a year to the same individuals. As indicated in Section 3.2.3.2, there are two non-SRS nuclear facilities within 50 miles of the SRS. These are the Chem-Nuclear Systems, Inc. Low-Level Radioactive Waste Disposal Facility in Barnwell, SC, and the Vogtle Electric Generating Plant, a nuclear power facility located across the Savannah River from the SRS. Radionuclide releases from these facilities or residual radioactive material left at these facilities when operations cease could impact members of the public that are potentially impacted by SRS. However, since the maximum dose from the CA is about three mrem/year, which is less than the dose constraint of 30 mrem/year, these non-SRS facilities need not be considered.

10.4 REFERENCES

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11.0 FUTURE WORK

Because Performance Assessment (PA) and Composite Analysis (CA) results are in part based on technically uncertain data, conservative parameters, or both, maintenance programs have been instituted consistent with the "Maintenance Guide for DOE Low-Level Waste Disposal Facility PA and CAs" (November 10, 1999) as reflected in DOE Order 435.1. The purposes of maintenance programs are to confirm the continued adequacy of the PAs and CAs and to increase confidence in their results. As such the maintenance programs have been designed to reduce uncertainty in the inputs and assumptions providing greater confidence in the results of the analyses and in the long-term plans for public and environmental protection. Additionally, a disciplined process to address potential changes in disposal operations (e.g., new waste forms, change in unit design) is needed to ensure that proposed changes do not adversely affect disposal facility performance (SRNS 2009).

At the Savannah River Site (SRS) the following two PA/CA maintenance programs are in effect (SRNS 2009):

- Savannah River Nuclear Solutions (SRNS) manages the PA/CA maintenance program associated with the PA for the E-Area Low-Level Waste Facility (ELLWF), managed by Waste Management Area Projects (WMAP) and with the SRS CA.
- Savannah River Remediation (SRR) manages the PA maintenance program associated with the PAs for the Saltstone Disposal Facility (SDF), F Tank Farm (FTF), and, when the PA is completed, H Tank Farm (HTF) managed by the Liquid Waste Organization (LWO)

SRNS and SRR have completed a consistency review between the two maintenance programs under the auspices of the DOE Order 435.1 Working Group. This review has ensured that the scope in both programs is consistent, avoids duplication of effort, and benefits from sharing of data. Plans for both maintenance programs have been prepared and they are updated each year and submitted to the Department of Energy (DOE) (SRNS 2009).

Based upon this maintenance program division, identified maintenance items necessary to support the CA are managed within the SRNS PA/CA maintenance program. The 2009 SRNS PA/CA maintenance plan is entitled, "E-Area Low-Level Waste Facility Performance Assessment and Savannah River Site Composite Analysis Maintenance Program, FY2009 Implementation Plan" (SRNS 2009). Section 11.1 below provides a listing of the maintenance items directly associated with the CA that are addressed within the 2009 SRNS PA/CA maintenance plan (SRNS 2009).

During development and review of this SRS CA revision, additional future work items were identified and categorized as either priority or potential future work items consistent with Carilli and Golian 2010. Section 11.2 provides separate listings for each category of future work items.

11.1 SRNS MAINTENANCE PLAN CA MAINTENANCE ITEMS

This section provides a listing of the maintenance items directly associated with the CA that are addressed within the current SRNS PA/CA maintenance plan (SRNS 2009). Two lists are provided. The first list provides maintenance items directly associated with the CA, which are currently scheduled to be performed. The second list provides maintenance items directly associated with the CA, which are dependent upon the availability of American Recovery & Reinvestment Act funding for implementation.

The following is the listing of CA related maintenance items, which are currently scheduled to be performed. The section number and title from the 2009 SRNS PA/CA maintenance plan (SRNS 2009) are provided along with a brief description for each item.

Section 1.1.10 Evaluate Onsite disposal of Pu238 Transuranic Waste per 40 CFR 191 and DOE 435.1 Requirements
This task is to evaluate near-surface disposal of high-risk (from worker repackaging perspective) Pu238 transuranic (TRU) waste in the vicinity of TRU Pad #1. This disposal will be assessed per 40 CFR 191 performance requirements to determine whether this material needs the degree of isolation afforded by a geologic repository. This disposal will also be evaluated against DOE 435.1 performance requirements for low-level waste.

Section 1.1.13 Improvements to ELLWF GoldSim™ Trench Models and Development of Vault Models
Initial GoldSim™ models of the Slit and Engineered Trenches, Component in Grout (CIG) Trenches, and Naval Reactor Component Disposal Areas (NRCDAs) were developed as part of the recently approved ELLWF PA revision. Enhancements to these models will be made to increase confidence in the robustness of the radionuclide disposal limits. In addition to this work, Savannah River National Laboratory (SRNL) will undertake the initial development of GoldSim™ models representing the Low Activity Waste Vault (LAWV) and Intermediate Level Vault (ILV). The final step will be to incorporate all components into an integrated ELLWF model.

Section 1.2.9 Maintain the SRS CA Radionuclide Inventory Database
This task involves annually updating the SRS CA Radionuclide Inventory database and report with new information pertaining to SRS facility decommissioning and deactivation (D&D), waste site remediation or closure (pursuant to the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) or the Resource Conservation and Recovery Act (RCRA)), and tank closure, as well as changes in site use planning or acceptance of new missions. Annual changes to this baseline database and report will be reviewed as appropriate through the Unreviewed Disposal Question (UDQ) Evaluation process to determine if such changes impact the conclusions of the SRS CA. Accomplishing this annual review will require the involvement of multiple site organizations (e.g., SRNL, Regulatory Integration and Environmental Services (RI&ES), LWO, Soil and Groundwater Closure Projects (SGCP), Site Decommissioning and Deactivation (SDD), etc.). This dataset will be placed under quality assurance (QA) and configuration control.

Section 1.2.11 Evaluate Implications of ELLWF and SDF SA's on CA Assumptions and Results

Special analyses will be required based on the following:

- Any changes in land use plans or remediation or closure plans of any of the facilities considered in the CA
- Upgrading the existing CA analyses (e.g., incorporate research and development (R&D) results)
- Evaluating the impact of PA special analyses or PA revisions on the results of the CA

Therefore, as part of each of the PA/CA Annual Reviews, an evaluation will be included to assess the impact of any Special Analyses (SAs) performed on the CA.

Section 3.2 CA Annual Review and Maintenance Plan Update

CA reviews have been conducted annually since FY2000. The CA review approach will be conducted in a systematic manner that incorporates all of the following considerations:

- Site Future Land Use Plans – These plans will be reviewed in regards to the impact of any changes on the CA results and conclusions.
- PA analyses – All PA analyses will be reviewed and the impact on the CA results will be addressed.
- Changes in remediation or closure plans – The review will consider changes in remediation or closure plans for any of the facilities considered in the CA and the impact that those changes will have on the CA results.
- Changes in inventory estimates – The review will consider changes in the inventory estimates (residual radioactive material) considered in the CA to determine the impact.
- Results of monitoring and R&D.

All of these factors will be reviewed annually to evaluate the need to conduct special studies or to prepare a revision of the CA. A report will be generated each year documenting the results of the study. The report will be submitted as part of the ELLWF PA Annual Review and submitted to the Department of Energy-Savannah River (DOE-SR) for review. The CA Annual Summary will be prepared by DOE-SR and submitted to the Department of Energy-Headquarters (DOE-HQ) Low-Level Waste Disposal Facility Federal Review Group (LFRG) as part of the ELLWF PA Annual Summary. The CA section of the Maintenance Plan is updated annually. This update involves removal of completed scope, identification of new scope, adjusting the funding cycle on work tasks based on current WMAP priorities, and updating current year and outyear budget estimates.

Section 5.1.1 Revise Material Properties Data Package

Conceptual models of the disposal system and the latest information on hydraulic properties for the various materials comprising the disposal system that were used in the ELLWF and SDF PA revisions were documented in a material properties data report (WSRC-STI-2006-00198). This "snapshot" of models and data will need to be updated as new information becomes available to support future PA and CA work. For planning purposes, this baseline reference will be scheduled to be revised every three years with the latest material properties data and conceptual models.

Section 5.1.2 Revise Geochemical Data Package

Conceptual models of the disposal system and the latest information on geochemical properties for the various materials comprising the disposal system that were used in the ELLWF and SDF PA revisions were documented in a geochemical data report (WSRC-TR-2006-00004). This "snapshot" of models and data will need to be updated as new information becomes available to support future PA and CA work. For planning purposes this baseline reference will be scheduled to be revised every three years with the latest geochemical data and conceptual models.

Section 5.1.3 Evaluate Available Groundwater Codes for Potential Use in Future PA's and CA's

The capabilities and limitations of PORFLOW were exposed in the process of developing the ELLWF and FTF PA's over the last two years. These problems included; difficulties converging on flow field solutions under certain assumptions for unsaturated zone hydraulic properties, and issues with calculating solubility-controlled limits. These concerns were exacerbated by the lack of access to the source code. The cost to maintain a proprietary code like PORFLOW includes the initial licensing and annual maintenance agreements. SRNL will pursue the following program:

- Conduct a relatively quick paper study to survey and screen available open/closed source and proprietary/free groundwater codes that we should consider for use in future PA efforts. This process is anticipated to result in no more than two or three codes for further evaluation.
- Perform a head-to-head comparison of the remaining codes leading to selection of at least one as a second PA level code.

- Upgrade the selected code to perform PA-critical functions, solve problems representative of SRS groundwater flow and transport conditions, and add unique features/commands found in PORFLOW.
- Develop software QA Plan and document Testing & Verification.

Section 5.1.9 Clarify Understanding of Sorption Behavior for Key PA Radionuclides
SRNL will perform laboratory experiments to develop additional site-specific distribution (or sorption or partition) coefficient (K_d) values. SRNL will evaluate Np237 in both cementitious and natural subsurface environments. Neptunium exists in two oxidation states (IV and V). The ELLWF PA assumes that it is in the more mobile of the two forms. SRNL will develop an analytical technique to separate the two forms and observe the change in oxidation as a function of time under cementitious, reducing grout and natural soil conditions.

Section 5.2.1 Investigate the Geochemistry and Environmental Fate of C14 in the SRS Environment
C14 loadings into fish are one of the key risk drivers for the CA. Such geochemical processes as volatilization, C14 precipitation near concrete, pH effects on aqueous C14 chemistry, and isotopic dilution are processes that are not presently accounted for in the CA that may influence C14 migration. By quantifying these processes under site-specific conditions, it will be possible to include less conservative estimates of C14 attenuation in the CA. SRNL will complete a literature review of C14 transport modeling and geochemistry and lab testing of volatilization versus pH and organic matter/soil microbe conditions and C14 sorption under various pH environments.

Section 5.2.2 Investigate Dose Consequences in Humans of C14 from Eating Fish.
The C14 dose to humans resulting from fish consumption is computed on the basis of a generic uptake or bioaccumulation factor relating the C14 concentration in fish to the C14 concentration in water. However, bioaccumulation factors are often strongly affected by site-specific conditions. Important parameters must be validated and adjusted to ensure accurate results. SRNL will undertake a literature search and summary to investigate C14 bioaccumulation in fish as a basis for determining the accuracy of currently used factors and adjusting them for more realistic site specific results. The deliverable will be a report summarizing existing information on C14 bioaccumulation in fish and presenting recommendations and justification for changes in computation methods or inputs. This task was completed with issuance of the “Systems Model of Carbon Dynamics in the Fourmile Branch on the Savannah River Site“ (Hinton et al. 2009). However, follow-on work is recommended as outlined in Table 11-1 and item 2 of Table 11-3, Potential Future Work Items.

Section 5.2.3 Investigate Phenomenon of I129 Concentrating at the Seepline
I129 may be naturally attenuated in the SRS by processes that are not accounted for in the present CA. These attenuating processes occur in organic rich environments, such as seeplines/wetlands; they include 1) greater sorption of I129 (greater K_d values), and 2) greater volatilization. Recent field data suggests strong attenuation of I129 is occurring in F-Area. I129 was believed to be placed in the F-Area seepage basins between 1955 and 1988 at concentrations of <100 pCi/L. I129 groundwater concentrations 1-km down gradient in a seepline are as high as 1600 pCi/L, indicating that some biogeochemical process has been concentrating the I129 in the wetland. SRNL will collect field samples from F-Area along the seepline where I129 concentrations are high and identify through laboratory testing the responsible attenuating process (e.g., partition to soil-bound organic matter, sorption to unique wetland mineral assemblages). SRNL will also conduct laboratory tests using E-Area soil & groundwater to determine the range of conditions under which these attenuation processes occur. This is important because it is necessary to define the range of environmental conditions under which I129 attenuation occurs. Work will also include quantification of the important iodine attenuation processes identified in a manner that can be incorporated into the CA. SRNL will evaluate how robust the geochemical model is. In a parallel activity, SRNL will be receiving funding for three years from DOE-Office of Science to study iodine geochemistry in an adjoining aquifer, F-Area. This is a basic research study, meaning it is designed to provide basic fundamental principles which can then later be used by engineers or more applied researchers to provide data of direct relevance for solving a problem. For instance, no K_ds will be measured in the Office of Science study, but this information will be used to provide guidance for setting up a K_d experiment, the subject of this CA funded project. It is anticipated that the two projects will synergize well with each other.

Table 11-1 provides a listing of maintenance items directly associated with the CA, which are dependent upon the availability of American Recovery & Reinvestment Act funding for implementation. This list was extracted from Appendix C of the 2009 SRNS PA/CA maintenance plan (SRNS 2009)

Table 11-1. Potential ARRA Funded CA Maintenance Items

Task	Notes
Extend/expand/promote CA to support SRS end-state decisions	Modify CA models to obtain groundwater concentrations. Expand to nonrads. Set up and test models. Excludes running models.
Produce new General Separation Area (GSA) model	Synthesize new hydrologic data since 1995. Select groundwater code. Calibrate using optimizing software.
Develop an SRS regional groundwater model encompassing the entire SRS	An SRS regional groundwater model, encompassing the entire SRS, should be developed from which boundary controls for smaller SRS groundwater models with greater grid resolution can be extracted and the impacts of transient drought and wet conditions on contaminant transport can be evaluated. This effort at a minimum should include: identification of model boundaries and boundary conditions; synthesis of existing hydrogeologic data; database development; identification of data gaps; collection of field data; archiving field samples and data as appropriate; development of appropriate recharge (infiltration) / drain top surface boundaries based upon surface conditions; development of low, average, and high potentiometric surfaces for the water table and underlying aquifers; model development; and development of advanced visualization tools.
Develop GoldSim™ models of ILV and LAWV and build integrated ELLWF model	Includes improvements in trench models
Sample analysis for validation of C14 bioconcentration model	Supports CA revision dose modeling
Dose studies	Identify important nuclides in PA/CA. Assess existing dose parameters. Identify opportunities for reducing conservatism. One possibility is considering specific activity of I129 and iodine metabolic model.

11.2 FUTURE WORK ITEMS

During development and review of this SRS CA revision, additional future work items were identified and categorized as either priority or potential future work items. Table 11-2 provides a listing of the priority future work items per the following categories:

- SRS CA Inventory Database Maintenance,
- SRS CA Flow and Transport Characterization and Modeling, and
- SRS CA Sensitivity and Uncertainty Analysis.

The future work items listed in Table 11-2 were identified as priority future work items by the Low-Level Waste Disposal Facility Federal Review Group (LFRG) SRS CA Review Team (Carilli and Golian 2010). Three of these items have already been specially added to the 2010 SRNS PA/CA maintenance plan (SRNS 2010) as denoted within Table 11-2. The remaining Table 11-2 items will be incorporated into the future 2011 SRNS PA/CA maintenance plan as outlined within the 2010 SRNS PA/CA maintenance plan (SRNS 2010) Section 3.2.2, Address Issues Assigned to Maintenance from the LFRG Review of the SRS CA.

Table 11-3 provides a listing of the potential future work items per the following categories:

- SRS CA Base Case Input Data Improvements,
- SRS CA Uncertainty Analysis Input Data Improvements,
- SRS CA Model Improvements, and
- SRS CA Monitoring Plan Update

The Table 11-3 items will be evaluated during production of the 2011 SRNS PA/CA maintenance plan and added to the plan as warranted.

Table 11-2. Priority Future Work Items

Priority Future Work Category: SRS CA Inventory Database Maintenance	
Item	**Description**
1	A re-evaluation of SRS facility and waste site lists will be conducted to identify any facilities or waste sites, which were overlooked in the 2009 CA inventory (i.e., SRS facilities and waste sites anticipated to have an "End State" radionuclide inventory that were not included in the 2009 CA inventory). This item was specifically added to the 2010 SRNS PA/CA maintenance plan, and work on this item has been initiated.
2	The CA inventory report (Hiergesell 2008) should be revised with corrections made during CA development (e.g., see Tables A-17, A-46, A-61, A-73, A-74, A-75, and A-78 in Volume II) and other appropriate changes (Smith et al. 2009a).
3	FTF and SDF PAs are currently under review by the LFRG and/or Nuclear Regulatory Commission (NRC). Results of those reviews could impact the inventories or base case flux to the water table for both PAs. In fact, revision 1 of the FTF PA is under development to incorporate comments from the Nuclear Regulatory Commission (NRC), Environmental Protection Agency (EPA), and South Carolina Department of Health and Environmental Control (SCDHEC), and it will be issued in FY 2010. The CA should consider any such future changes to these PAs.
4	Methodologies to estimate the inventory uncertainty associated with significant radionuclide source locations should be developed and implemented. The effort should focus on the most significant sources first, with significance defined in terms of the maximum dose from Table 9-26 through Table 9-30. The initial effort will focus on the LTR IOU as outlined in Item 5 below. Work on other significant sources should follow, such as the FMB and SC/PB IOUs (Item 6 below) and the H-Canyon (Item 7 below). Additionally, defensible criteria to categorize whether sources require a distribution or not should be established.
5	As summarized in Section 10.0, Cs137 in the Lower Three Runs (LTR) Integrator Operable Unit (IOU) (i.e., streambed and floodplain) is the primary CA dose driver. Therefore the uncertainty associated with the LTR IOU inventory (i.e., inventory distribution) will be developed along with a re-evaluation of the base case inventory. While Cs137 is the most significant and abundant radionuclide associated with the LTR IOU, it is not the only radionuclide. Therefore streambed inventories and distributions for other radionuclides will also be developed. This effort will initially focus on existing sampling and analysis data. However this effort may require additional streambed and floodplain sampling and analysis that may include horizontal and vertical distributions of the radionuclides and correlation with water concentrations including Cs137. This item was specifically added to the 2010 SRNS PA/CA maintenance plan, and work on this item has been initiated. (Hiergesell et al. 2008 and Smith et al. 2009a)
6	As summarized in Section 10.0, Cs137, from the FMB and SC/PB IOUs (i.e., streambed and floodplain) are the primary dose driver for those respective Points of Assessment (POA). Therefore the uncertainty associated with the FMB and SC/PB IOU inventories (i.e., inventory distribution) should be developed along with a re-evaluation of the base case inventories. While Cs137 is the most significant and abundant radionuclide associated with the FMB and SC/PB IOUs, it is not the only radionuclide. Therefore streambed inventories and distributions for other significant radionuclides should also be developed. This effort should initially focus on existing sampling and analysis data. However this effort may require additional streambed and floodplain sampling and analysis that may include horizontal and vertical distributions of the radionuclides and correlation with water concentrations including Cs137. (Hiergesell et al. 2008 and Smith et al. 2009a)

Table 11-2. Priority Future Work Items -- continued

Priority Future Work Category: SRS CA Inventory Database Maintenance - continued	
Item	**Description**
7	As summarized in Section 10.0, H-Canyon and its associated Np237 inventory is the primary dose driver for the UTR POA. Therefore the uncertainty associated with the H-Canyon inventory (i.e., inventory distribution) should be developed along with a re-evaluation of the base case inventory. Additionally an investigation of H-Canyon Np237 should be conducted to determine how and in what form the Np237 is distributed, and whether or not the large end-state inventory calculated from the SAR information is credible.
8	The inventory and inventory distribution for radionuclides within the F and H-Area Seepage Basins groundwater plumes, such as Strontium 90, Technetium 99, Cesium 137, Iodine 129, should be developed in addition to that for tritium. This inventory and its distribution should be evaluated within the SRS CA through screening or an actual dose assessment, as appropriate.
9	A method has been developed to estimate the residual inventory for operational facilities and future facilities whose end states are slated to be in situ disposal (ISD) or demolish to slab. The method consists of using facilities for which safety documentation, both during operation and following deactivation, exists. The ratio of inventories provides an estimate of the factor by which the operational inventory might be reduced prior to reaching the End State. At this time the reduction factors are based upon two facilities, the F Canyon complex and the 321-M building. As more facilities are deinventoried and either closed by in situ disposal or demolish to slab, the inventory estimation ratio should be revised based on the new final data from those facilities. (Hiergesell et al. 2008)
10	The CA radionuclide screening (Taylor et al. 2008) should be revised by using the CA base case model (transport plus dose modules) and also by considering key radionuclides associated with the SDF PA. Key radionuclides from the ELLWF and FTF PAs were considered during the CA radionuclide screening, but the SDF PA had not been performed at the time of the screening
Priority Future Work Category: SRS CA Flow and Transport Characterization and Modeling	
Item	**Description**
11	A water balance study to provide estimates of natural stream flow for Upper Three Runs (UTR), Fourmile Branch (FMB), Steel Creek/Pen Branch (SC/PB), and Lower Three Runs (LTR) should be performed. Such a study could also potentially correlate real-time precipitation with stream flow variations and assist in better quantification of deep infiltration, runoff, evapotranspiration, and groundwater-surface water interactions. Years wherein reactor cooling water discharges, the largest anthropogenic contributor to on-site stream flow, occurred have not been included in the stream flow estimates used in the CA. However, other, much smaller and often intermittent industrial NPDES discharges to streams have not been subtracted from the USGS stream flow measurements used. The current SRS Land Use Plan states that the entire site will be owned, controlled, and maintained by the federal government, most likely by the DOE, in perpetuity, as established by Congress. Site boundaries will remain unchanged and the Site will be used for industrial purposes for future DOE and non-DOE missions. Based upon these SRS land-use plans, the current stream flow estimates provide a reasonable estimate into the future, but the determination of natural stream flows would provide a basis for a low-flow sensitivity other than the 7Q10 low-flow values used by the CA. (Jones 2009)
12	An SRS regional groundwater model, encompassing the entire SRS, should be developed as outlined in Table 11-1. This regional groundwater model should be used to establish boundary controls for smaller SRS groundwater models with greater grid resolution and to evaluate the impacts of transient drought and wet conditions on contaminant transport. As part of the evaluation of the impacts of transient drought and wet conditions, the model should include low, average, and high potentiometric surfaces of the water table and underlying aquifers, so that distributions about the aquifer flow path parameters can be developed (Hamm et al. 2009 and Smith et al. 2009b).

Table 11-2. Priority Future Work Items -- continued

Priority Future Work Category: SRS CA Flow and Transport Characterization and Modeling - continued	
Item	**Description**
13	The following field characterization associated with the UTR, FMB, SC/PB, and LTR streambeds should be performed: • Streambed vertical gradients, sediment types, and saturated hydraulic conductivities (groundwater-surface water interactions) • Streambed sediment scour, deposition, and transport • Streambed K_ds for the predominant radionuclides. This item along with Items 5, 6, 11, and 14 will help validate the CA streambed release modeling and further reduce the uncertainty associated with the release of radionuclides from streambed sediments. (Smith et al. 2009a)
14	The distribution of uranium within Tims Branch between that dissolved in the water, that bound to the streambed sediment, and that bound to particulates in transit should be investigated. The implications of this distribution on uranium mobilization and the rate of uranium transport to the CA point of assessment (mouth of UTR) should be determined. Finally the resulting dose implications of such uranium distributions, mobilization, and transport should be determined. Such an effort may require additional streambed sampling and analysis. (Smith et al. 2009a)
Priority Future Work Category: SRS CA Sensitivity and Uncertainty Analysis	
Item	**Description**
15	GoldSim™ has a distributed processing (DP) capability that can be used when performing probabilistic calculations. Using this feature, individual realizations can be run on as many processors as the master GoldSim™ simulation can be linked to. The basic GoldSim™ software is limited to using four processors one of which is reserved for the master simulation that farms out realizations to the connected processors. This capability was utilized in performing the CA uncertainty calculations which reduced the simulation run time by a factor of three. However, by adding the GoldSim™ DP module, available from GoldSim Technology Group at a nominal cost, a probabilistic GoldSim™ simulation can be connected to as many processors as are available. This offers the possibility of dramatically decreasing probabilistic simulation run times and increasing the capability of performing uncertainty calculations including more sources with more realizations. Utilizing this approach is currently limited by our inability to access other computers through the SRS network primarily from computer security concerns. If a large cluster of Windows machines could be assembled off the SRS network, all of the processors could be accessed by GoldSim™ for probabilistic calculations. This item was specifically added to the 2010 SRNS PA/CA maintenance plan, and work on this item has been initiated. (Smith 2009b)

Table 11-2. Priority Future Work Items -- continued

Priority Future Work Category: SRS CA Sensitivity and Uncertainty Analysis	
Item	**Description**
16	A systematic sensitivity analysis should be performed to identify the model parameters that have the greatest impact on CA results. This analysis should investigate the additive and multiplicative effects of parameters on the CA results. This analysis should investigate parameters in the transport model and in the dose model separately. This systematic sensitivity analysis along with a more structured uncertainty analysis (Item 17) will assist in future work prioritization. Expertise in the SRNL statistical group should be utilized to help structure this investigation and interpret the results.
17	A more structured uncertainty analysis should be performed to identify both those stochastic variables that have the greatest impact on model results and stochastic variables that have an insignificant impact on model results and can be eliminated from the uncertainty analysis. In particular inventory uncertainty distributions developed from Items 4, 5, 6, and 7 should be included in the uncertainty analysis. This structured uncertainty analysis along with a more systematic sensitivity analysis (Item 16) will assist in future work prioritization. Expertise in the SRNL statistical group should be utilized to help structure this investigation and interpret the results.

Note to Table 11-2: Items 5, 6, 11, 13, and 14, associated with the streambed radionuclide inventory and transport, should consider the physical and chemical characteristics of the radionuclides within the streambed, the streambed characteristics, and the effects of any Comprehensive Environmental Responses, Compensation, and Liability Act (CERCLA) actions prescribed in the Record of Decisions (RODs) or similar binding agreements for the contaminated streambeds. Additionally any extrapolation made from known data to estimate the streambed radionuclide inventory should be adequately justified.

Table 11-3. Potential Future Work Items

Potential Future Work Category: SRS CA Base Case Input Data Improvements	
Item	**Description**
1	Radionuclide mass fluxes into stream bed sediments should be better quantified as part of the Table 11-2 Item 11 water balance study and Table 11-2 Item 12 SRS regional groundwater model development. [1]
2	Comprehensive daily SRS stream flow measurements are not currently being gathered and archived. In the past, the United States Geologic Survey (USGS) operated a comprehensive stream gauging system at SRS. USGS continues to monitor several Savannah River stations, but stopped monitoring SRS stream stations in 2002. A comprehensive daily stream flow monitoring system should be reestablished at SRS. Such a comprehensive daily stream flow monitoring system should, at a minimum, include the following former USGS stations in order to support the CA (Jones 2009): • Station 02197315 • Station 02197310 • Station 02197342 • Station 02197344 • Station 02197348 • Station 021973565 • Station 02197400 • Station 02197415
3	The systems model of carbon dynamics in Fourmile Branch (Hinton et al. 2009), which resulted in the 3 L/Kg C14 water-to-fish bioconcentration factor, should be further validated by either attempting to use more sensitive analytical means to detect C14 or collecting larger masses of Fourmile Branch fish and stream water for subsequent C14 analyses.
4	Actual SRS site-specific AgI solubility data should be developed to replace the conservative AgI solubility estimate currently used in the CA. (Smith et al. 2009a)
5	The AgI solubility controlled release mechanism for I129 was not applied to the Mixed Waste Management Facility (MWMF) because an estimate for the split between I129 on berl saddles and generic I129 could not be found. Additional investigations into historic data should be conducted in order to determine the split between I129 on berl saddles and generic I129 in the MWMF, so that the AgI solubility controlled release mechanism can be utilized for the MWMF. (Smith et al. 2009a)
6	The F-Area and H-Area Seepage Basins were closed with a clay cap in 1990. However, the year 2002 was set as the Cap Placement Year within the CA Base Case model, and infiltration associated with a clay cap was implemented at that time for purposes of the CA calculation. Because of the short time lapse, the difference between the actual and model closure dates was deemed insignificant, but should be corrected in future calculations. (Smith et al. 2009a)

Table 11-3. Potential Future Work Items – continued

Potential Future Work Category: SRS CA Uncertainty Analysis Input Data Improvements	
Item	**Description**
7	The daily mean flow rate time series, utilized to produce the distributions of the means of stream flow rates, is relatively short particularly for the SC/PB combined flow and LTR. These distributions should be improved by one or both of the following methods (Shine 2009): • Obtain additional daily stream flow data as outlined in Item 1, and/or • Correlate stream flow with an extended time series for local rainfall or other relevant available data.
8	Consideration should be given to whether three-sigma, two-sigma, or some other bounds to parameter distributions are most appropriate. This consideration should strive to balance avoidance of spurious parameter tail value combinations, which can lead to physically implausible model output, with elimination of potential plausible combinations from consideration.
9	It should be determined whether or not the dose module stochastic variables, which were assigned triangular distributions with their deterministic (most likely) value set to the minimum or maximum extreme values of their distributions, should be revised so that the deterministic value is the mean of the distributions (Phifer et al. 2009 and Smith et al. 2009b).
Potential Future Work Category: SRS CA Model Improvements	
Item	**Description**
10	In order to cut run times, future updates of the SRS CA GoldSim™ model should consider updating to the latest GoldSim™ version so that the new pipe elements, which can handle distributed inputs, could be used rather than the huge number of cell elements used in the current SRS CA GoldSim™ model. [1]
11	Once the Table 11-2 Item 11, water balance study and Table 11-2 Item 12, SRS regional groundwater model development have been completed, this information and model should be used for future SRS Performance Assessment (PA) and future CA models to improve consistency between them. [1]
12	The three dimensional flow model(s) from which CA aquifer flow path parameters are derived should consider the impacts associated with the presence of closure caps over large portions of SRS. These models were previously calibrated to data where such extensive covering was not present. (Hamm et al. 2009)
13	The CA GoldSim™ model should be revised so that the barrier layer associated with use of source release models can be specified as concrete, clay, or sand. (Smith et al. 2009a)
14	After Item 15 in Table 11-2 is completed, a fully probabilistic analysis should be conducted over 10,000 years both at the base case POAs and at the Alternate Industrial POAs.
15	The dose module (Phifer et al. 2009) should be revised to include transfer factor distributions (soil-to-vegetable, feed-to-milk, feed-to-meat, and water-to-fish transfer factors) provided by Taylor et al. 2008.
16	A GIS system should be developed to increase the utility of the CA by more clearly displaying the CA results and related information using the CA model.
Potential Future Work Category: SRS CA Monitoring Plan Update	
Item	**Description**
17	The PA/CA Monitoring Plan should be revised to include monitoring specific to this revised CA in order to validate the CA results once the CA has been approved.

[1] These potential future items (i.e., 1, 10, and 11) were identified by the LFRG SRS CA Review Team as items that should be considered for addition to the SRNS PA/CA maintenance plan (Carilli and Golian 2010).

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13.0 PREPARERS

13.1 LIST OF PREPARERS

A list of the Composite Analysis preparers, along with a short biography for each and a statement of their contribution to the Composite Analysis, is provided below.

BACH, MORGANA, Savannah River National Laboratory

Ph.D. Environmental Engineering – University of Florida
B.S. Environmental Engineering – University of Florida

Experience: Dr. Bach has worked at the Savannah River National Laboratory for two years. Her research at the University of Florida was on adsorption systems with a specific focus on the surface chemistry of adsorbents. Her current work is on adsorption systems with an overall focus on water quality of natural waters and the water chemistry of pollutants.

Contribution: Dr. Bach performed design checks of work and reports.

BAKER, RACHEL A., Savannah River National Laboratory

M.S. Statistics – University of Florida
B.S. Statistics – University of Georgia

Experience: Ms. Baker is a Principal Scientist at the Savannah River National Laboratory with 19 years of experience at SRS. She is the developer of the Aqueous Waste Software Application (AWSA), used to determine containerization requirements for purge water generated during well sampling and to produce Special Reports of groundwater data and of well sampling activity. Ms. Baker also provides statistical support for the material control and accountability function at SRS.

Contribution: Ms. Baker conducted a design check of the stream flow distribution report.

BIRK, MARCIA, Savannah River Remediation

B.S. Chemical Engineering – University of Minnesota, Institute of Technology

Experience: Ms. Birk has over 19 years of experience at SRS in waste management engineering and environmental compliance. In her assignment in waste management engineering, she was responsible for radioactive and chemical characterization of mixed waste to meet offsite treatment facility waste acceptance criteria and Department of Transportation requirements for radioactive waste shipments. She has also held responsibility for development and review of the F-Tank Farm and Saltstone Disposal Facility performance assessments.

Contribution: Assisted in development of the inventory report to characterize SRS facilities.

BUTCHER, BYRON T., Savannah River National Laboratory

M.S. Environmental Engineering – University of Tennessee
B.S. Civil Engineering – University of Tennessee

Experience: Mr. Butcher is currently program manager of the E-Area Low-Level Waste Facility Performance Assessment and SRS Composite Analysis program. In this assignment he is responsible for coordinating work, budget, schedule and resources providing technology development and applications in support of the SRS low-level solid waste mission. Areas of support include geochemical and groundwater modeling; radiological performance assessments and low-level waste disposal technology; radioactive waste laboratory treatability studies; and field programs evaluating performance of waste forms and barriers. Previous assignments at SRS have included 13 years as line manager of the Radiological Performance Assessment Group, and three years each as an environmental project manager in the Environmental Restoration Department and as a solid waste technology manager (and process engineer) in the Waste Management Technology Department.

Contributions: Composite Analysis project planning and management oversight.

COLLARD, LEONARD B., Savannah River National Laboratory

M.S. Civil Engineering, University of Arizona
B.A. Mathematics, Duke University

Experience: Mr. Collard is a Principal Engineer for Savannah River National Laboratory with over twenty years of experience. Recent work has focused on Performance Assessment activities. Earlier work included water and sewer planning and foundation analyses.

Contribution: Mr. Collard was a co-author of the aquifer flow path parameters report and provided a design check of the use of the LAW Vault Performance Assessment flux to the water table within the CA.

COOK, JAMES R., Consultant

M.S., Geochemistry - State University of New York at Binghamton, 1977
B.S., Geology - University of Arizona, Tucson, 1970

Experience: Jim Cook retired as a Senior Fellow Geologist from the Savannah River National Laboratory in 2006 after working for 25 years in various aspects of low-level waste research. Research topics have included site selection, site characterization, site closure, performance assessment and probabilistic uncertainty analysis. He was the technical team lead for the original SRS Composite Analysis, as well as early SRS Performance Assessments for the E-Area and Saltstone disposal facilities. Mr. Cook is currently a consultant.

Contribution: Mr. Cook is a co-author of the radionuclide inventory report and the dose module report. He reviewed a number of reports and CA sections and served on the Core Team.

CRAPSE, KIMBERLY POWELL, Savannah River National Laboratory

Ph.D. Organometallic Chemistry – University of North Carolina at Chapel Hill
B.S. with Honors in Chemistry – Wake Forest University

Experience: Dr. Crapse has 9 years experience at SRS. Her assignments at the Savannah River National Laboratory include environmental research and high-level and low-level waste research. Dr. Crapse supports the Composite Analysis Monitoring Program including annual reviews of performance and periodic revisions to the CA Monitoring Plan.

Contributions: Dr. Crapse completed the design check review of the distribution (or sorption or partition) coefficients (Kds), their distributions, and the cellulose degradation product correction factors used in the CA. Dr. Crapse participated in the CA Core Team Meetings.

DEAN, BEN, Savannah River Remediation C&WDA

B.S. Chemical Engineering – Clemson University

Experience: With over 12 years experience, six at SRS, Mr. Dean has primarily worked with characterization of waste within the tank farms. Prior to joining WSRC, Mr. Dean spent six years in the chemical industry (chlor-alkali and fiberglass) with experience in operations, process engineering and statistics.

Contribution: Assisted in development of the SRS end state residual radioactive material inventories for HTF, FTF, DWPF, SPF, and SWPF.

DIXON, KENNETH, Savannah River National Laboratory

M.E. Civil Engineering – University of South Carolina
M.S. Agricultural Engineering – University of Georgia
B.S. Agricultural Engineering – University of Georgia

Experience: Mr. Dixon has 18 years experience at SRS. His research has focused primarily on soil and groundwater characterization and remediation. He has conducted pilot scale testing of innovative remedial technologies and numerical modeling of contaminant fate and transport in the vadose zone.

Contribution: Mr. Dixon is a co-author for the vadose zone data report and material property, infiltration, and saturation data report. In addition he provided design checks of various portions of the CA work.

EDWARDS, THOMAS B., Savannah River National Laboratory

Ph.D. Management Science – Clemson University
M.S. Mathematics – Clemson University
B.S. Mathematics – Clemson University

Experience: Dr. Edwards has 19 years experience at SRS. His assignments include statistical support for the Defense Waste Processing Facility and for the Saltstone variability study.

Contribution: Dr. Edwards served as technical reviewer for some of the supporting statistical analyses.

FLACH, GREGORY, Savannah River National Laboratory

Ph D. Mechanical Engineering – North Carolina State University
M.S. Mechanical Engineering – North Carolina State University
B.S. Mechanical Engineering – University of Kentucky

Experience: Dr. Flach is a Fellow Engineer at SRNL with over 20 years of experience related to groundwater hydrology, computational simulation, and numerical code development. He has been the principal investigator on a number of groundwater modeling studies at SRS involving regional and local scale hydrology, contaminant migration from waste sites, and evaluation of environmental cleanup alternatives. Over the last decade his efforts have focused on PA and CA related projects and research involving dual-domain formulations of contaminant transport.

Contribution: Dr. Flach performed Design Checks on aquifer flow and transport modeling.

FLETCHER, DEAN E., Savannah River Ecology Laboratory

Experience: Mr. Fletcher has worked extensively in SRS streams over the past 20 years. Much work has focused on the life histories of southeastern fishes with emphasis on the reproductive biology of sunfish and minnow species. However, he has conducted a diverse array of projects including characterization of fish movements in contaminated streams, quantification of reservoir escapement, stream habitat and fish community assessments, use of animals as bioindicators of metal and metalloid contamination, and co-authoring a regional fish book. His program is focused on establishing a baseline of disturbance for SRS streams, with a long term goal of establishing a headwater stream mitigation bank.

Contribution: Mr. Fletcher is a co-author of the SRS C14 in fish study.

HAMM, L. LARRY, Savannah River National Laboratory

B.S., M.S., PhD., Chemical Engineering, University of South Carolina

Experience: Dr. Hamm, with twenty-five years of experience, currently serves as a Fellow Engineer at SRNL who has been a technical contributor, technical director, and program manager on a wide variety of DOE nuclear reactor, waste management, and environmental cleanup related programs. Dr. Hamm co-authored the FACT code used for modeling subsurface contaminant transport at various DOE waste sites. Recent work has focused on Performance Assessment and Composite Analysis activities at SRS.

Contribution: Dr. Hamm developed the aquifer flow path parameters input for the CA and performed the uncertainty analysis.

HIERGESELL, ROBERT, Savannah River National Laboratory

M.S. Hydrogeology – University of Nebraska-Lincoln
B.S. Geology – Virginia Polytechnic University

Experience: Mr. Hiergesell is a Fellow Scientist in the Radiological Performance Assessment Group of Savannah River National Laboratory. He has been employed at the SRNL for the past 20 years and prior to this he was employed on the Basalt Waste Isolation Project at the DOE Hanford Site. His experience includes applied research in the arenas of groundwater monitoring, subsurface flow and transport simulation, environmental remediation and in performance assessment for low-level radioactive waste disposal. He has been the lead technical investigator for numerous environmental restoration and waste management projects at the DOE Savannah River Site.

Contribution: Mr. Hiergesell is a co-author for the inventory, vadose zone data, CA base case model, and CA sensitivity and uncertainty model reports. In addition he provided design checks of various portions of the CA work.

HINTON, THOMAS GLENN, Savannah River Ecology Laboratory

Ph.D. Radiation Ecology, Colorado State University, Fort Collins
M.S. Radiation Biology, Colorado State University, Fort Collins
B.S. Wildlife Biology, Colorado State University, Fort Collins

Experience: Dr. Hinton has over 20 years experience in ecological research. His research and teaching interests include transport and fate of radioactive contaminants in aquatic and terrestrial ecosystems; effects of low-level, chronic irradiation on biota; human and ecological effects of chronic, low-dose contaminant mixtures using medaka fish as a model organism; transgenerational effects from contaminant exposures; coupling effects across levels of biological organization -- from molecules to ecosystems; human and ecological risk analyses; remediation of radioactively contaminated ecosystems; urban radiation ecology; contaminant modeling.

Contribution: Dr. Hinton is the principal author of the SRS C14 in fish study.

JANNIK, G. TIMOTHY, Savannah River National Laboratory

M.S. Health Physics – Georgia Institute of Technology
B.S. Mechanical Engineering – Villanova University

Experience: Mr. Jannik is a Senior Technical Advisor at SRNL. He has over 30 years of experience in the nuclear industry, 20 of which have been at the Savannah River Site. He is the technical lead of the Environmental Dosimetry Group at SRNL and he specializes in 1) environmental dosimetry, 2) effluent and environmental monitoring, and 3) human health risk analyses.

Contribution: Mr. Jannik consulted on development of the dose models and review of the dose calculations, source term, and uncertainty analysis.

JONES, WILLIAM E., Savannah River National Laboratory

M.S. Geology – East Carolina University
B.A. Anthropology – East Carolina University

Experience: Mr. Jones is a Principal Scientist at SRNL, with over 20 years of experience related to geology, groundwater hydrology, geochemistry, and geotechnical engineering. He has been involved in closure cap installation, design, and performance evaluation and principal investigator for a variety of groundwater geochemistry and environmental clean-up projects.

Contribution: Mr. Jones' focus for the Composite Analysis (CA) has been development of stream-flow values for input to CA modeling document (Savannah River Site Surface Water Flow Rates for Composite Analysis Dose Calculations SRNL-STI-2009-00077), design check for input data to the GoldSim™ model used to perform generic dose calculations, and design check for sensitivity analyses.

KAPLAN, DANIEL, Savannah River National Laboratory

Ph.D. Geochemistry - University of Georgia
M.S. Soil Science - University of New Hampshire
B.S. Plant and Soil Science - University of New Hampshire

Experience: Dr. Kaplan has 16 years experience working with developing geochemical models for Performance Assessments, the first five years while working for the Pacific Northwest National Laboratory and the remaining 11 years while working for the SRNL. He has contributed to various performance assessments at the Hanford Site and the SRS, and the Yucca Mountain Repository.

Contribution: Dr. Kaplan conducted many of the geochemical experiments used to develop the geochemical conceptual models employed by the Composite Assessment models. He has also developed the geochemical data package from which all geochemical input data, including the conceptual models, are based. He is co-author of the C14 geochemistry, distribution coefficients, and carbon dynamics in Fourmile Branch reports.

LEE, PATRICIA L., Savannah River National Laboratory

Ph.D., Health Physics, Georgia Institute of Technology, Atlanta, 1998
M.S., Physics, Clark Atlanta University of California, Atlanta, 1992
B.S., Physics, Lincoln University (PA), 1987

Experience: Dr. Lee is the manager for the Applied Computational Engineering and Statistics Group in Savannah River National Laboratory (SRNL). She has a diverse scientific background including over 18 years of experience in Health Physics. For the past nine years, she has worked in the SRNL supporting and performing research associated with dose and risk assessments for a variety of Savannah River Site (SRS) projects. Her support of performance assessment activities at SRS is primarily focused on inadvertent intruder analysis, air-pathway modeling and site-specific input parameters.

Contribution: Dr. Lee supported development of the radionuclide screening model for the Savannah River Site's Composite Analysis and served as a technical reviewer where appropriate.

MCARTHUR, J. VAUN, Savannah River Ecology Laboratory

Ph.D. Biology/Ecology, Kansas State University, Manhattan, KS
M.S. Zoology, Brigham Young University, Provo, UT
B.S. Zoology, Brigham Young University, Provo, UT

Experience: Dr. McArthur has over 20 years experience in ecological research. His research interests include aquatic microbial ecology, ecological genetics of microbes, interactions between terrestrial and aquatic ecosystems, stream community metabolism, and macroinvertebrate ecology. His principal research projects include indirect selection for antibiotic resistance in coastal oceans; indirect selection for antibiotic-resistant bacteria by heavy metal exposure; diversity and distribution of mercury resistance genes in lotic bacteria of the SRS; and genetic probes for bacteria involved in the degradation of TCE.

Contribution: Dr. McArthur is a co-author of the SRS C14 in fish study.

MCDOWELL-BOYER, LAURA, Consultant

Ph.D. Civil/Environmental Engineering - University of California, Berkeley
M.S. Radiological Health Physics - Colorado State University, Fort Collins
B.S. Physical Science – Colorado State University, Fort Collins

Experience: Dr. McDowell-Boyer has 24 years experience related to radiological exposure assessment, including multi-media environmental transport models, mechanisms of subsurface contaminant migration via colloids, and groundwater flow and transport. Dr. McDowell-Boyer was a co-principal investigator from ORNL on the first SRS Z-Area and E-Area PAs, contributed to the revised E-Area PA, and was responsible for pathway and dose analysis as well as much of the documentation of the first SRS CA. She has developed source terms for health risk assessments at LLNL, and conducted environmental dose reconstructions for workers at the Rocky Flats Plant, LANL, ANL, the Weldon Spring Plant, and SNL in California.

Contribution: Dr. McDowell-Boyer co-authored the background information report, the radionuclide screening report, the Kd/CDP report, and the dose module report, participated in design checks on inventory and decay chains, and carried out many of the tasks required for compiling the final CA report.

NOFFSINGER, DAVID, Principal Engineer, Area Completion Projects

M.S. Hydrogeology – Kent State University (1994)
B.S. Geology – University of Akron (1984)

Experience: Mr. Noffsinger has 12 years experience at SRS. His assignments include performing and managing numerical groundwater modeling studies for SRS operable units, and conducting environmental fate and transport evaluations for site closure. Prior to SRS, Mr. Noffsinger worked in environmental consulting for six years, and for BP Research for seven years. While at BP, Mr. Noffsinger worked in the Petroleum Geochemistry, and Hydrocarbon and Particle Technology groups.

Contribution: Mr. Noffsinger is a co-author of the vadose zone data report, and consulted in development of groundwater pathway analysis and contaminant flux rates to surface water.

PERONA, RALPH A., Neptune and Company, Inc.

M.S. Environmental Health, University of Washington, 1992
B.S., Food Science, Rutgers University, 1985
D.A.B.T., Diplomate of the American Board of Toxicology, 2006

Experience: Mr. Perona has 17 years of experience in assessing human exposure and associated health risks related to environmental contamination. In this practice Mr. Perona integrates numerous disciplines including toxicology, health physics, chemistry, statistics, exposure assessment and computer modeling of contaminant fate and transport.

Contribution: Mr. Perona developed the Dose Module component of the CA GoldSim™ computer model.

PHIFER, MARK A., Savannah River National Laboratory

M.S. Civil Engineering (Environmental and Geotechnical) – University of Tennessee
B.S. Civil Engineering – Tennessee Tech
South Carolina Registered Professional Engineer (No. 12310)

Experience: Mr. Phifer is a Senior Fellow Engineer with the Savannah River National Laboratory. He has 26 years of environmental and geotechnical experience at SRS. The first 10 years included environmental regulatory compliance, civil/environmental design, project engineering (closure of a mixed waste landfill and basins (80 acres)), and management (environmental remediation technology). The subsequent 16 years have been at the Savannah River National Laboratory developing, deploying, and evaluating waste site closure, groundwater remediation, and radioactive waste disposal technologies. These technologies include horizontal and vertical barrier systems, diffusion barriers, closure caps (including their degradation), waste subsidence, low-level radioactive waste disposal facilities, cementitious barriers and waste forms, permeable reactive barriers, GeoSiphon/GeoFlow groundwater treatment systems, sulfate reduction remediation, reductive dechlorination, and vadose zone and aquifer characterization and testing. For the last 7 years Mr. Phifer has in addition worked on Performance Assessment and Composite Analysis related activities.

Contribution: Mr. Phifer was the SRNL technical lead responsible for development of the CA, meeting the schedule, leading the CA Core Team, and providing interface with review teams. He was co-author of the various CA planning reports and many of the technical reports including the background information; inventory; conceptual model; vadose zone data, material property, infiltration, and saturation data; aquifer flow path parameter; dose module; CA base case model; and CA sensitivity and uncertainty model reports. In addition he provided design checks of various portions of the CA work.

ROBERTS, KIMBERLY, Savannah River National Laboratory

Ph.D. Chemical Oceanography- Texas A&M University
M.S. Marine Environmental Science – SUNY-Stony Brook
A.B. Geology – Mount Holyoke College

Experience: Dr. Kimberly Roberts is a Senior Scientist at Savannah River National Laboratory. She has 1.5 years experience at SRS and 8 years experience in environmental geochemistry.

Contribution: Dr. Roberts has conducted experimental analyses related to Kd measurements, colloidal interactions, specific radionuclide behavior and is co-author of C14 geochemistry at the Savannah River Site for the SRS composite analysis.

ROMANEK, CHRISTOPHER S., University of Georgia

Ph.D. Geology -Texas A&M University 1991
MS Geology- University of Florida 1985

Experience: Dr. Romanek is an Associate Professor at the University of Georgia and he currently holds a joint appointment in the Department of Geology and the Savannah River Ecology Laboratory. His area of specialty is low-temperature geochemistry, particularly the application of stable isotopes to problems in the earth, ecological and environmental sciences. Prior to arriving at the University of Georgia, Dr. Romanek held a National Research Council Fellowship at NASA's Johnson Space Center, where he studied the low-temperature geochemistry of extraterrestrial materials, including those rocks thought to originate from Mars.

Contribution: Dr. Romanek conducted research related to dynamic modeling of the risk posed by C14 in Four-Mile Branch wetland ecosystem on SRS.

ROWLAND, PAULA, Savannah River National Laboratory

B.A. English – Charleston Southern University
A.A.S. Nuclear Engineering Technology – Aiken Technical College

Experience: Ms. Rowland has 22 years experience at SRS. Assignments have included analytical laboratory sample analyses, DWPF and DP training, SRNL procedures, and technical document editing.

Contribution: Ms. Rowland compiled and provided technical editing for production of the Composite Analysis.

SALDIVAR, Jr., ELOY, Manager, Regulatory Integration and Business Management Services

B.S. Mechanical Engineering – Texas A&M University

Experience: Mr. Saldivar has 28 years of nuclear/non-nuclear experience at the Savannah River Site and DuPont Petrochemicals working in the areas of Design Authority Engineering, Project Management, Program Management, Configuration Management, Risk Management, Maintenance, Training, Procurement, Program Planning, and most recently integration and business management for Regulatory/Environmental Services. His most significant contributions have been related to Design Authority engineering activities in support of Waste Removal, Tank Farm and Salt Processing Program Projects in which he managed engineering disciplines and worked directly with project management processes. He was the Principle Investigator for SRS for Waste Removal/Tank Closure technologies for six years working as a liaison with all DOE Complex sites involved with Waste Removal/Tank Closure technology deployments. His most significant accomplishment was the successful engineering management of the first two high level waste tank closures in the DOE Complex. Mr. Saldivar is a certified Project Management Professional currently serving on the Board of Directors for the local Project Management Institute Organization.

Contribution: Mr. Saldivar participated in development of the CA Steering and CA Core Teams. He was also co-author of several of the CA planning documents.

SHINE, GENE (E.) P., Savannah River National Laboratory

M.S. Statistics – Texas A & M.
B.S. Engineering – Pittsburgh

Experience: Mr. Shine is a Principal Scientist at SRNL and has 24 years of experience at SRS. He is a statistician engaged in a broad range of activities including nuclear nonproliferation, waste characterization, and environmental studies. His primary areas of statistical interest are finite population problems, resampling and permutation tests, calibration problems, multivariate analyses, and sampling. Previously, Gene has taught at the level of Lecturer at Texas A & M University and Visiting Assistant Professor at Oklahoma State University, and worked as an Engineer for the Eastman Kodak Company in Rochester, New York.

Contribution: Mr. Shine is author of the stream flow and infiltration distribution reports.

SMITH, FRANK G., Savannah River National Laboratory

Sc.D. Chemical Engineering – Massachusetts Institute of Technology
M.S. Chemical Engineering – California Institute of Technology
B.S. Chemical Engineering – University of Louisville

Experience: Dr. Smith has 28 years experience at SRS. His assignments have included three years in atmospheric transport modeling, eight years thermal-hydraulic modeling as part of the SRS reactor restart and APT programs, and 15 years working in various aspects of waste management R&D. Since 2007 Dr. Smith has worked in the Computational Sciences Division at SRNL primarily on chemical process modeling.

Contribution: Dr. Smith led the effort to develop the GoldSim™ transport model and perform the CA base case deterministic, sensitivity and uncertainty calculations. Dr. Smith was lead author on the reports documenting the CA base case deterministic calculations and the sensitivity and uncertainty calculations and co-author on the conceptual model and aquifer transport modeling reports.

SWINGLE, II, ROBERT F., Savannah River National Laboratory

B.S., Chemical Engineering, University of South Carolina, Columbia, 1981
B.S., Chemistry, University of South Carolina, Columbia, 1979

Experience: Mr. Swingle is a Senior Technical Advisor with the Savannah River National Laboratory, with twenty-eight years of experience at the Savannah River Site. He began his career by working thirteen years with nuclear production reactor thermo-hydraulics, physics, and moderator chemistry. This was followed by ten years researching treatment and characterizing radioactive waste. For the last five years, he has been working for the Environmental Analysis and Performance Assessment Group of the Savannah River National Laboratory supporting low level radioactive waste disposal activities at the Savannah River Site.

Contribution: Mr. Swingle supported source release modeling, flow and inventory sensitivity studies and technical review.

TAUXE, JOHN, PhD, PE, Neptune and Company, Inc.

Ph.D., Civil Engineering, University of Texas at Austin, Austin, Texas, 1994
M.S., Civil Engineering, University of Texas at Austin, Austin, Texas, 1990
B.A., Earth Science, Wesleyan University, Middletown, Connecticut, 1984

Experience: Dr. Tauxe has been working in the earth and environmental sciences and engineering since 1981, and has developed expertise in quantitative hydrology and hydrogeology, and in computer programming, concentrating in the modeling of contaminant fate and transport in the environment. His professional experience is broad, however, including marine geology, radiolimnology, water resources assessment, hydropower systems modeling, regulatory interpretation, metrication, watershed mapping, radiological performance assessment, and training of environmental professionals. Since 1998, Dr. Tauxe has been advancing the field of probabilistic performance assessment within the Department of Energy and the Nuclear Regulatory Commission, and is an internationally recognized expert in radiological performance assessment modeling.

Contribution: For this study, Dr. Tauxe has served as advisor to the GoldSim™ programmers of the CA and participated in the development of the CA dose module.

TAYLOR, GLENN A., Savannah River National Laboratory

M.S. Mechanical Engineering, University of Texas-Austin, 1979
B.S. Engineering Physics, University of Louisville, 1977

Experience: Mr. Taylor has 30 years of experience in code development, modeling, and research. His primary emphasis has been non-equilibrium thermal-hydraulics and chemical process methods development and modeling. He has been involved with PRA-type analyses since the mid-1980's. He has been doing groundwater modeling and contaminant transport modeling for the past 5 years.

Contribution: Mr. Taylor conducted the radionuclide screening analysis and assisted in the design check of the CA Base Case transport model.

WHITESIDE, TAD, Savannah River National Laboratory

Ph. D. Physical Chemistry – University of Georgia
B.S. Chemistry – Erskine College

Experience: Dr. Whiteside is a Senior Scientist in the Environmental Science and Biotechnology division at the Savannah River National Laboratory. He did his undergraduate work at Erskine College where he majored in Chemistry and Physics. He then went on to the University of Georgia and worked with Lionel Carreira to expand the SPARC modeling program to predict the heats of formation of organic compounds. He obtained his PhD in 2004 and stayed on with Dr. Carreira as a post-doc where he interfaced with the US Environmental Protection Agency to integrate SPARC for use as part of their chemical regulatory program. While doing this he continued his molecular modeling research by developing models that predict how phosphate-esters hydrolyze in acidic, neutral, and basic solutions. In his current position at SRNL his primary job is to take the physical properties of various nuclides and the composition of the surrounding region and model how these contaminants will move through the environment under various scenarios. He has multiple collaborations in various areas. These include working with Dr. Cliff Padgett (Armstrong Atlantic State University) to predict the physical and mechanical properties of nano-diamond structures.

Contribution: Dr. Whiteside performed the design check of the PORFLOW modeling conducted to develop saturation estimates and distributions for the CA.

WILHITE, ELMER, Savannah River National Laboratory

M.S. Inorganic Chemistry – Washington University
B.S. Chemistry – University of Missouri, Columbia

Experience: Mr. Wilhite has 40 years experience at SRS. His assignments include environmental research, high-level and LLW research, and supervision of environmental monitoring and analytical chemistry groups. Mr. Wilhite has served as a consultant to DOE Headquarters on LLW management for 23 years. He was the chairman of the former DOE Peer Review Panel and the technical lead for DOE for the radiological assessment section of the response to the DNFSB recommendation 94-2.

Contribution: Mr. Wilhite consulted on development of the conceptual model, is a co-author of the pathway screening and radionuclide screening reports, and consulted in development of the GoldSim™ transport and dose models.

YOUNG, KAREN E., Consultant

B.S., Environmental Resource Management, The Pennsylvania State University, 1986

Experience: Ms. Young has over 19 years of experience as an environmental scientist with expertise in regulatory compliance. She specializes in project management, regulatory agency interface/liaison services, and preparation of environmental compliance documentation. She has assisted DOE, DOD, and private industry with RCRA, CERCLA, NEPA, NRC, and DOE Order compliance and has assisted the EPA in developing RCRA regulations. Ms. Young has managed the preparations of regulatory compliance and remediation permit applications and reporting documentation for the Mixed Waste Management Facility, the Old Radioactive Waste Burial Ground, and the F- and H-Seepage Basins at the Savannah River Site as well as providing support services for preparation of the Z-Area and E-Area Low-level Waste Facility Performance Assessments, Monitoring Plans, Closure Plans and the Composite Analysis at the Savannah River Site.

Contribution: Ms. Young assisted with documentation preparation and coordination, DOE Order compliance, and consistency, comprehensiveness, and design check review.

13.2 COMPOSITE ANALYSIS CORE TEAM MEMBERS

Table 13-1 provides a list of the Composite Analysis Core Team Members, along with the organization they represented.

Table 13-1. Composite Analysis Core Team Members

Composite Analysis Core Team Member	Current Organization	Composite Analysis Function
Tom Butcher	SRNS Savannah River National Laboratory	Budget
Jim Cook	Consultant	PA/CA Expert
Dave Crowley	SRNS Savannah River National Laboratory	Management
Sonny Goldston	SRNS Regulatory Integration & Business Management	Solid Waste Representative and Manager of Disposal Authorization Statement
Larry Hamm	SRNS Savannah River National Laboratory	Modeling
Bob Hiergesell	SRNS Savannah River National Laboratory	Modeling
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Table 13-1. Composite Analysis Core Team Members - continued

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[1] Prior to his retirement from DOE in January, 2009, Howard Pope was the DOE customer representative.

Other contributors worthy of recognition include:

- Greg Rucker of SRNS Area Completion Projects - Remediation Support who stood in as the Environmental Restoration Representative on the Composite Analysis Core Team when Cathy Lewis was not available.
- Don Sink of SRNS Waste Management - Facility/LLW Engineer who assisted Marshall Looper as the Solid Waste Representative on the Composite Analysis Core Team.
- James Vaughan of Savannah River National Laboratory Quality Assurance who assisted with the Composite Analysis Quality Assurance and Quality Control.

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VOLUME II (APPENDICES)

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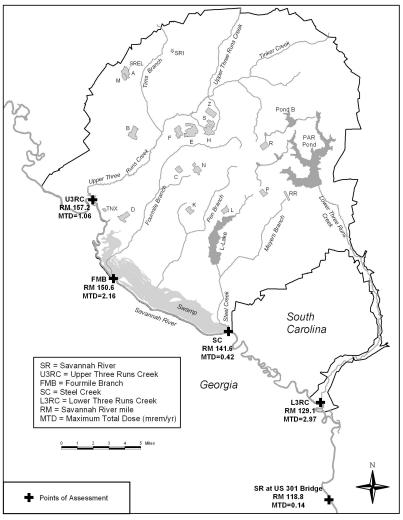

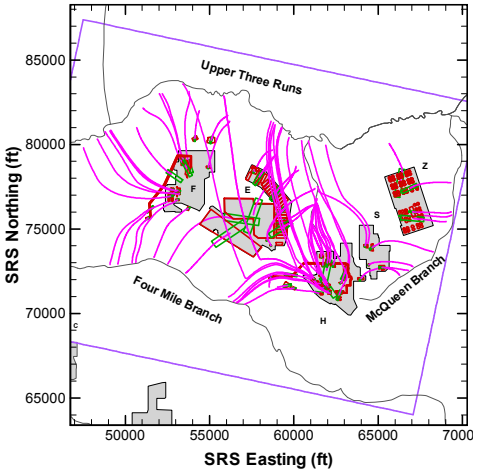

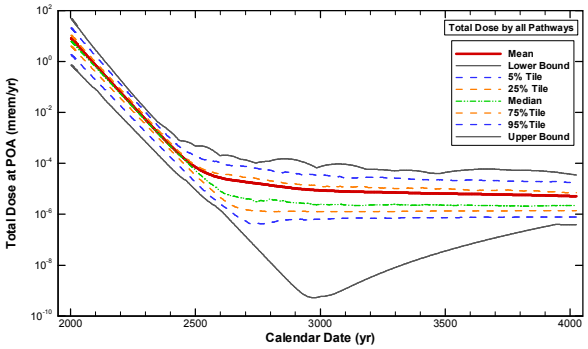

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LIST OF ACRONYMS AND ABBREVIATIONS.

See Volume I for the list of acronyms and abbreviations.

A. APPENDIX A: COMPOSITE ANALYSIS INVENTORY TABLES

All inventories contained in this appendix, except where otherwise noted, were obtained from Hiergesell et al. (2008).

A.1 A/M FACILITIES

Table A-1. 2002 Radionuclide Inventory for the M-Area HWMF

Radionuclide	Inventory (Ci)
U235	8.99E-03
U238	6.98E-01
U234	6.50E-02

Table A-2. 2002 Radionuclide Inventory for the M-Area Inactive Process Sewer Line

Radionuclide	Inventory (Ci)
U235	6.48E-05
U238	5.03E-03
U234	4.69E-04

Table A-3. 2002 Radionuclide Inventory for the SRL Seepage Basins

Radionuclide	Inventory (Ci)
Ac228	3.50E-02
Co60	1.00E-02
Cs137	1.33E+00
K40	4.60E-02
Pb212	2.80E-01
Ra228	3.50E-02

Table A-4. 2025 773-A Residual Inventory

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Am241	2.4E-01	Pu238	1.6E+00
Am243	6.4E-03	Pu239	1.5E+00
Ba137m	4.2E+01	Pu240	3.3E-01
C14	7.3E-04	Pu241	1.2E+01
Ce144	1.8E+00	Pu242	4.8E-04
Cf249	1.4E-04	Rh106	1.5E+00
Cf252	3.8E-01	Ru106	1.5E+00
Cm244	1.8E+00	Se79	1.9E-04
Cm246	6.7E-03	Sn126	2.3E-05
Co60	5.9E+01	Sr90	6.4E+01
Cs137	4.4E+01	Tc99	7.3E-03
H3	1.1E+01	Th232	3.6E-05
I129	1.3E-05	U233	1.3E-10
Ni59	8.6E-04	U234	2.0E-03
Np237	4.6E-04	U235	9.9E-05
Pm147	2.1E+01	U236	3.2E-04
Pr144	1.8E+00	U238	6.3E-03
Pr144m	2.6E-02	Y90	6.4E+01

Table A-5. 2025 776-A Residual Inventory

Radionuclide	Inventory (Ci)
Am241	6.5E-05
Pu238	3.1E-02
Pu239	1.9E-04
Pu240	6.8E-05
Pu241	1.8E-02
Pu242	2.9E-07

Table A-6. 2025 778-A Residual Inventory

Radionuclide	Inventory (Ci)
Am241	5.8E-04
Cf249	8.0E-09
Cm244	8.1E-05
Cm246	6.1E-07
Pu238	9.2E-04
Pu239	1.7E-03
Pu240	4.0E-04
Pu241	5.8E-03
Pu242	2.6E-06
Sr90	1.4E-03
U235	1.3E-08
U238	6.7E-07
Y90	1.4E-03

Table A-7. 2002 Radionuclide Inventory for Steed Pond and Tims Branch

Radionuclide	Inventory (Ci)
Steed Pond	
U235	1.25E-01
U238	9.73E+00
U234	9.06E-01
Tims Branch	
U235	6.27E-02
U238	4.87E+00
U234	4.53E-01

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A.2 D-AREA FACILITIES

Table A-8. 2006 Radionuclide Inventory for Building 420-D

Radionuclide	Inventory (Ci)
H3	2.75E+00
Th232	3.10E-04
Pu239	4.90E-04
Cs137	5.62E-04

Table A-9. 2005 Radionuclide Inventory for Building 420-2D Rework Handling Facility

Radionuclide	Inventory (Ci)
H3	3.19E-01
Ac228	4.12E-06
Ra226	2.78E-06
Cs137	2.08E-06
K40	1.61E-04

Table A-10. 2005 Radionuclide Inventory for Building 421-D Finishing Building

Radionuclide	Inventory (Ci)
H3	1.07E-01

Table A-11. 2004 Radionuclide Inventory for Building 421-2D Moderator Handling and Storage

Radionuclide	Inventory (Ci)
H3	3.92E-01
Co60	5.60E-05
Cs137	2.70E-06
Eu152	3.15E-06
Eu154	2.25E-06
Am241	4.05E-06

Table A-12. 2004 Radionuclide Inventory for Building 772-D Control Laboratory and Supervisor Office

Radionuclide	Inventory (Ci)
H3	4.90E-02
K40	3.27E-05
Co60	9.98E-06
Cs137	1.33E-04

A.3 E-AREA FACILITIES

Table A-13. 2025 Radionuclide Inventory for CIG Trenches

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Ac227	2.00E+00	Eu152	3.16E-01	Sb124	2.00E+00
Ag108m	2.00E-03	Eu154	2.10E-01	Sb125	4.52E-02
Al26	2.20E-03	Eu155	3.24E-02	Sc46	2.00E+00
Am241	4.78E-01	H3	2.14E+04	Se75	2.00E+00
Am242m	3.34E-06	I129	5.42E-05	Se79	2.58E-03
Am243	5.01E-03	I129_ETF [2]	0.00E+00	Sm151	5.06E-02
Ar39	2.00E+00	I129_KB [2]	2.42E-03	Sn113	2.00E-12
Ba133	6.00E-01	K40	8.98E-09	Sn119m	2.00E+00
Bi207	2.00E+00	Kr85	2.06E-01	Sn121	2.00E+00
Bk249	2.00E+00	Mo93	2.00E-03	Sn121m	2.00E+00
C14	2.18E-01	Na22	2.00E-06	Sn123	2.00E+00
C14_KB [2]	4.60E-01	Nb93m	2.00E-02	Sn126	2.88E-04
Ca41	1.00E-04	Nb94	9.20E-04	Sr90	5.96E+01
Cd113m	2.00E-07	Ni59	1.16E-02	Tc99	2.60E-02
Cf249	5.86E-04	Ni63	1.16E+00	Tc99_KB [2]	7.52E-02
Cf250	1.21E-05	Np237	8.34E-03	Th228	2.62E-02
Cf251	5.94E-04	Pa231	2.00E-07	Th229	2.00E-04
Cf252	6.36E-04	Pb210	5.76E-11	Th230	2.00E-04
Cl36	2.00E-03	Pd107	4.50E-05	Th232	2.00E-03
Cm242	4.98E-06	Pu238	1.52E+00	U232	2.62E-02
Cm243	9.98E-04	Pu239	2.06E+00	U233	1.69E+00
Cm244	1.31E+00	Pu240	1.97E-01	U234	2.60E-01
Cm245	1.03E-04	Pu241	6.20E+00	U235	6.46E-03
Cm246	5.84E-04	Pu242	9.04E-04	U236	6.58E-03
Cm247	5.84E-04	Pu244	2.00E-15	U238	9.30E-01
Cm248	5.84E-04	Ra226	5.22E-11	W181	0.00E+00
Co60	1.10E+00	Ra228	2.00E-03	W185	0.00E+00
Cs134	2.80E-01	Rb87	3.50E-11	W188	0.00E+00
Cs135	2.98E-05	S35	2.00E+00	Zr93	6.78E-03
Cs137	1.38E+04				

[1] Total E-Area LLWF CIG Trench closure inventory for two CIG footprints.
[2] Special waste form considered in the E-Area PA.

Table A-14. 2025 Radionuclide Inventory for Engineered Trenches

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Ac225	1.19E-02	Np237	2.52E-02
Ac227	2.00E-03	Np239	1.52E-03
Ac228	1.19E-02	Pa231	2.00E-03
Ag108	3.66E-08	Pa233	1.07E-03
Ag108m	3.92E-07	Pa234	3.76E-01
Ag109m	1.09E-04	Pa234m	5.98E-01
Ag110m	6.24E-06	Pb209	1.19E-02
Al26	1.29E-10	Pb210	9.36E-03
Am241	6.66E+00	Pb212	7.32E-02
Am242	3.12E-08	Pb214	1.03E-02
Am242m	6.12E-01	Pd107	2.00E-03
Am243	1.26E-01	Pm147	1.83E+00
Ar39	2.00E-03	Po210	2.56E-03
At217	1.19E-02	Po212	4.32E-02
Ba133	2.10E-04	Po213	1.14E-02
Ba137m	8.44E+01	Po214	1.03E-02
Be7	1.14E-05	Po216	7.36E-02
Bi207	2.04E-05	Po218	1.03E-02
Bi210	2.54E-03	Pr144	7.80E-02
Bi211	1.21E-06	Pr144m	1.71E-04
Bi212	7.36E-02	Pu238	1.13E+01
Bi213	1.19E-02	Pu239	3.34E+01
Bi214	1.03E-02	Pu240	7.26E+00
Bk249	2.00E-03	Pu241	1.98E+01
C14	2.60E-01	Pu242	2.06E-01
Ca41	2.00E-03	Pu244	1.03E-14
Cd109	1.44E-04	Ra224	7.36E-02
Cd113m	8.60E-11	Ra225	1.19E-02
Ce139	1.47E-10	Ra226	1.04E-02
Ce144	1.20E-01	Ra228	1.06E-02
Cf249	1.41E-01	Rb86	1.21E-04
Cf250	3.64E-04	Rb87	2.00E-14
Cf251	1.29E-01	Rh103m	4.42E-07
Cf252	1.73E-03	Rh106	2.32E-02
Cl36	1.82E-04	Rn220	7.36E-02
Cm242	8.20E-05	Rn222	1.03E-02
Cm243	2.00E-03	Ru103	4.42E-07
Cm244	1.07E+01	Ru106	5.14E-02
Cm245	2.64E-03	S35	9.34E-21
Cm246	1.17E-03	Sb124	2.00E-03
Cm247	1.40E-03	Sb125	2.52E-01
Cm248	2.06E-14	Sb126	1.90E-12

Table A-14. 2025 Radionuclide Inventory for Engineered Trenches - continued

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Co57	1.21E+01	Sb126m	4.42E-05
Co58	2.28E+00	Sc46	5.96E-21
Co60	2.68E+01	Se75	1.00E-03
Cr51	2.04E-02	Se79	1.95E-02
Cs134	1.16E-01	Sm151	8.56E-05
Cs135	3.44E-11	Sn113	3.74E-06
Cs137	8.92E+01	Sn119m	4.26E-05
Eu152	6.10E+01	Sn121	1.37E-12
Eu154	2.42E+01	Sn121m	1.58E-05
Eu155	8.04E-01	Sn123	2.00E-03
Fe55	8.86E+01	Sn126	1.29E-04
Fe59	6.10E-02	Sr85	4.00E-03
Fr221	1.19E-02	Sr89	2.00E-03
H3	3.00E+00	Sr90	1.43E+02
H3_ETF_Carbon [2]	0.00E+00	Ta182	4.06E-06
Hf175	1.28E-05	Tc99	2.00E-01
Hf181	2.22E-02	Te123m	6.74E-07
Hg203	3.36E-05	Te125m	7.88E-03
I129	1.40E-04	Th228	7.36E-02
I129_ETF_Carbon [2]	0.00E+00	Th229	1.19E-02
I129_ETF_Carbon [2]	0.00E+00	Th230	1.72E-02
I129_F_Dowex_21K [2]	2.64E-03	Th231	1.64E-02
I129_H_Dowex_21K [2]	1.81E-03	Th232	1.10E-02
I129_H_Area_Filtercake [2]	0.00E+00	Th234	8.52E-01
I129_F_Area_CG8 [2]	3.12E-06	Tl208	2.42E-02
I129_H_Area_CG8 [2]	1.91E-05	Tl209	4.60E-04
I129_ETF_GT73 [2]	1.49E-06	Tm170	1.14E-04
I129_F_Area_Filtercake [2]	7.20E-05	U232	6.22E-02
In113m	3.74E-06	U233	4.62E+00
K40	3.42E-04	U234	1.04E+00
Kr85	1.00E-01	U235	3.48E-02
Mn54	5.16E-01	U236	6.50E-02
Mo93	5.40E-03	U238	8.78E-01
Na22	1.03E-05	W181	2.00E-03
Nb93m	1.58E-01	W185	2.00E-03
Nb94	5.54E-03	W188	2.00E-03
Nb95	1.12E-01	Y90	1.43E+02
Nb95m	2.82E-05	Zn65	6.30E-02
Ni59	1.84E-01	Zr93	4.42E-05
Ni63	1.72E+01	Zr95	4.50E-02

[1] Total E-Area LLWF Engineered Trench closure inventory for two Engineered Trenches.
[2] Special waste form considered in the E-Area PA.

Table A-15. 2025 Radionuclide Inventory for Intermediate Level Vault

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Ac227	0.00E+00	Fe59	6.36E-02	Rb87	0.00E+00
Ac228	7.16E-05	H3	3.61E+06	Rh103m	2.74E-04
Ag108m	0.00E+00	H3_TPBAR [2]	2.71E+06	Rh106	1.73E-01
Ag110m	1.20E-08	Hf175	7.48E-05	Rn220	5.14E-04
Al26	0.00E+00	Hf181	2.44E-02	Rn222	3.07E+00
Am241	1.54E+00	I129	4.40E-03	Ru103	2.74E-04
Am242m	2.76E-03	I129_ETF [2]	1.59E-01	Ru106	4.36E-01
Am243	4.16E-03	I129_KB [2]	6.03E-04	S35	0.00E+00
Ar39	0.00E+00	In113m	2.00E-05	Sb124	0.00E+00
Ba133	0.00E+00	K40	3.10E-06	Sb125	4.11E-01
Ba137m	7.09E+02	Kr85	8.13E+01	Sc46	0.00E+00
Bi207	0.00E+00	Mo93	0.00E+00	Se75	0.00E+00
Bi210	2.44E-06	Mn54	1.69E+00	Se79	4.03E-04
Bi211	1.27E-06	Na22	0.00E+00	Sm151	0.00E+00
Bi212	5.12E-04	Nb93m	1.07E-02	Sn113	2.00E-05
Bi214	3.07E+00	Nb94	3.88E+01	Sn119m	0.00E+00
Bk249	0.00E+00	Nb95	9.83E-02	Sn121	0.00E+00
C14	3.88E+00	Nb95m	3.88E-04	Sn121m	0.00E+00
C14_KB [2]	1.72E+01	Ni59	2.81E-01	Sn123	0.00E+00
Ca41	0.00E+00	Ni63	5.48E+01	Sn126	1.94E-03
Cd113m	0.00E+00	Np237	1.95E-02	Sr90	1.13E+02
Ce144	1.78E+00	Pa231	0.00E+00	Ta182	1.38E-04
Cf249	2.23E-06	Pa234	5.18E-01	Tc99	6.73E-01
Cf250	0.00E+00	Pa234m	7.10E-01	Tc99_KB [2]	8.16E-02
Cf251	6.82E-06	Pb210	3.07E+00	Te125m	1.42E-02
Cf252	0.00E+00	Pb212	5.14E-04	Th228	5.14E-04
Cl36	0.00E+00	Pb214	3.07E+00	Th229	0.00E+00
Cm242	8.07E-05	Pd107	0.00E+00	Th230	7.28E-05
Cm243	2.85E-02	Pm147	1.30E+01	Th231	7.76E-03
Cm244	2.38E+00	Po210	2.44E-06	Th232	1.97E-04
Cm245	4.24E-05	Po212	2.10E-06	Th234	1.23E+00
Cm246	6.73E-05	Po214	3.07E+00	Tl208	4.59E-07
Cm247	9.48E-10	Po216	5.14E-04	U232	4.25E-04
Cm248	1.22E-14	Po218	3.07E+00	U233	4.91E-01

Table A-15. 2025 Radionuclide Inventory for Intermediate Level Vault - continued

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Co57	2.03E-04	Pr144	1.53E+00	U234	9.64E-01
Co58	9.20E+00	Pr144m	1.35E-02	U235	3.42E-02
Co60	2.76E+06	Pu238	5.81E+00	U236	6.78E-03
Cr51	3.37E-02	Pu239	1.45E+00	U238	2.30E+00
Cs134	3.67E+00	Pu240	2.45E-01	W181	0.00E+00
Cs135	0.00E+00	Pu241	6.10E+00	W185	0.00E+00
Cs137	9.87E+02	Pu242	1.33E-02	W188	0.00E+00
Eu152	1.66E-04	Pu244	4.62E-02	Y90	9.44E+01
Eu154	8.97E-01	Ra224	5.14E-04	Zn65	1.43E+00
Eu155	6.13E-02	Ra226	3.07E+00	Zr93	2.56E-05
Fe55	3.46E+02	Ra228	7.16E-05	Zr95	5.82E-02

[1] Total E-Area LLWF ILV closure inventory for one ILV.
[2] Special waste form considered in the E-Area PA.

Table A-16. 2025 Radionuclide Inventory for Low Activity Waste Vault

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Ac225	8.72E-04	Fe59	7.14E-02	Rh106	2.01E-01
Ac227	0.00E+00	Fr221	8.72E-04	Rn220	3.25E-02
Ac228	7.30E-03	H3	2.13E+07	Ru106	5.97E-01
Ag108m	0.00E+00	Hf181	1.91E-02	S35	0.00E+00
Al26	0.00E+00	Hg203	4.27E-06	Sb124	0.00E+00
Am241	3.27E+00	I129	1.08E-03	Sb125	1.21E+00
Am242m	2.35E-02	I129_H [2]	1.69E-04	Sb126m	5.52E-06
Am243	8.87E-03	I129_J [2]	1.51E-04	Sc46	0.00E+00
Ar39	0.00E+00	K40	2.93E-05	Se75	0.00E+00
At217	8.72E-04	Kr85	1.10E+01	Se79	4.62E-01
Ba133	3.27E-01	Mn54	4.49E-01	Sm151	2.22E-04
Ba137m	3.16E+02	Mo93	0.00E+00	Sn113	3.63E-06

Table A-16. 2025 Radionuclide Inventory for Low Activity Waste Vault - continued

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Bi207	0.00E+00	Na22	3.29E-06	Sn119m	0.00E+00
Bi211	9.35E-05	Nb93m	3.45E-02	Sn121	0.00E+00
Bi212	3.25E-02	Nb94	4.50E-02	Sn121m	0.00E+00
Bi213	8.72E-04	Nb95	1.48E-01	Sn123	0.00E+00
Bi214	3.65E-06	Nb95m	3.42E-05	Sn126	1.70E-03
Bk249	0.00E+00	Ni59	3.72E-01	Sr85	7.52E-03
C14	1.46E+00	Ni63	1.79E+01	Sr89	8.43E-07
Ca41	0.00E+00	Np237	2.12E-01	Sr90	1.19E+03
Cd109	7.14E-05	Pa231	0.00E+00	Ta182	4.72E-05
Cd113m	0.00E+00	Pa233	7.67E-02	Tc99	5.23E-01
Ce139	2.12E-06	Pa234	7.35E-03	Te125m	3.52E-02
Ce144	5.62E+00	Pa234m	4.69E+00	Th228	3.25E-02
Cf249	7.28E-05	Pb209	8.72E-04	Th229	8.72E-04
Cf250	4.80E-03	Pb210	0.00E+00	Th230	3.10E-04
Cf251	2.98E-04	Pb212	3.26E-02	Th231	5.07E-01
Cf252	2.90E-02	Pb214	3.16E-06	Th232	1.75E-03
Cl36	3.98E-03	Pd107	0.00E+00	Th234	5.34E+00
Cm242	2.82E-04	Pm147	4.57E+01	Tl204	7.80E-07
Cm243	2.79E-04	Po210	7.55E-15	Tl208	2.10E-03
Cm244	5.54E+00	Po212	3.73E-03	Tl209	3.52E-05
Cm245	1.14E-02	Po213	8.37E-04	U232	3.92E-02
Cm246	2.03E-02	Po216	3.25E-02	U233	2.10E+00
Cm247	4.15E-11	Pr144	5.22E+00	U234	1.75E+01
Cm248	1.28E-11	Pr144m	5.21E-03	U235	5.87E-01
Co57	4.22E-04	Pu238	2.01E+01	U236	2.96E-01
Co58	2.50E+00	Pu239	8.65E+00	U238	4.76E+00
Co60	1.70E+01	Pu240	2.36E+00	W181	0.00E+00
Cr51	1.38E-02	Pu241	7.97E+01	W185	0.00E+00
Cs134	1.28E+01	Pu242	1.44E-02	W188	0.00E+00
Cs135	2.28E-09	Pu244	2.23E-15	Y90	1.17E+03
Cs137	3.48E+02	Ra224	2.52E-02	Zn65	3.94E-01
Eu152	1.17E+01	Ra225	8.72E-04	Zr93	2.84E-05
Eu154	1.01E+01	Ra226	1.25E-06	Zr95	6.71E-02
Eu155	6.91E-02	Ra228	9.08E-03		
Fe55	3.32E+01	Rb87	0.00E+00		

[1] Total E-Area LLWF LAW Vault closure inventory for one LAW Vault.
[2] Special waste form considered in the E-Area PA.

Table A-17. 2025 Radionuclide Inventory for 643-7E NRCDA

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Am241	3.52E-01	Fe55	9.03E+04	S35	3.09E-03
Am242m	8.03E-06	Fe59	7.48E+03	Sb125	4.07E+04
Am243	2.41E-03	H3	1.34E+02	Sc46	3.26E-03
Ba137m	5.28E+00	Hf181	7.46E+03	Se79	1.22E-03
C14	1.39E+02	I129	1.48E-05 [2]	Sm151	5.38E-02
Ca45	1.34E-04	In113m	4.87E+03	Sn113	4.87E+03
Ce144	5.14E+00	Kr85	5.71E-03	Sn119m	8.08E+04
Cf249	1.25E-10	Mn54	1.39E+03	Sn123	2.35E+03
Cf251	2.70E-12	Mo93	1.43E+00	Sn126	8.59E-06
Cl36	1.80E-05	Nb93m	7.46E+03	Sr90	5.39E+00
Cm242	5.22E+00	Nb94	6.54E+00	Ta182	1.76E+04
Cm243	7.90E-06	Nb95	1.31E+05	Tc99	1.46E-01
Cm244	1.92E-01	Nb95m	1.31E+03	Te125m	2.54E+04
Cm245	1.02E-05	Ni59	1.55E+03	Th232	3.02E-10
Cm246	3.95E-06	Ni63	1.80E+05	U232	4.77E-06
Cm247	7.96E-12	Np237	4.03E-06	U233	7.83E-07
Cm248	1.89E-11	Pm147	3.05E+00	U234	3.64E-06
Co58	2.07E+04	Pr144	2.20E-01	U235	2.06E-07
Co60	9.85E+04	Pu238	2.69E-01	U236	4.21E-06
Cr51	7.47E+03	Pu239	1.23E-01	U238	2.32E-05
Cs134	5.33E-02	Pu240	1.11E-01	Y90	5.39E+00
Cs135	3.45E-05	Pu241	3.40E+01	Zn65	1.13E+01
Cs137	5.29E+00	Pu242	4.07E-04	Zr93	7.46E+03
Eu154	6.72E-02	Pu244	2.77E-11	Zr95	6.16E+04
Eu155	3.83E-02	Ru106	6.60E-01		

[1] Total E-Area LLWF NRCDA closure inventory for the 643-7E NRCDA.

[2] Value corrected from Hiergesell et al. (2008), and is now consistent with Sink (2007). Per Table 11-2, item 14 in Volume I, Hiergesell et al. (2008) will be revised.

Table A-18. 2025 Radionuclide Inventory for 643-26E NRCDA

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Am241	8.8E-01	Fe55	2.2E+05	Ru106	1.1E+00
Am242m	7.9E-06	Fe59	1.9E+04	Sb125	1.0E+05
Am243	6.0E-03	H3	3.4E+02	Se79	3.1E-03
Ba137m	1.3E+01	Hf181	1.9E+04	Sm151	1.4E-01
C14	3.4E+02	I129	9.4E-06	Sn113	1.2E+04
Ce144	1.2E+01	In113m	1.2E+04	Sn119m	2.0E+05
Cf249	3.1E-10	Mn54	3.4E+03	Sn123	5.9E+03
Cf251	6.6E-12	Mo93	3.6E+00	Sn126	1.9E-03
Cm242	1.3E+01	Nb93m	1.9E+04	Sr90	1.3E+01
Cm243	4.6E-06	Nb94	1.6E+01	Ta182	4.4E+04
Cm244	4.8E-01	Nb95	3.3E+05	Tc99	3.6E-01
Cm245	2.6E-05	Nb95m	3.3E+03	Te125m	6.4E+04
Cm246	9.8E-06	Ni59	3.9E+03	Th232	1.7E-10
Cm247	2.0E-11	Ni63	4.5E+05	U232	2.8E-06
Cm248	4.7E-11	Np237	1.0E-05	U234	7.0E-04
Co58	5.1E+04	Pm147	7.5E+00	U235	5.2E-07
Co60	2.5E+05	Pu238	6.7E-01	U236	1.1E-03
Cr51	1.9E+04	Pu239	3.1E-01	U238	5.8E-05
Cs135	8.7E-05	Pu240	2.8E-01	Y90	1.3E+01
Cs137	1.3E+01	Pu241	8.5E+01	Zr93	1.9E+04
Eu154	1.7E-01	Pu242	1.0E-03	Zr95	1.5E+05
Eu155	9.6E-02	Pu244	6.9E-11		

Table A-19. 2025 Radionuclide Inventory for Slit Trench East Grouping

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Ac225	1.7E-03	Ni63	5.0E+01
Ac227	7.7E-03	Np237	6.2E-02
Ac228	8.1E-02	Np239	1.2E+02
Ag108	2.1E-09	Pa231	7.7E-03
Ag108m	2.4E-08	Pa233	3.3E-03
Ag109m	8.0E-02	Pa234	1.1E-01
Ag110	1.5E-11	Pa234m	9.1E+01
Ag110m	1.1E-09	Pb209	1.7E-03
Al26	7.7E-03	Pb210	3.1E-01
Am241	4.1E+00	Pb212	8.1E-02
Am242	1.8E-10	Pb214	3.1E-01
Am242m	2.7E+00	Pd107	8.4E-07
Am243	3.3E-01	Pm147	1.5E+01
Ar39	7.7E-03	Po210	3.1E-01
At217	1.7E-03	Po212	1.1E-02
Ba133	6.4E-05	Po213	1.6E-03
Ba137m	1.6E+02	Po214	2.7E-03
Bi207	7.7E-03	Po216	8.1E-02
Bi210	3.1E-01	Po218	3.1E-01
Bi211	7.0E-04	Pr144	2.7E+00
Bi212	8.1E-02	Pr144m	1.2E-02
Bi213	1.7E-03	Pu238	1.9E+02
Bi214	3.1E-01	Pu239	1.3E+01
Bk249	7.7E-03	Pu240	3.5E+00
C14	2.3E-01	Pu241	6.4E+01
C14_NR.Pump [2]	6.3E-01	Pu242	1.2E-01
Ca41	7.7E-03	Pu244	3.9E-14
Cd109	8.0E-02	Ra224	8.1E-02
Cd113m	1.7E-08	Ra225	1.7E-03
Ce141	1.2E-07	Ra226	2.4E-02
Ce144	2.8E+00	Ra228	8.1E-02
Cf249	6.4E-01	Rb87	6.6E-13
Cf250	2.8E-01	Rh103m	4.0E-05
Cf251	5.8E-01	Rh106	2.7E+00
Cf252	3.9E-02	Rn220	8.1E-02
Cl36	8.1E-05	Rn222	3.1E-01
Cm242	1.3E-03	Ru103	4.0E-05
Cm243	8.0E-03	Ru106	2.7E+00
Cm244	2.5E+01	S35	2.3E-02
Cm245	8.3E-03	Sb124	7.7E-03
Cm246	3.1E-03	Sb125	6.0E+00
Cm247	8.2E-03	Sb126	1.3E-06

Table A-19. 2025 Radionuclide Inventory for Slit Trench East Grouping - continued

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Cm248	4.7E-04	Sb126m	2.4E-04
Co57	1.6E-09	Sc46	7.7E-03
Co58	3.9E+01	Se75	7.7E-03
Co60	9.7E+05	Se79	6.6E-02
Cr51	9.4E-02	Sm151	9.3E-01
Cs134	1.5E+00	Sn113	5.0E-05
Cs135	5.4E-07	Sn119m	1.2E-11
Cs137	2.1E+02	Sn121	3.1E-14
Eu152	5.4E-01	Sn121m	3.9E-14
Eu154	1.7E+02	Sn123	7.7E-03
Eu155	7.1E+00	Sn126	1.4E-03
Fe55	2.6E+02	Sr89	7.7E-03
Fe59	2.8E-01	Sr90	9.4E+01
Fr221	1.7E-03	Ta182	3.9E-02
H3	9.2E+00	Tc99	5.4E-02
H3_ETF_Carbon [2]	0.0E+00	Te125m	1.5E-01
Hf181	8.9E-02	Th228	8.1E-02
I129	3.8E-04	Th229	1.7E-03
I129_ETF_Carbon [2]	0.0E+00	Th230	3.0E-03
I129_ETF_GT73 [2]	6.6E-04	Th231	2.7E-01
I129_F_Area_Carbon [2]	0.0E+00	Th232	8.1E-02
I129_F_Area_CG8 [2]	0.0E+00	Th234	9.1E+01
I129_F_Dowex_21K [2]	0.0E+00	Tl208	6.6E-03
I129_F_Area_Filtercake [2]	0.0E+00	Tl209	6.7E-05
I129_H_Area_Carbon [2]	0.0E+00	U232	2.9E-01
I129_H_Area_CG8 [2]	0.0E+00	U233	1.6E+01
I129_H_Dowex_21K [2]	0.0E+00	U234	1.4E+01
I129_H_Area_Filtercake [2]	0.0E+00	U235	3.0E-01
In113m	5.0E-05	U236	2.0E-01
K40	7.7E-04	U238	2.5E+01
Kr85	2.1E-01	W181	4.5E-07
Mn54	7.6E+00	W185	7.7E-03
Mo93	7.8E-03	W188	7.7E-03
Na22	6.1E-06	Y90	1.5E+02
Nb93m	1.2E+00	Y91	4.5E-62
Nb94	3.8E-03	Zn65	3.7E-01
Nb95	7.0E-01	Zr93	2.1E-04
Nb95m	2.8E-05	Zr95	3.2E-01
Ni59	2.8E-01		

[1] Total E-Area LLWF Slit East closure inventory for an equivalent of 7.66 standard Slit Trench footprints.

[2] Special waste form considered in the E-Area PA.

Table A-20. 2025 Radionuclide Inventory for Slit Trench Central Grouping

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Ac225	1.55E-03	Pa231	7.00E-03
Ac227	7.00E-03	Pa233	2.98E-03
Ac228	7.42E-02	Pa234	9.87E-02
Ag108	1.94E-09	Pa234m	8.33E+01
Ag108m	2.23E-08	Pb209	1.55E-03
Ag109m	7.28E-02	Pb210	2.79E-01
Ag110	1.38E-11	Pb212	7.42E-02
Ag110m	1.02E-09	Pb214	2.79E-01
Al26	7.00E-03	Pd107	7.70E-07
Am241	3.72E+00	Pm147	1.38E+01
Am242	1.61E-10	Po210	2.79E-01
Am242m	2.47E+00	Po212	1.05E-02
Am243	3.02E-01	Po213	1.50E-03
Ar39	7.00E-03	Po214	2.49E-03
At217	1.55E-03	Po216	7.42E-02
Ba133	5.80E-05	Po218	2.79E-01
Ba137m	1.42E+02	Pr144	2.49E+00
Bi207	7.00E-03	Pr144m	1.06E-02
Bi210	2.79E-01	Pu238	1.72E+02
Bi211	6.40E-04	Pu239	1.15E+01
Bi212	7.42E-02	Pu240	3.16E+00
Bi213	1.55E-03	Pu241	5.85E+01
Bi214	2.79E-01	Pu242	1.06E-01
Bk249	7.00E-03	Pu244	3.57E-14
C14	7.00E-02	Ra224	7.42E-02
C14_NR.Pump [2]	5.77E-01	Ra225	1.55E-03
Ca41	7.00E-03	Ra226	2.23E-02
Cd109	7.28E-02	Ra226_Cooling_Tower [2]	2.77E-01
Cd113m	1.57E-08	Ra228	7.42E-02
Ce141	1.11E-07	Rb87	6.01E-13
Ce144	2.52E+00	Rh103m	3.65E-05
Cf249	5.80E-01	Rh106	2.46E+00
Cf250	2.53E-01	Rn220	7.42E-02
Cf251	5.29E-01	Rn222	2.79E-01
Cf252	3.55E-02	Ru103	3.65E-05
Cl36	7.42E-05	Ru106	2.47E+00
Cm242	1.16E-03	S35	2.07E-02
Cm243	7.28E-03	Sb124	7.00E-03
Cm244	2.24E+01	Sb125	5.51E+00
Cm245	7.56E-03	Sb126	1.23E-06
Cm246	2.86E-03	Sb126m	2.19E-04

Table A-20. 2025 Radionuclide Inventory for Slit Trench Central Grouping - continued

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Cm247	7.49E-03	Sc46	7.00E-03
Cm248	4.26E-04	Se75	7.00E-03
Co57	1.44E-09	Se79	6.01E-02
Co58	3.53E+01	Sm151	8.47E-01
Co60	8.89E+05	Sn113	4.54E-05
Cr51	8.61E-02	Sn119m	1.13E-11
Cs134	1.33E+00	Sn121	2.79E-14
Cs135	4.96E-07	Sn121m	3.58E-14
Cs137	1.96E+02	Sn123	7.00E-03
Eu152	4.95E-01	Sn126	1.27E-03
Eu154	1.57E+02	Sr85	2.07E-03
Eu155	6.51E+00	Sr90	8.61E+01
Fe55	2.36E+02	Sr90_MK50A [2]	5.19E+01
Fe59	2.52E-01	Ta182	3.57E-02
Fr221	1.55E-03	Tc99	3.50E-01
H3	7.00E+00	Tc99_MK50A [2]	1.25E-02
H3_ETF_Carbon [2]	1.94E+00	Te125m	1.40E-01
H3_Concrete [2]	2.71E+01	Th228	7.42E-02
Hf181	8.12E-02	Th229	1.55E-03
I129	2.10E-04	Th230	2.75E-03
I129_MK50A [2]	5.73E-05	Th230_Cooling_Tower [2]	2.77E-01
I129_H_Area_Carbon [2]	0.00E+00	Th231	2.46E-01
I129_F_Area_Carbon [2]	0.00E+00	Th232	7.42E-02
I129_ETF_Carbon [2]	1.15E-01	Th234	8.33E+01
I129_F_Dowex_21K [2]	3.09E-02	Tl208	6.00E-03
I129_H_Dowex_21K [2]	0.00E+00	Tl209	6.10E-05
I129_H_Area_Filtercake [2]	1.94E-06	U232	2.63E-01
I129_F_Area_CG8 [2]	3.61E-04	U233	1.42E+01
I129_H_Area_CG8 [2]	8.26E-04	U234	1.27E+01
I129_ETF_GT73 [2]	6.05E-04	U234_M_Glass [2]	1.96E+01
I129_F_Area_Filtercake [2]	4.90E-04	U235	2.75E-01
In113m	4.54E-05	U235_Paducah_Cask [2]	2.74E+00
K40	7.00E-04	U235_M_Glass [2]	1.31E+00
Kr85	1.93E-01	U236	1.80E-01
Mn54	6.99E+00	U236_M_Glass [2]	9.94E-01
Mo93	7.14E-03	U238	2.30E+01
Na22	5.54E-06	U238_M_Glass [2]	7.35E+01
Nb93m	1.14E+00	W181	4.07E-07
Nb94	3.50E-03	W185	7.00E-03
Nb95	6.40E-01	W188	7.00E-03
Nb95m	2.58E-05	Y90	1.38E+02

Table A-20. 2025 Radionuclide Inventory for Slit Trench Central Grouping - continued

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Ni59	2.56E-01	Zn65	3.35E-01
Ni63	4.56E+01	Zr93	1.90E-04
Np237	5.62E-02	Zr95	2.96E-01
Np239	1.13E+02		

[1] Total E-Area LLWF Slit Central closure inventory for 7 standard Slit Trench footprints.
[2] Special waste form considered in the E-Area PA.

Table A-21. 2025 Radionuclide Inventory for Slit Trench West Grouping

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Ac225	1.09E-03	Np237	2.27E-03
Ac227	4.93E-03	Np239	4.26E+01
Ac228	4.93E-02	Pa231	4.93E-03
Ag108	4.93E-10	Pa233	5.82E-04
Ag108m	4.93E-09	Pa234	4.93E-02
Ag109m	4.93E-02	Pa234m	1.38E+00
Ag110	4.93E-12	Pb209	9.86E-04
Ag110m	4.93E-10	Pb210	4.93E-04
Al26	4.93E-03	Pb212	4.93E-02
Am241	2.37E+00	Pb214	1.97E-01
Am242	4.93E-11	Pd107	4.93E-07
Am242m	8.82E+00	Pm147	8.04E-03
Am243	1.04E+00	Po210	1.97E-01
Ar39	4.93E-03	Po212	4.93E-03
At217	9.86E-04	Po213	9.86E-04
Ba133	4.93E-06	Po214	1.48E-03
Ba137m	5.42E+00	Po216	4.93E-02
Bi207	4.93E-03	Po218	1.48E-01
Bi210	1.97E-01	Pr144	1.48E+00
Bi211	4.93E-05	Pr144m	4.93E-03
Bi212	4.93E-02	Pu238	4.90E+01
Bi213	4.93E-04	Pu239	1.53E+01
Bk249	4.93E-03	Pu240	4.58E+00
C14	9.06E-02	Pu241	4.65E+01
C14_NR_Pump [2]	4.93E-03	Pu242	2.92E+00
Ca41	4.93E-03	Pu244	4.93E-15
Cd109	4.93E-02	Ra224	4.93E-02
Cd113m	4.93E-13	Ra225	9.86E-04
Ce141	4.93E-08	Ra226	4.93E-04
Ce144	1.97E+00	Ra228	4.93E-05

Table A-21. 2025 Radionuclide Inventory for Slit Trench West Grouping - continued

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Cf249	2.08E+00	Rb87	4.93E-14
Cf250	2.00E-01	Rh103m	2.47E-05
Cf251	1.89E+00	Rh106	1.48E+00
Cf252	2.80E-02	Rn220	4.93E-02
Cl36	4.93E-06	Rn222	4.93E-02
Cm242	4.93E-04	Ru103	4.93E-06
Cm243	3.87E-02	Ru106	4.93E-01
Cm244	7.40E+01	S35	4.93E-03
Cm245	3.58E-02	Sb124	4.93E-02
Cm246	4.73E-05	Sb125	1.44E+00
Cm247	7.84E-02	Sb126	4.93E-07
Cm248	1.47E-04	Sb126m	4.93E-05
Co57	4.93E-10	Sc46	4.93E-03
Co58	4.93E+00	Se75	4.93E-03
Co60	8.53E-02	Se79	2.05E-04
Cr51	4.93E-02	Sm151	1.18E-06
Cs134	4.93E-02	Sn113	4.93E-06
Cs135	4.93E-14	Sn119m	4.93E-13
Cs137	2.73E+01	Sn121	4.93E-15
Eu152	5.57E-05	Sn121m	4.93E-15
Eu154	5.97E+01	Sn123	4.93E-03
Eu155	5.08E+00	Sn126	2.96E-07
Fe55	1.48E+02	Sr85	4.93E-04
Fe59	1.97E-01	Sr90	2.02E+01
Fr221	1.48E-03	Ta182	4.93E-03
H3	2.71E+00	Tc99	2.65E-02
I129	6.41E-04	Te125m	1.44E-02
I129_H_Area_Carbon [2]	0.00E+00	Th228	4.93E-03
I129_F_Area_Carbon [2]	0.00E+00	Th229	4.93E-04
I129_ETF_Carbon [2]	0.00E+00	Th230	4.93E-04
I129_F_Dowex_21K [2]	0.00E+00	Th231	4.77E-04
I129_H_Dowex_21K [2]	0.00E+00	Th232	2.79E-07
I129_H_Area_Filtercake [2]	0.00E+00	Th234	1.38E+00
I129_F_Area_CG8 [2]	0.00E+00	Tl208	4.93E-04
I129_H_Area_CG8 [2]	0.00E+00	Tl209	4.93E-06
I129_ETF_GT73 [2]	0.00E+00	U232	4.93E-03
I129_F_Area_Filtercake [2]	0.00E+00	U233	4.10E+00
In113m	2.96E-05	U234	3.04E+00
K40	4.93E-04	U235	1.24E-03
Kr85	4.93E-02	U236	8.58E-06
Mn54	4.88E+00	U238	1.42E+00
Mo93	4.93E-03	W181	4.93E-03

Table A-21. 2025 Radionuclide Inventory for Slit Trench West Grouping - continued

Radionuclide	Inventory [1] (Ci)	Radionuclide	Inventory [1] (Ci)
Na22	4.93E-07	W185	4.93E-03
Nb93m	4.93E-02	W188	4.93E-03
Nb94	4.93E-04	Y90	2.01E+01
Nb95	4.44E-01	Zn65	4.93E-02
Nb95m	1.48E-05	Zr93	4.93E-06
Ni59	2.53E-04	Zr95	4.93E-02
Ni63	2.51E-07		

[1] Total E-Area LLWF Slit West closure inventory for an equivalent of 4.93 standard Slit Trench footprints.

[2] Special waste form considered in the E-Area PA.

Table A-22. 1972 Radionuclide Inventory for TRU Pad 1

Radionuclide	Inventory (Ci)
Pu238	2.35E+05
Pu239	1.64E+02
Pu240	1.20E+02
Pu241	1.70E+04
Pu242	1.37E-01

Table A-23. 2002 Radionuclide Inventory for the MWMF and LLRWDF

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Am241	1.98E+01	Pu239	6.57E+01
Am243	9.94E-04	Pu240	1.53E+01
C14	3.71E+03	Pu241	3.21E+02
Cf252	7.49E+00	Pu242	1.25E-03
Cm244	1.37E+04	Sb125	6.17E+01
Co60	2.16E+06	Se79	1.13E-01
Cs134	3.92E+01	Sm151	2.97E+02
Cs137	2.00E+04	Sn126	1.55E-01
Eu154	4.39E+02	Sr90	2.00E+04
Eu155	6.81E+00	Tc99	1.40E+01
H3	2.39E+06	Te125m	1.10E-10
I129	1.00E+01	Th232	3.92E+00
Mn54	9.60E-03	U233	2.04E+00
Ni59	1.82E+03	U234	5.05E+01
Ni63	2.25E+05	U235	1.56E+00
Np237	1.00E-01	U236	5.88E+00
Pu238	4.09E+03	U238	4.62E+01

Table A-24. 2002 Radionuclide Inventory for ORWBG/GSACU

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Ac228	8.18E-02	Pm147	6.19E-02
Am241	2.20E+02	Pu238	2.05E+04
Bi214	3.02E-02	Pu239	2.00E+03
C14	4.00E+03	Pu240	3.33E+02
Cf252	2.91E-03	Pu241	2.78E+03
Cm242	1.29E-03	Ra226	7.57E-02
Cm243	1.06E+01	Ra228	1.78E-01
Cm244	8.13E+03	Sb125	1.77E+00
Cm245	8.41E-04	Se79	7.21E-01
Cm246	8.41E-04	Sm151	1.67E+03
Co60	3.19E+04	Sn126	9.88E-01
Cs134	6.50E-01	Sr90	5.37E+04
Cs137	7.81E+04	Tc99	2.61E+01
Eu152	3.18E-04	Te125m	6.88E-55
Eu154	7.29E+02	Th228	1.29E-01
Eu155	3.72E+00	Th230	5.28E-02
H3	4.91E+05	Th232	3.71E+00
I129	1.10E+01	Th234	6.45E-02
K40	1.19E-01	U233	5.27E-01
Mn54	7.64E-04	U234	2.01E+01
Na22	5.03E-04	U235	1.25E+00
Ni59	3.71E+03	U236	2.85E+00
Ni63	4.11E+05	U238	1.69E+01
Np237	2.01E+00	Y88	3.36E-05
Pb212	9.88E-02		

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A.4 F-AREA FACILITIES

Table A-25. 2003 Radionuclide Inventory for FAMS

Radionuclide	Inventory (Ci)
Am241	7.0E-03
Np237	9.1E-04
Pu238	1.0E+02
Pu239	6.2E-02
Pu240	3.3E-02
Pu241	3.4E+00
Pu242	4.5E-05
Th232	4.1E-10

Table A-26. 2002 Radionuclide Inventory for the F-Area Inactive Process Sewer Lines

Radionuclide	Inventory (Ci)
H3	4.16E+03
Cs137	1.07E+02
Sr90	5.22E+01
Am241	3.00E-02
Cm244	5.72E-02
Eu152	3.81E-02
Eu155	7.63E-03
Np237	2.15E-02
Pu238	2.72E-01
Pu239	7.57E+00
Ra226	8.72E-02
Ru106	2.21E-01
Tc99	2.21E-01
Th232	6.59E-02
U234	4.90E-01
U238	5.58E+00

Table A-27. 2007 F-Canyon Complex Residual Inventory

Radionuclide	Inventory (Ci)
Am241	2.2E+04
Ba137m	1.3E+05
C14	2.3E+00
Ce144	8.5E+04
Co60	9.2E+02
Cs137	1.4E+05
H3	5.5E+02
I129	4.2E-02
Ni59	2.7E+00
Np237	4.3E-01
Pm147	6.6E+04
Pr144	8.5E+04
Pr144m	8.5E+04
Pu236	0.0E+00
Pu238	4.4E+02
Pu239	3.7E+03
Pu240	8.6E+02
Pu241	9.9E+03
Pu242	1.3E-01
Rh106	4.8E+03
Ru106	2.9E+04
Se79	6.2E-01
Sn126	7.4E-02
Sr90	3.6E+03
Tc99	2.3E+01
U232	0.0E+00
U233	4.0E-07
U234	7.1E-04
U235	1.1E-02
U236	1.2E-02
U238	1.0E+00
Y90	3.6E+03

Table A-28. 2007 F Canyon Process Vessels Residual Inventory

Radionuclide	Inventory (Ci)
Am241	5.7E+00
Pu238	4.0E+02
Pu239	2.4E+01
Pu240	7.8E+00
Pu241	2.3E+02
Pu242	4.6E-03

Table A-29. 2007 FB Line Residual Inventory

Radionuclide	Inventory (Ci)
Am241	4.4E+03
Pu236	1.1E+01
Pu238	1.5E+04
Pu239	1.2E+03
Pu240	1.2E+03
Pu241	6.6E+04
U232	4.5E-04

Table A-30. 2006 211-F Residual Inventory

Radionuclide	Waste Handling Vault (Ci)	Sludge (Tanks 804, 808, & 809) (Ci)	Segregated Solvent Facility (Ci)	Acid Recovery Unit (Ci)	Water Handling Facility (Ci)	GP Tanks (Ci)	GP Evaporators (Ci)	Total Inventory (Ci)
Ac228	0.00E+00	0.00E+00	6.20E-05	3.51E-05	8.04E-05	7.26E-05	2.53E-05	2.75E-04
Am241	7.55E-01	5.16E+00	3.16E-03	2.56E-05	7.26E-05	7.26E-05	8.62E-06	5.92E+00
Am243	0.00E+00	1.87E-03	3.58E-04	0.00E+00	0.00E+00	0.00E+00	9.07E-06	2.24E-03
Bi212	0.00E+00	0.00E+00	5.23E-05	2.74E-05	6.24E-05	5.41E-05	2.22E-05	2.18E-04
Bi214	0.00E+00	0.00E+00	4.03E-05	2.87E-05	6.29E-05	5.26E-05	2.59E-05	2.10E-04
C14	6.03E-03	1.21E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.24E-03
Cm242	0.00E+00	5.25E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.25E-04
Cm243	2.01E-01	2.17E-01	0.00E+00	0.00E+00	1.45E-05	3.90E-05	7.60E-06	4.18E-01
Cm244	2.01E-01	2.17E-01	0.00E+00	0.00E+00	1.45E-05	3.90E-05	7.60E-06	4.18E-01
Co60	9.86E-03	1.85E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.95E-01
Cs134	1.69E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.04E-06	9.86E-07	1.69E-02
Cs137	9.01E+00	8.67E-01	7.55E-04	2.43E-05	7.31E-05	3.39E-04	8.60E-04	9.88E+00
Eu154	5.42E-03	4.20E-02	7.22E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.75E-02
Eu155	1.77E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.77E-03
H3	0.00E+00	0.00E+00	3.51E-02	0.00E+00	0.00E+00	7.00E-04	4.23E-04	3.63E-02
I129	4.57E-04	8.41E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.30E-03
K40	3.52E-04	0.00E+00	5.35E-04	3.88E-04	8.67E-04	7.50E-04	2.73E-04	3.17E-03
M-54	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.63E-06	0.00E+00	0.00E+00	2.63E-06
Ni59	5.39E-01	5.34E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.45E-01
Ni63	5.42E-02	7.74E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.28E-01
Np237	5.27E-03	3.01E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.53E-02
Pb212	0.00E+00	0.00E+00	6.10E-05	3.54E-05	7.50E-05	6.87E-05	3.36E-05	2.74E-04
Pu238	1.76E+02	1.55E+02	0.00E+00	9.87E-06	6.29E-05	2.68E-05	0.00E+00	3.30E+02

Table A-30. 2006 211-F Residual Inventory -continued

Radionuclide	Waste Handling Vault (Ci)	Sludge (Tanks 804, 808, & 809) (Ci)	Segregated Solvent Facility (Ci)	Acid Recovery Unit (Ci)	Water Handling Facility (Ci)	GP Tanks (Ci)	GP Evaporators (Ci)	Total Inventory (Ci)
Pu239	1.04E+01	1.02E+02	2.44E-02	8.86E-05	5.36E-04	1.83E-04	1.33E-05	1.12E+02
Pu240	1.04E+01	1.02E+02	2.44E-02	8.86E-05	5.36E-04	1.83E-04	1.33E-05	1.12E+02
Pu241	2.39E+01	3.57E+02	0.00E+00	3.37E-04	1.00E-03	0.00E+00	0.00E+00	3.81E+02
Pu242	7.58E-03	9.92E-03	0.00E+00	3.29E-04	0.00E+00	5.80E-04	0.00E+00	1.84E-02
Ra226	4.09E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.09E-01
Sb125	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.92E-06	0.00E+00	0.00E+00	8.92E-06
Sb126	0.00E+00	5.36E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.36E-03
Se79	0.00E+00	0.00E+00	5.68E-04	1.24E-04	0.00E+00	0.00E+00	0.00E+00	6.93E-04
Sn126	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.32E-06	2.32E-06
Sr90	3.87E+00	8.03E+01	4.39E-04	2.98E-05	0.00E+00	2.79E-04	7.42E-05	8.42E+01
Th232	2.94E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.94E-06
Tl208	0.00E+00	0.00E+00	1.85E-05	1.05E-05	2.36E-05	2.37E-05	1.05E-05	8.67E-05
U233	1.30E-02	1.54E-02	5.71E-05	3.57E-05	7.65E-05	1.24E-04	3.55E-05	2.87E-02
U234	1.30E-02	1.54E-02	5.71E-05	3.57E-05	7.65E-05	1.24E-04	3.55E-05	2.87E-02
U235	6.25E-05	2.77E-04	1.80E-05	8.10E-06	1.10E-05	2.06E-05	3.42E-06	4.00E-04
U236	6.25E-05	2.77E-04	1.80E-05	8.10E-06	1.10E-05	2.06E-05	3.42E-06	4.00E-04
U238	1.80E-02	7.04E-04	1.22E-04	4.98E-05	7.80E-05	1.23E-04	3.59E-05	1.91E-02

Table A-31. 2002 Radionuclide Inventory for F-Area Retention Basin 281-3F

Radionuclide	Inventory (Ci)
Am241	9.50E-02
Ce144	1.03E-03
Co60	3.98E-03
Cs137	3.94E+01
Eu154	1.75E-02
Eu155	1.96E-02
H3	3.89E-01
I129	1.99E-01
Pm147	8.57E-02
Pu238	1.62E+00
Pu239	1.06E+00
Pu240	2.52E-01
Ra226	5.20E-02
Sr90	1.96E+01
Tc99	3.05E-02
Th234	2.17E-01
U233	3.37E-03
U234	2.87E-01
U235	1.92E+01
U238	3.84E-01

Table A-32. 2002 Radionuclide Inventory for F-Area Seepage Basins

Radionuclide	Inventory (Ci)
Am241	3.29E-02
Cs137	3.43E+02
H3	3.45E+05
I129	3.57E-02
Pu238	2.49E+00
Pu239	9.81E+00
Sr90	4.10E+01
Tc99	8.80E-02
U234	6.34E+00
U235	2.23E+00
U238	1.11E+01

Table A-33. 2020 FTF Estimated Radiological Inventory (Ci) at Closure

Tank	Ac227	Al26	Am241	Am242m	Am243	Ba137m	Bk249	C14	Ce144	Cf249	Cm242	Cm243	Cm244
1	1.93E-07	8.32E-04	4.66E+01	5.44E-02	1.57E-04	5.88E+02	3.06E-36	1.42E-03	1.27E-16	3.35E-19	3.87E-36	1.15E-03	4.68E-03
2	1.54E-08	1.97E-04	1.23E+01	1.41E-02	4.16E-05	1.40E+02	1.27E-39	1.25E-03	2.03E-20	8.85E-20	6.38E-43	2.72E-04	1.03E-03
3	3.84E-08	1.73E-04	1.31E+01	1.19E-02	3.49E-05	1.23E+02	1.35E-38	1.23E-03	3.33E-19	7.43E-20	2.61E-40	2.39E-04	9.31E-04
4	4.70E-08	2.80E-04	1.03E-01	1.51E-02	8.65E-02	1.95E+02	3.59E-32	1.13E-03	5.61E-12	9.02E-20	1.02E-27	3.87E-04	1.18E+01
5	2.63E-07	9.51E-04	4.91E+01	5.81E-02	1.65E-04	6.69E+02	4.63E-35	1.38E-03	2.56E-15	3.55E-19	3.85E-34	1.31E-03	5.61E-03
6	2.05E-07	1.24E-03	5.95E+01	7.07E-02	1.36E-01	8.68E+02	4.87E-34	1.13E-03	3.59E-14	4.26E-19	3.98E-32	1.71E-03	5.20E+00
7	1.54E-08	3.13E-05	1.78E+00	1.74E-03	1.83E-02	2.42E+02	3.54E-34	1.20E-03	5.54E-14	1.08E-20	2.91E-29	4.30E-05	1.64E+00
8	2.79E-08	7.81E-05	4.13E+00	4.24E-03	5.62E-02	5.45E+01	1.08E-33	1.21E-03	1.67E-13	2.58E-20	5.25E-29	1.08E-04	4.20E+00
17	NE	NE	8.43E+00	NE	NE	1.04E+01	NE	3.11E-03	1.58E-13	NE	NE	NE	3.00E-04
18	2.45E-07	4.28E-03	1.86E+01	1.87E-01	1.55E-06	2.01E+03	1.61E-34	1.41E-02	1.67E-13	3.40E-21	6.56E-32	1.63E-05	2.28E+01
19	4.10E-08	1.71E-02	2.90E+00	4.52E-01	4.45E-08	7.83E+03	1.41E-36	2.83E-02	1.11E-13	9.74E-23	1.71E-34	4.49E-07	3.91E-04
20	NE	NE	1.64E+00	NE	NE	2.30E+01	NE	6.58E-04	2.88E-14	NE	NE	NE	NE
25	5.54E-09	5.50E-06	3.80E-01	4.40E-02	5.56E-07	4.10E+02	1.59E-29	1.73E-03	1.67E-08	1.25E-21	4.76E-20	7.61E-06	4.37E-05
26	9.74E-09	9.68E-06	6.69E-01	4.40E-02	9.79E-07	6.60E+00	2.80E-29	1.93E-03	2.94E-08	2.20E-21	8.37E-20	1.34E-05	7.70E-05
27	4.56E-09	4.53E-06	3.13E-01	4.40E-02	4.58E-07	4.87E+02	1.31E-29	1.68E-03	1.38E-08	1.03E-21	3.92E-20	6.27E-06	3.60E-05
28	9.74E-09	9.68E-06	6.69E-01	4.40E-02	9.79E-07	6.60E+00	2.80E-29	1.93E-03	2.94E-08	2.20E-21	8.37E-20	1.34E-05	7.70E-05
33	2.47E-07	1.32E-03	6.30E+01	5.02E-02	1.28E-04	9.00E+02	4.23E-27	1.79E-03	3.86E-06	2.89E-19	3.81E-18	1.82E-03	1.09E-02
34	5.63E-07	5.34E-03	1.59E+03	2.18E-01	5.61E-04	3.65E+03	8.96E-28	1.46E-03	3.59E-07	1.26E-18	4.61E-20	3.88E-02	4.18E-02
44	9.74E-09	9.68E-06	6.69E-01	4.40E-02	9.79E-07	6.60E+00	2.80E-29	1.93E-03	2.94E-08	2.20E-21	8.37E-20	1.34E-05	7.70E-05
45	9.74E-09	9.68E-06	6.69E-01	4.40E-02	9.79E-07	6.60E+00	2.80E-29	1.93E-03	2.94E-08	2.20E-21	8.37E-20	1.34E-05	7.70E-05
46	9.74E-09	9.68E-06	6.69E-01	4.40E-02	9.79E-07	6.60E+00	2.80E-29	1.93E-03	2.94E-08	2.20E-21	8.37E-20	1.34E-05	7.70E-05
47	9.74E-09	9.68E-06	6.69E-01	4.40E-02	9.79E-07	6.60E+00	2.80E-29	1.93E-03	2.94E-08	2.20E-21	8.37E-20	1.34E-05	7.70E-05
Total	**1.97E-06**	**3.19E-02**	**1.88E+03**	**1.49E+00**	**2.98E-01**	**1.82E+04**	**5.32E-27**	**7.44E-02**	**4.43E-06**	**2.97E-18**	**4.45E-18**	**4.59E-02**	**4.57E+01**

NE = Not Estimated.

NOTE: All values, except Tanks 17 through 20 are decayed to 9/30/2020. Tanks 17 through 20 are decayed to 1/1/2020.

Table A-33. 2020 FTF Estimated Radiological Inventory (Ci) at Closure - continued

Tank	Cm245	Cm247	Cm248	Co60	Cs134	Cs135	Cs137	Eu152	Eu154	Eu155	H3	I129
1	1.02E-08	5.55E-17	1.28E-17	5.93E-01	1.44E-06	4.18E-03	6.21E+02	1.03E+00	6.20E+00	1.37E+00	2.72E-02	2.98E-05
2	2.73E-09	1.48E-17	3.40E-18	7.46E-02	3.73E-08	1.11E-03	1.48E+02	2.11E-01	1.09E+00	1.63E-01	2.72E-02	7.92E-06
3	2.28E-09	1.24E-17	2.85E-18	8.21E-02	7.09E-08	9.32E-04	1.30E+02	1.96E-01	1.07E+00	1.83E-01	2.72E-02	6.64E-06
4	9.18E-04	1.46E-17	3.38E-18	7.39E-01	3.15E-05	1.10E-03	2.06E+02	4.72E-01	3.90E+00	1.91E+00	2.72E-02	7.86E-06
5	1.08E-08	5.83E-17	1.34E-17	9.76E-01	4.83E-06	4.40E-03	7.07E+02	1.30E+00	8.53E+00	2.32E+00	2.72E-02	3.13E-05
6	1.28E-08	6.94E-17	1.60E-17	1.79E+00	1.48E-05	5.24E-03	9.18E+02	1.87E+00	1.34E+01	4.36E+00	2.72E-02	3.73E-05
7	4.51E-05	1.76E-18	4.05E-19	5.40E-02	1.01E-05	1.47E-03	2.56E+02	4.76E-02	3.57E-01	1.33E-01	2.72E-02	9.48E-07
8	1.38E-04	4.20E-18	9.68E-19	1.52E-01	2.70E-06	3.16E-04	5.76E+01	1.25E-01	9.63E-01	3.81E-01	2.72E-02	2.25E-06
17	4.39E-10	NE	NE	3.72E-02	2.82E-06	1.74E-04	1.10E+01	NE	2.43E-02	2.82E-03	6.09E+00	1.28E-06
18	5.48E-10	5.48E-19	1.27E-19	1.27E-01	3.57E-06	2.21E-04	2.24E+03	2.15E-02	7.34E-01	6.15E-02	6.15E-03	1.62E-06
19	5.01E-10	1.57E-20	3.62E-21	1.75E-01	4.62E-06	2.02E-04	8.30E+03	5.70E-04	9.11E-01	1.41E-03	3.51E-02	1.26E-05
20	NE	NE	NE	5.34E-03	9.41E-07	3.60E-05	2.43E+01	NE	1.66E-01	NE	NE	2.60E-07
25	3.66E-11	1.96E-19	4.55E-20	5.89E-02	2.93E-03	1.61E-03	4.33E+02	1.38E-02	1.63E-01	1.66E-01	3.52E-02	1.05E-07
26	6.44E-11	3.46E-19	8.01E-20	1.04E-01	4.72E-05	2.60E-05	6.98E+00	2.43E-02	2.87E-01	2.92E-01	3.52E-02	1.85E-07
27	3.01E-11	1.62E-19	3.75E-20	4.85E-02	3.48E-03	1.92E-03	5.14E+02	1.14E-02	1.34E-01	1.37E-01	3.52E-02	8.66E-08
28	6.44E-11	3.46E-19	8.01E-20	1.04E-01	4.72E-05	2.60E-05	6.98E+00	2.43E-02	2.87E-01	2.92E-01	3.52E-02	1.85E-07
33	8.35E-09	4.52E-17	1.04E-17	1.71E+01	1.06E-02	3.41E-03	9.51E+02	3.46E+00	4.32E+01	4.76E+01	3.52E-02	2.43E-05
34	3.66E-08	1.98E-16	4.58E-17	4.67E+01	1.20E-02	1.50E-02	3.86E+03	1.28E+01	1.44E+02	1.30E+02	3.52E-02	1.06E-04
44	6.44E-11	3.46E-19	8.01E-20	1.04E-01	4.72E-05	2.60E-05	6.98E+00	2.43E-02	2.87E-01	2.92E-01	3.52E-02	1.85E-07
45	6.44E-11	3.46E-19	8.01E-20	1.04E-01	4.72E-05	2.60E-05	6.98E+00	2.43E-02	2.87E-01	2.92E-01	3.52E-02	1.85E-07
46	6.44E-11	3.46E-19	8.01E-20	1.04E-01	4.72E-05	2.60E-05	6.98E+00	2.43E-02	2.87E-01	2.92E-01	3.52E-02	1.85E-07
47	6.44E-11	3.46E-19	8.01E-20	1.04E-01	4.72E-05	2.60E-05	6.98E+00	2.43E-02	2.87E-01	2.92E-01	3.52E-02	1.85E-07
Total	**1.10E-03**	**4.77E-16**	**1.10E-16**	**6.93E+01**	**2.94E-02**	**4.15E-02**	**1.94E+04**	**2.17E+01**	**2.27E+02**	**1.91E+02**	**6.70E+00**	**2.71E-04**

NE = Not Estimated.

NOTE: All values, except Tanks 17 through 20 are decayed to 9/30/2020. Tanks 17 through 20 are decayed to 1/1/2020.

Table A-33. 2020 FTF Estimated Radiological Inventory (Ci) at Closure - continued

Tank	Na22	Nb94	Ni59	Ni63	Np237	Pa231	Pm147	Pr144	Pu238	Pu239	Pu240	Pu241
1	2.64E-05	2.70E-04	5.20E-01	3.82E+01	1.54E-02	4.06E-07	6.46E-02	1.27E-16	1.06E+01	3.20E+00	7.13E-01	1.87E+00
2	6.26E-06	7.19E-05	1.40E-01	1.00E+01	8.82E-03	3.24E-08	3.03E-03	2.03E-20	5.10E+00	8.23E-01	1.84E-01	3.68E-01
3	5.49E-06	6.02E-05	1.18E-01	8.50E+00	1.58E-02	8.09E-08	4.67E-03	3.33E-19	6.14E+00	9.90E-01	2.21E-01	5.33E-01
4	8.89E-06	7.15E-05	1.33E-01	1.05E+01	3.98E-03	9.91E-08	5.14E-01	5.61E-12	7.15E-01	7.96E-01	1.78E-01	7.17E-01
5	3.02E-05	2.84E-04	5.43E-01	4.09E+01	2.32E-02	5.55E-07	1.69E-01	2.56E-15	1.42E+01	2.67E+00	6.36E-01	2.03E+00
6	3.94E-05	3.39E-04	6.31E-01	4.89E+01	5.86E-03	4.32E-07	4.33E-01	3.59E-14	7.17E-01	1.73E+00	6.00E-01	3.21E+00
7	9.93E-07	8.55E-06	1.98E-02	1.54E+00	6.33E-04	3.25E-08	2.03E-02	5.54E-14	2.31E+00	5.03E-01	1.21E-01	7.51E-01
8	2.48E-06	2.04E-05	4.24E-02	3.33E+00	1.28E-03	5.88E-08	5.99E-02	1.67E-13	3.26E+00	8.06E-01	1.90E-01	8.61E-01
17	NE	NE	1.83E-01	NE	1.38E-02	NE	5.69E-02	1.58E-13	5.43E+01	1.48E+01	3.35E+00	9.68E+01
18	1.12E-02	1.01E-05	2.30E-01	1.81E+01	3.81E-02	1.15E-06	6.10E-02	1.67E-13	1.56E+01	3.72E+01	2.15E+01	2.89E+01
19	4.47E-02	9.24E-06	2.11E-01	1.72E+01	2.84E-03	2.01E-07	9.19E-02	1.11E-13	5.39E+00	7.92E+00	2.82E+00	5.89E+00
20	NE	NE	3.90E-02	NE	7.16E-04	NE	1.87E-02	2.88E-14	6.15E+00	8.53E-01	1.77E-01	1.67E+01
25	1.74E-07	6.77E-07	1.54E-02	1.35E+00	3.78E-06	1.17E-08	1.84E-01	1.67E-08	6.90E+00	1.22E+00	2.73E-01	3.06E+00
26	3.07E-07	1.19E-06	2.72E-02	2.38E+00	6.64E-06	4.51E-06	3.24E-01	2.94E-08	1.22E+01	2.15E+00	4.80E-01	5.38E+00
27	1.44E-07	5.58E-07	1.27E-02	1.12E+00	3.11E-06	9.62E-09	1.52E-01	1.38E-08	5.69E+00	1.01E+00	2.25E-01	2.52E+00
28	3.07E-07	1.19E-06	2.72E-02	2.38E+00	6.64E-06	2.05E-08	3.24E-01	2.94E-08	1.22E+01	2.15E+00	4.80E-01	5.38E+00
33	4.19E-05	2.21E-04	4.28E-01	3.81E+01	2.47E-02	5.20E-07	6.70E+01	3.86E-06	3.57E+01	2.20E+01	3.85E+00	5.45E+01
34	1.69E-04	9.69E-04	1.80E+00	1.57E+02	6.76E-02	1.19E-06	1.05E+02	3.59E-07	9.28E-01	1.41E+01	3.15E+00	3.11E+01
44	3.07E-07	1.19E-06	2.72E-02	2.38E+00	6.64E-06	2.05E-08	3.24E-01	2.94E-08	1.22E+01	2.15E+00	4.80E-01	5.38E+00
45	3.07E-07	1.19E-06	2.72E-02	2.38E+00	6.64E-06	2.05E-08	3.24E-01	2.94E-08	1.22E+01	2.15E+00	4.80E-01	5.38E+00
46	3.07E-07	1.19E-06	2.72E-02	2.38E+00	6.64E-06	2.05E-08	3.24E-01	2.94E-08	1.22E+01	2.15E+00	4.80E-01	5.38E+00
47	3.07E-07	1.19E-06	2.72E-02	2.38E+00	6.64E-06	2.05E-08	3.24E-01	2.94E-08	1.22E+01	2.15E+00	4.80E-01	5.38E+00
Total	**5.63E-02**	**2.34E-03**	**5.23E+00**	**4.09E+02**	**2.23E-01**	**9.40E-06**	**1.76E+02**	**4.43E-06**	**2.47E+02**	**1.24E+02**	**4.11E+01**	**2.82E+02**

NE = Not Estimated.

NOTE: All values, except Tanks 17 through 20 are decayed to 9/30/2020. Tanks 17 through 20 are decayed to 1/1/2020.

Table A-33. 2020 FTF Estimated Radiological Inventory (Ci) at Closure - continued

Tank	Pu242	Pu244	Ra226	Rh106	Ru106	Sb125	Sb126	Sb126m	Se79	Sm151	Sn126	Sr90
1	1.47E-04	6.72E-07	1.54E-02	3.03E-12	3.03E-12	5.45E-03	9.41E-02	6.72E-01	3.61E-01	9.27E+02	6.72E-01	8.59E+03
2	3.78E-05	1.73E-07	1.22E-03	3.62E-15	3.62E-15	2.78E-04	2.50E-02	1.79E-01	9.61E-02	2.38E+02	1.79E-01	2.04E+03
3	4.56E-05	2.08E-07	3.06E-03	2.40E-14	2.40E-14	4.16E-04	2.10E-02	1.50E-01	8.05E-02	2.02E+02	1.50E-01	1.79E+03
4	3.66E-05	1.67E-07	6.47E-03	7.83E-09	7.83E-09	3.75E-02	2.48E-02	1.77E-01	9.54E-02	2.63E+02	1.77E-01	2.89E+03
5	1.88E-04	8.59E-07	1.99E-02	3.26E-11	3.26E-11	1.37E-02	9.88E-02	7.06E-01	3.80E-01	1.00E+03	7.06E-01	9.82E+03
6	1.18E-03	5.41E-06	2.45E-02	2.49E-10	2.49E-10	3.44E-02	1.18E-01	8.41E-01	4.52E-01	1.23E+03	8.41E-01	1.28E+04
7	1.35E-04	6.19E-07	2.03E-03	8.46E-11	8.46E-11	1.56E-03	3.00E-03	2.14E-02	1.15E-02	3.09E+01	2.14E-02	3.24E+02
8	2.40E-04	1.09E-06	4.11E-03	2.53E-10	2.53E-10	4.58E-03	7.12E-03	5.08E-02	2.73E-02	7.50E+01	5.08E-02	8.06E+02
17	5.32E-03	NE	NE	3.30E-10	3.30E-10	4.28E-03	3.98E-03	2.84E-02	1.56E-02	NE	2.84E-02	6.69E+01
18	2.60E-01	8.56E-05	1.23E-07	3.50E-10	3.50E-10	4.57E-03	4.95E-03	3.53E-02	1.28E-02	1.02E+01	3.53E-02	2.38E+02
19	9.46E-02	4.23E-04	2.68E-08	3.44E-10	3.44E-10	6.90E-03	4.53E-03	3.24E-02	1.50E-02	2.89E-01	3.24E-02	6.60E+00
20	1.60E-03	NE	NE	8.07E-11	8.07E-11	1.38E-03	8.26E-04	5.90E-03	3.20E-03	NE	5.90E-03	2.30E+01
25	5.44E-05	2.49E-07	2.12E-03	8.91E-07	8.91E-07	1.19E-02	6.18E-04	2.37E-03	1.28E-03	4.01E+00	2.37E-03	5.67E+01
26	9.57E-05	4.38E-07	3.74E-03	1.57E-06	1.57E-06	2.10E-02	5.84E-04	4.17E-03	2.25E-03	7.06E+00	4.17E-03	9.99E+01
27	4.48E-05	2.05E-07	1.76E-03	7.34E-07	7.34E-07	9.82E-03	5.84E-04	1.95E-03	1.05E-03	3.31E+00	1.95E-03	4.67E+01
28	9.57E-05	4.38E-07	3.74E-03	1.57E-06	1.57E-06	2.10E-02	5.84E-04	4.17E-03	2.25E-03	7.06E+00	4.17E-03	9.99E+01
33	2.23E-04	1.02E-06	1.14E-01	2.88E-04	2.88E-04	4.28E+00	7.67E-02	5.48E-01	2.95E-01	9.32E+02	5.48E-01	1.36E+04
34	6.55E-04	3.00E-06	1.27E-01	6.80E-05	6.80E-05	7.01E+00	3.36E-01	2.40E+00	1.29E+00	3.99E+03	2.40E+00	5.51E+04
44	9.57E-05	4.38E-07	3.74E-03	1.57E-06	1.57E-06	2.10E-02	5.84E-04	4.17E-03	2.25E-03	7.06E+00	4.17E-03	9.99E+01
45	9.57E-05	4.38E-07	3.74E-03	1.57E-06	1.57E-06	2.10E-02	5.84E-04	4.17E-03	2.25E-03	7.06E+00	4.17E-03	9.99E+01
46	9.57E-05	4.38E-07	3.74E-03	1.57E-06	1.57E-06	2.10E-02	5.84E-04	4.17E-03	2.25E-03	7.06E+00	4.17E-03	9.99E+01
47	9.57E-05	4.38E-07	3.74E-03	1.57E-06	1.57E-06	2.10E-02	5.84E-04	4.17E-03	2.25E-03	7.06E+00	4.17E-03	9.99E+01
Total	**3.65E-01**	**5.25E-04**	**3.44E-01**	**3.67E-04**	**3.67E-04**	**1.16E+01**	**8.24E-01**	**5.88E+00**	**3.15E+00**	**8.95E+03**	**5.88E+00**	**1.09E+05**

NE = Not Estimated.

NOTE: All values, except Tanks 17 through 20 are decayed to 9/30/2020. Tanks 17 through 20 are decayed to 1/1/2020.

Table A-33. 2020 FTF Estimated Radiological Inventory (Ci) at Closure - continued

Tank	Tc99	Te125m	Th229	Th230	U232	U233	U234	U235	U236	U238	Y90
1	6.25E+00	1.33E-03	1.62E-02	1.53E-02	2.37E-04	1.55E-02	1.06E-02	4.24E-04	2.18E-04	1.06E-02	8.59E+03
2	1.66E+00	6.78E-05	9.28E-03	1.22E-03	6.06E-05	8.86E-03	8.44E-04	3.38E-05	1.74E-05	8.44E-04	2.04E+03
3	1.39E+00	1.02E-04	1.66E-02	3.04E-03	5.17E-05	1.58E-02	2.11E-03	8.45E-05	4.35E-05	2.11E-03	1.79E+03
4	1.65E+00	9.16E-03	4.18E-03	6.43E-03	6.80E-05	3.99E-03	4.46E-03	1.04E-04	9.12E-05	4.46E-03	2.89E+03
5	6.57E+00	3.35E-03	2.44E-02	1.98E-02	2.57E-04	2.33E-02	1.37E-02	5.79E-04	2.83E-04	1.37E-02	9.82E+03
6	7.82E+00	8.41E-03	6.17E-03	2.44E-02	3.15E-04	5.88E-03	1.69E-02	4.52E-04	3.46E-04	1.69E-02	1.28E+04
7	1.99E-01	3.80E-04	6.66E-04	2.02E-03	8.59E-06	6.35E-04	1.40E-03	3.40E-05	2.86E-05	1.40E-03	3.24E+02
8	4.73E-01	1.12E-03	1.35E-03	4.08E-03	2.00E-05	1.29E-03	2.83E-03	6.14E-05	5.78E-05	2.83E-03	8.06E+02
17	9.01E-01	1.05E-03	NE	NE	3.75E-05	NE	NE	3.03E-04	NE	6.42E-03	6.69E+01
18	3.72E-01	1.12E-03	5.72E-04	1.51E-05	4.69E-05	3.85E-01	1.72E-01	1.93E-03	3.10E-03	4.92E-02	2.38E+02
19	1.72E+00	1.69E-03	6.64E-04	3.29E-06	4.50E-05	2.36E-01	1.33E-02	3.22E-04	7.35E-04	1.09E-02	6.60E+00
20	8.50E-01	3.37E-04	NE	NE	8.03E-06	NE	NE	1.90E-05	2.70E-05	5.58E-04	2.30E+01
25	2.21E-02	2.91E-03	3.97E-06	2.11E-03	3.66E-06	3.79E-06	1.46E-03	1.22E-05	2.96E-05	1.46E-03	5.67E+01
26	3.90E-02	5.12E-03	6.98E-06	3.72E-03	6.44E-06	6.66E-06	2.58E-03	2.15E-05	5.23E-05	2.58E-03	9.99E+01
27	1.82E-02	2.40E-03	3.27E-06	1.74E-03	3.01E-06	3.12E-06	1.21E-03	1.00E-05	2.45E-05	1.21E-03	4.67E+01
28	3.90E-02	5.11E-03	6.98E-06	3.72E-03	6.44E-06	6.66E-06	2.58E-03	2.15E-05	5.23E-05	2.58E-03	9.99E+01
33	5.10E+00	1.05E+00	2.60E-02	1.13E-01	2.52E-04	2.48E-02	7.86E-02	5.43E-04	1.59E-03	7.86E-02	1.36E+04
34	2.23E+01	1.71E+00	7.11E-02	1.26E-01	1.06E-03	6.78E-02	8.75E-02	1.24E-03	1.78E-03	8.75E-02	5.51E+04
44	3.90E-02	5.12E-03	6.98E-06	3.72E-03	6.44E-06	6.66E-06	2.58E-03	2.15E-05	5.23E-05	2.58E-03	9.99E+01
45	3.90E-02	5.12E-03	6.98E-06	3.72E-03	6.44E-06	6.66E-06	2.58E-03	2.15E-05	5.23E-05	2.58E-03	9.99E+01
46	3.90E-02	5.12E-03	6.98E-06	3.72E-03	6.44E-06	6.66E-06	2.58E-03	2.15E-05	5.23E-05	2.58E-03	9.99E+01
47	3.90E-02	5.12E-03	6.98E-06	3.72E-03	6.44E-06	6.66E-06	2.58E-03	2.15E-05	5.23E-05	2.58E-03	9.99E+01
Total	**5.75E+01**	**2.82E+00**	**1.77E-01**	**3.41E-01**	**2.51E-03**	**7.89E-01**	**4.23E-01**	**6.28E-03**	**8.69E-03**	**3.04E-01**	**1.09E+05**

NE = Not Estimated.

NOTE: All values, except Tanks 17 through 20 are decayed to 9/30/2020. Tanks 17 through 20 are decayed to 1/1/2020.

Table A-34. 2020 Radionuclide Inventory for the FTF Ancillary Equipment

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Ac227	1.05E+00	Eu152	2.64E-01	Rh106	1.06E-05
Al26	1.24E-04	Eu154	2.86E+00	Ru106	1.06E-05
Am241	7.62E-01	Eu155	2.42E+00	Sb125	1.62E-01
Am242m	1.24E-02	H3	2.89E-01	Sb126	9.38E-03
Am243	1.37E-02	I129	3.25E-06	Sb126m	6.70E-02
Ba137m	8.33E+02	Na22	3.93E-06	Se79	3.59E-02
Bk249	8.14E-29	Nb94	2.66E-05	Sm151	1.03E+02
C14	7.08E-04	Ni59	6.93E-02	Sn126	5.39E+01
Ce144	3.13E-07	Ni63	5.77E+00	Sr90	1.29E+03
Cf249	3.41E-20	Np237	3.32E-03	Tc99	7.48E-01
Cm242	8.90E-20	Pa231	3.86E-07	Te125	3.66E-02
Cm243	5.28E-04	Pm147	2.48E+00	Th229	1.78E-05
Cm244	2.32E+00	Pr144	3.13E-07	Th230	4.48E-07
Cm245	3.79E-05	Pu238	8.70E+00	U232	3.10E-05
Cm247	5.46E-18	Pu239	4.26E+00	U233	3.59E-02
Cm248	1.26E-18	Pu240	1.19E+00	U234	2.05E-02
Co60	9.10E-01	Pu241	1.16E+01	U235	1.32E-04
Cs134	3.45E-04	Pu242	5.71E-03	U236	2.58E-04
Cs135	4.18E-04	Pu244	3.88E-06	U238	1.50E+01
Cs137	8.91E+02	Ra226	8.12E-09	Y90	1.34E+03

Table A-35. 2025 Inventory Estimates Developed for the Central Laboratory Facility

Radionuclide	772-F Inventory (Ci)	772-1F Inventory (Ci)	772-4F Inventory (Ci)
Am241	1.3E-01	1.3E-01	8.5E-07
Am243	5.2E-03	5.2E-03	--
Ba137m	6.7E-01	6.2E-02	3.4E-05
C14	1.2E-05	1.1E-06	6.0E-10
Ce144	2.9E-02	2.7E-03	8.1E-06
Cf249	3.7E+02	3.7E+02	--
Cf251	2.9E+02	2.9E+02	--
Cm244	1.5E+01	1.5E+01	1.0E-07
Cm245	5.2E+00	5.2E+00	--
Cm247	9.3E-05	9.3E-05	--
Co60	4.6E-03	4.3E-04	2.4E-07
Cs134	--	--	5.5E-06
Cs137	7.1E-01	6.5E-02	3.6E-05
H3	1.1E+01	1.0E-01	--
I129	2.1E-07	2.0E-08	1.1E-11
Ni59	1.4E-05	1.3E-06	7.0E-10
Np237	2.2E-05	2.1E-05	--
Pm147	3.3E-01	2.7E-03	1.7E-05
Pr144	2.9E-02	2.7E-03	8.1E-06
Pr144m	2.9E-02	2.7E-03	8.1E-06
Pu238	3.9E+00	3.9E+00	3.7E-05
Pu239	1.9E-01	1.2E-01	4.1E-06
Pu240	4.2E-02	1.0E-02	--
Pu241	4.7E+02	5.0E+02	--
Pu242	5.9E-05	5.9E-05	--
Rh106	2.4E-02	2.2E-03	1.2E-06
Ru106	2.4E-02	2.2E-03	1.2E-06
Se79	3.1E-06	2.9E-07	1.6E-10
Sn126	3.7E-07	3.4E-08	1.9E-11
Sr90	3.8E-01	3.5E-02	7.5E-06
Tc99	1.2E-04	1.1E-05	6.0E-09
U233	1.7E-02	1.7E-02	--
U234	3.2E-04	4.4E-05	--
U235	1.4E-05	2.8E-06	3.0E-07
U236	5.1E-05	7.0E-06	--
U238	9.1E-04	1.2E-04	--
Y90	3.8E-01	3.5E-02	7.5E-06

Table A-36. 2002 Radionuclide Inventory for OFASB

Radionuclide	Inventory (Ci)
Am241	1.89E+00
Co60	3.17E-02
Cs137	1.86E+01
Eu154	4.82E-02
I129	2.48E-01
K40	6.19E-02
Nb95	7.16E-03
Pm147	1.18E-01
Pu238	3.77E-01
Pu239	2.96E+00
Pu240	1.29E-01
Ra226	2.20E-02
Ra228	3.17E-02
Sr90	6.10E-01
Tc99	7.47E-01
Th232	1.10E-02
U234	1.46E-01
U235	1.36E-01
U238	5.20E-01

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A.5 H-AREA FACILITIES

Table A-37. 2032 HTF Estimated Radionuclide Inventory (Ci) at Closure

Tank	Ac227	Al26	Am241	Am242m	Am243	Ba137m	Bk249	C14	Ce144	Cf249	Cf251	Cf252
9	7.6E-09	3.6E-04	1.9E+01	2.2E-02	6.1E-02	1.7E+02	1.5E-28	1.9E-04	7.5E-25	2.6E-12	9.3E-14	4.3E-18
10	1.5E-09	3.8E-05	2.0E+00	2.2E-03	6.4E-03	1.8E+01	1.6E-29	2.0E-05	1.9E-25	2.8E-13	9.7E-15	4.4E-19
11	3.9E-09	3.5E-04	1.5E+01	8.6E-03	5.9E-02	1.7E+02	1.5E-28	1.7E-04	6.6E-17	2.6E-12	9.0E-14	4.1E-18
12	6.2E-09	4.2E-04	1.5E+01	1.2E-02	7.1E-02	2.0E+02	1.8E-28	2.3E-04	4.4E-19	3.0E-12	1.1E-13	4.9E-18
13	1.4E-08	3.8E-04	1.5E+01	1.5E-02	6.5E-02	1.8E+02	1.6E-28	1.5E-04	2.4E-19	2.8E-12	9.8E-14	4.5E-18
14	3.3E-09	9.9E-05	4.8E+00	5.2E-03	1.7E-02	4.7E+01	4.2E-29	9.2E-06	2.7E-23	7.2E-13	2.5E-14	1.2E-18
15	9.2E-09	5.8E-04	1.9E+01	1.6E-02	9.9E-02	2.8E+02	2.5E-28	3.3E-04	4.9E-18	4.3E-12	1.5E-13	6.8E-18
16	9.2E-09	5.8E-04	1.9E+01	1.6E-02	9.9E-02	2.8E+02	2.5E-28	3.3E-04	4.9E-18	4.3E-12	1.5E-13	6.8E-18
21	6.2E-09	6.5E-05	1.1E+00	7.0E-03	1.1E-02	3.1E+01	2.8E-29	6.0E-05	1.2E-12	4.8E-13	1.7E-14	7.7E-19
22	2.5E-09	2.5E-05	3.7E-01	4.0E-04	4.3E-03	1.2E+01	1.1E-29	2.4E-04	2.1E-17	1.8E-13	6.5E-15	3.0E-19
23	4.5E-09	2.0E-06	2.6E+00	1.6E-03	3.5E-04	9.7E-01	8.7E-31	1.8E-05	2.2E-12	1.5E-14	5.3E-16	2.4E-20
24	4.5E-09	9.8E-05	2.6E+00	1.6E-03	1.7E-02	5.4E+02	4.2E-29	1.8E-05	2.2E-12	7.2E-13	2.5E-14	1.2E-18
29	1.4E-08	3.8E-04	1.5E+01	1.5E-02	6.5E-02	1.8E+02	1.6E-28	1.5E-04	2.4E-19	2.8E-12	9.8E-14	4.5E-18
30	1.2E-08	7.6E-04	5.2E+01	1.4E-02	1.3E-01	3.6E+02	3.2E-28	2.7E-04	6.1E-13	5.5E-12	1.9E-13	8.9E-18
31	1.4E-08	3.8E-04	1.5E+01	1.5E-02	6.5E-02	1.8E+02	1.6E-28	1.5E-04	2.4E-19	2.8E-12	9.8E-14	4.5E-18
32	7.0E-09	9.5E-04	4.5E+01	2.1E-02	1.6E-01	4.9E+02	4.1E-28	4.2E-04	2.2E-13	6.9E-12	2.4E-13	1.1E-17
35	1.7E-08	1.7E-03	8.4E+01	3.6E-02	2.9E-01	8.1E+02	7.3E-28	7.0E-04	6.6E-13	1.2E-11	4.4E-13	2.0E-17
36	1.7E-08	6.4E-04	3.9E+01	1.4E-02	1.1E-01	3.1E+02	2.8E-28	2.8E-04	5.9E-16	4.7E-12	1.7E-13	7.6E-18
37	7.0E-09	9.5E-04	4.5E+01	2.1E-02	1.6E-01	4.5E+02	4.1E-28	4.2E-04	2.2E-13	6.9E-12	2.4E-13	1.1E-17
38	4.5E-09	9.8E-05	2.6E+00	1.6E-03	1.7E-02	5.5E+02	4.2E-29	1.8E-05	2.2E-12	7.2E-13	2.5E-14	1.2E-18
39	4.2E-08	1.0E-03	5.1E+01	1.9E-02	1.8E-01	5.0E+02	4.5E-28	3.7E-04	1.8E-10	7.6E-12	2.7E-13	1.2E-17
40	1.1E-08	4.6E-04	1.8E+01	1.2E-02	7.9E-02	2.2E+02	2.0E-28	2.1E-04	1.4E-11	3.4E-12	1.2E-13	5.5E-18
41	1.1E-08	4.6E-04	1.8E+01	1.2E-02	7.9E-02	2.2E+02	2.0E-28	2.1E-04	1.4E-11	3.4E-12	1.2E-13	5.5E-18
42	6.7E-09	2.1E-04	6.6E+00	5.4E-03	3.5E-02	1.2E+02	8.8E-29	2.4E-04	3.0E-17	1.5E-12	5.3E-14	2.4E-18
43	4.5E-09	9.8E-05	2.6E+00	1.6E-03	1.7E-02	4.7E+01	4.2E-29	1.8E-05	2.2E-12	7.2E-13	2.5E-14	1.2E-18
48	1.1E-08	4.6E-04	1.8E+01	1.2E-02	7.9E-02	2.2E+02	2.0E-28	2.1E-04	1.4E-11	3.4E-12	1.2E-13	5.5E-18
49	1.1E-08	4.6E-04	1.8E+01	1.2E-02	7.9E-02	2.2E+02	2.0E-28	2.1E-04	1.4E-11	3.4E-12	1.2E-13	5.5E-18
50	1.1E-08	4.6E-04	1.8E+01	1.2E-02	7.9E-02	2.2E+02	2.0E-28	2.1E-04	1.4E-11	3.4E-12	1.2E-13	5.5E-18
51	1.1E-08	4.6E-04	1.8E+01	1.2E-02	7.9E-02	2.2E+02	2.0E-28	2.1E-04	1.4E-11	3.4E-12	1.2E-13	5.5E-18
Total	**2.8E-07**	**1.3E-02**	**5.8E+02**	**3.4E-01**	**2.2E+00**	**7.2E+03**	**5.6E-27**	**6.0E-03**	**2.8E-10**	**9.5E-11**	**3.3E-12**	**1.5E-16**

Table A-37. 2032 HTF Estimated Radionuclide Inventory (Ci) at Closure - continued

Tank	Cl36	Cm242	Cm243	Cm244	Cm245	Cm247	Cm248	Co60	Cs134	Cs135	Cs137	Eu152
9	8.0E-02	2.1E-19	1.4E-03	1.1E-03	4.4E-09	4.8E-13	5.0E-13	2.5E-02	1.1E-09	1.8E-03	1.8E+02	3.5E-01
10	8.2E-03	2.2E-20	1.4E-04	1.1E-04	4.5E-10	5.0E-14	5.2E-14	2.8E-03	1.4E-10	1.9E-04	1.9E+01	3.7E-02
11	3.9E-02	8.5E-20	1.3E-03	9.7E-02	6.4E-06	4.6E-13	4.8E-13	3.0E-01	2.2E-06	1.2E-03	1.8E+02	3.4E-01
12	5.4E-02	1.2E-19	1.6E-03	1.5E-01	7.4E-06	5.5E-13	5.8E-13	2.3E-01	6.4E-07	1.6E-03	2.1E+02	4.0E-01
13	6.1E-02	1.4E-19	1.4E-03	1.9E-01	4.8E-06	5.1E-13	5.3E-13	1.2E-01	2.8E-07	1.6E-03	1.9E+02	3.7E-01
14	1.9E-02	5.1E-20	3.7E-04	9.8E-04	3.8E-07	1.3E-13	1.4E-13	1.2E-02	3.1E-09	4.6E-04	5.0E+01	9.6E-02
15	7.4E-02	1.6E-19	2.2E-03	1.2E-01	1.2E-05	7.7E-13	8.1E-13	2.8E-01	6.6E-07	2.3E-03	2.9E+02	5.7E-01
16	7.4E-02	1.6E-19	2.2E-03	1.2E-01	1.2E-05	7.7E-13	8.1E-13	2.8E-01	6.6E-07	2.3E-03	2.9E+02	5.7E-01
21	6.0E-03	6.9E-20	2.5E-04	1.1E+00	3.6E-04	8.7E-14	9.0E-14	7.4E-02	3.7E-06	1.8E-04	3.3E+01	6.3E-02
22	2.5E-03	4.0E-21	9.6E-05	1.4E-03	4.1E-07	3.3E-14	3.5E-14	2.9E-02	5.4E-07	7.7E-05	1.3E+01	2.4E-02
23	7.5E-03	1.5E-20	7.7E-06	6.6E-03	1.2E-06	2.7E-15	2.8E-15	4.6E-05	5.1E-09	2.8E-03	1.0E+00	2.0E-03
24	7.5E-03	1.5E-20	3.7E-04	6.6E-03	1.2E-06	1.3E-13	1.4E-13	4.7E-01	5.2E-04	2.8E-03	5.7E+02	9.5E-02
29	6.1E-02	1.4E-19	1.4E-03	1.9E-01	4.8E-06	5.1E-13	5.3E-13	1.2E-01	2.8E-07	1.6E-03	1.9E+02	3.7E-01
30	6.0E-02	1.4E-19	2.9E-03	4.9E-02	9.8E-06	1.0E-12	1.0E-12	3.0E+00	2.5E-04	1.9E-03	3.8E+02	7.3E-01
31	6.1E-02	1.4E-19	1.4E-03	1.9E-01	4.8E-06	5.1E-13	5.3E-13	1.2E-01	2.8E-07	1.6E-03	1.9E+02	3.7E-01
32	9.4E-02	2.1E-19	3.6E-03	5.4E-02	1.5E-05	1.3E-12	1.3E-12	1.6E+00	7.7E-05	3.2E-03	5.2E+02	9.2E-01
35	1.6E-01	3.6E-19	6.5E-03	1.0E-01	2.6E-05	2.3E-12	2.4E-12	3.8E+00	1.6E-04	4.8E-03	8.6E+02	1.7E+00
36	6.1E-02	1.4E-19	2.4E-03	3.7E-02	1.0E-05	8.5E-13	8.9E-13	1.1E+00	1.9E-05	1.9E-03	3.2E+02	6.2E-01
37	9.4E-02	2.1E-19	3.6E-03	5.4E-02	1.5E-05	1.3E-12	1.3E-12	1.6E+00	7.4E-05	2.9E-03	4.8E+02	9.2E-01
38	7.5E-03	1.5E-20	3.7E-04	6.6E-03	1.2E-06	1.3E-13	1.4E-13	4.7E-01	5.3E-04	2.8E-03	5.8E+02	9.5E-02
39	8.1E-02	1.9E-19	4.0E-03	9.9E+01	2.7E-02	1.4E-12	1.4E-12	5.2E+00	1.2E-03	2.5E-03	5.3E+02	1.0E+00
40	4.6E-02	1.2E-19	1.8E-03	1.0E+01	2.9E-03	6.2E-13	6.4E-13	7.8E-01	1.1E-04	1.4E-03	2.4E+02	4.5E-01
41	4.6E-02	9.6E-03	1.8E-03	1.0E+01	2.9E-03	6.2E-13	6.4E-13	7.8E-01	1.1E-04	1.4E-03	2.3E+02	4.5E-01
42	2.6E-02	5.3E-20	7.8E-04	3.2E-01	4.2E-06	2.7E-13	2.8E-13	1.1E-01	7.4E-07	9.5E-04	1.2E+02	2.0E-01
43	7.5E-03	1.5E-20	3.7E-04	6.6E-03	1.2E-06	1.3E-13	1.4E-13	4.7E-01	8.3E-05	2.3E-04	4.9E+01	9.5E-02
48	4.6E-02	1.2E-19	1.8E-03	1.0E+01	2.9E-03	6.2E-13	6.4E-13	7.8E-01	1.1E-04	1.4E-03	2.3E+02	4.5E-01
49	4.6E-02	1.2E-19	1.8E-03	1.0E+01	2.9E-03	6.2E-13	6.4E-13	7.8E-01	1.1E-04	1.4E-03	2.3E+02	4.5E-01
50	4.6E-02	1.2E-19	1.8E-03	1.0E+01	2.9E-03	6.2E-13	6.4E-13	7.8E-01	1.1E-04	1.4E-03	2.3E+02	4.5E-01
51	4.6E-02	1.2E-19	1.8E-03	1.0E+01	2.9E-03	6.2E-13	6.4E-13	7.8E-01	1.1E-04	1.4E-03	2.3E+02	4.5E-01
Total	**1.4E+00**	**2.8E-01**	**4.9E-02**	**1.6E+02**	**4.5E-02**	**1.7E-11**	**1.8E-11**	**2.4E+01**	**3.6E-03**	**5.0E-02**	**7.7E+03**	**1.3E+01**

Table A-37. 2032 HTF Estimated Radionuclide Inventory (Ci) at Closure - continued

Tank	Eu154	Eu155	H3	I129	K40	Na22	Nb93m	Nb94	Ni59	Ni63	Np237	Pa231
9	6.5E-01	3.8E-01	1.6E-02	1.3E-05	5.9E-03	1.3E-06	4.3E+00	1.2E-04	2.3E-01	1.5E+01	1.3E-02	4.7E-08
10	7.1E-02	4.0E-02	1.6E-02	1.3E-06	6.1E-04	1.4E-07	4.4E-01	1.2E-05	2.3E-02	1.5E+00	2.7E-03	9.1E-09
11	6.4E+00	3.7E-01	1.6E-02	6.2E-06	5.7E-03	1.3E-06	9.8E-01	2.7E-05	1.9E-01	1.4E+01	1.3E-03	2.4E-08
12	5.7E+00	4.4E-01	1.6E-02	8.6E-06	6.8E-03	1.5E-06	1.6E+00	4.4E-05	2.5E-01	1.8E+01	7.1E-03	3.8E-08
13	3.5E+00	4.0E-01	2.0E-02	9.8E-06	6.2E-03	1.4E-06	2.4E+00	6.5E-05	2.2E-01	1.5E+01	1.3E-02	8.6E-08
14	3.4E-01	1.0E-01	2.0E-02	3.1E-06	1.6E-03	3.6E-07	9.7E-01	2.7E-05	5.8E-02	3.9E+00	4.3E-03	2.1E-08
15	7.7E+00	6.1E-01	2.0E-02	1.2E-05	9.5E-03	2.1E-06	1.9E+00	5.1E-05	3.7E-01	2.6E+01	5.2E-03	5.7E-08
16	7.7E+00	6.1E-01	2.0E-02	1.2E-05	9.5E-03	2.1E-06	1.9E+00	5.1E-05	3.7E-01	2.6E+01	5.2E-03	5.7E-08
21	1.5E+00	6.9E-02	2.0E-02	9.6E-07	1.1E-03	2.4E-07	1.1E-01	2.9E-06	2.1E-02	1.6E+00	3.2E-03	3.8E-08
22	6.1E-01	2.7E-02	2.0E-02	4.0E-07	4.1E-04	9.1E-08	4.5E-02	1.2E-06	8.8E-03	6.7E-01	9.7E-04	1.5E-08
23	4.5E+00	2.1E-03	2.0E-02	1.2E-06	3.3E-05	7.4E-09	1.6E-01	4.5E-06	3.2E-02	2.7E+00	3.4E-03	2.8E-08
24	4.5E+00	1.0E-01	2.0E-02	1.2E-06	1.6E-03	3.6E-07	1.6E-01	4.5E-06	3.2E-02	2.7E+00	3.4E-03	2.8E-08
29	3.5E+00	4.0E-01	2.0E-02	9.8E-06	6.2E-03	1.4E-06	2.4E+00	6.5E-05	2.2E-01	1.5E+01	1.3E-02	8.6E-08
30	3.1E+01	8.0E-01	2.0E-02	9.6E-06	1.2E-02	2.7E-06	1.5E+00	4.2E-05	3.0E-01	2.4E+01	2.9E-03	7.2E-08
31	3.5E+00	4.0E-01	2.0E-02	9.8E-06	6.2E-03	1.4E-06	2.4E+00	6.5E-05	2.2E-01	1.5E+01	1.3E-02	8.6E-08
32	2.4E+01	1.0E+00	2.0E-02	1.5E-05	1.5E-02	3.4E-06	2.4E+00	6.5E-05	4.6E-01	3.5E+01	2.0E-03	4.3E-08
35	5.0E+01	1.8E+00	2.0E-02	2.5E-05	2.8E-02	6.2E-06	4.0E+00	1.1E-04	7.8E-01	6.0E+01	3.4E-03	1.0E-07
36	1.7E+01	6.8E-01	2.0E-02	9.8E-06	1.0E-02	2.3E-06	1.5E+00	4.2E-05	3.0E-01	2.3E+01	2.4E-03	1.1E-07
37	2.4E+01	1.0E+00	2.0E-02	1.5E-05	1.5E-02	3.4E-06	2.4E+00	6.5E-05	4.6E-01	3.5E+01	2.0E-03	4.3E-08
38	4.5E+00	1.0E-01	2.0E-02	1.2E-06	1.6E-03	3.6E-07	1.6E-01	4.5E-06	3.2E-02	2.7E+00	3.4E-03	2.8E-08
39	4.6E+01	1.1E+00	2.0E-02	1.3E-05	1.7E-02	3.8E-06	2.1E+00	5.7E-05	4.0E-01	3.3E+01	1.2E-02	2.6E-07
40	9.8E+00	4.9E-01	2.0E-02	7.4E-06	7.5E-03	1.7E-06	1.3E+00	3.7E-05	2.1E-01	1.5E+01	7.0E-03	6.7E-08
41	9.8E+00	4.9E-01	2.0E-02	7.4E-06	7.5E-03	1.7E-06	1.3E+00	3.7E-05	2.1E-01	1.5E+01	7.0E-03	6.7E-08
42	2.9E+00	2.2E-01	2.0E-02	4.1E-06	3.3E-03	7.5E-07	6.4E-01	1.8E-05	1.3E-01	9.3E+00	2.3E-03	4.1E-08
43	4.5E+00	1.0E-01	2.0E-02	1.2E-06	1.6E-03	3.6E-07	1.6E-01	4.5E-06	3.2E-02	2.7E+00	3.4E-03	2.8E-08
48	9.8E+00	4.9E-01	2.0E-02	7.4E-06	7.5E-03	1.7E-06	1.3E+00	3.7E-05	2.1E-01	1.5E+01	7.0E-03	6.7E-08
49	9.8E+00	4.9E-01	2.0E-02	7.4E-06	7.5E-03	1.7E-06	1.3E+00	3.7E-05	2.1E-01	1.5E+01	7.0E-03	6.7E-08
50	9.8E+00	4.9E-01	2.0E-02	7.4E-06	7.5E-03	1.7E-06	1.3E+00	3.7E-05	2.1E-01	1.5E+01	7.0E-03	6.7E-08
51	9.8E+00	4.9E-01	2.0E-02	7.4E-06	7.5E-03	1.7E-06	1.3E+00	3.7E-05	2.1E-01	1.5E+01	7.0E-03	6.7E-08
Total	**3.1E+02**	**1.4E+01**	**5.7E-01**	**2.3E-04**	**2.1E-01**	**4.7E-05**	**4.3E+01**	**1.2E-03**	**6.4E+00**	**4.7E+02**	**1.6E-01**	**1.7E-06**

Table A-37. 2032 HTF Estimated Radionuclide Inventory (Ci) at Closure - continued

Tank	Pd107	Pm147	Pr144	Pt193	Pu238	Pu239	Pu240	Pu241	Pu242	Pu244	Ra226	Ra228
9	3.3E-02	2.0E-04	7.5E-25	6.4E+00	2.2E+00	3.9E-01	8.7E-02	9.6E-02	1.8E-05	8.2E-08	6.9E-10	4.2E-09
10	3.4E-03	2.6E-05	1.9E-25	6.6E-01	1.1E+00	2.0E-01	4.4E-02	5.2E-02	9.1E-06	4.2E-08	6.9E-10	4.2E-09
11	3.1E-02	8.7E-03	6.6E-17	6.0E+00	1.4E+02	1.8E+00	1.1E+00	1.7E+01	2.4E-03	1.1E-05	4.5E-09	8.3E-07
12	3.9E-02	3.6E-03	4.4E-19	7.6E+00	1.0E+02	2.1E+00	1.1E+00	7.7E+00	1.7E-03	7.6E-06	7.2E-09	5.9E-05
13	3.4E-02	1.7E-03	2.4E-19	6.7E+00	3.8E+01	1.3E+00	5.2E-01	2.4E+00	3.5E-04	1.6E-06	1.1E-08	8.3E-06
14	8.5E-03	1.2E-04	2.7E-23	1.7E+00	1.6E+00	4.7E-01	1.5E-01	2.1E-01	2.8E-05	1.3E-07	9.0E-10	4.0E-06
15	5.7E-02	3.8E-03	4.9E-18	1.1E+01	6.7E+01	1.8E+00	8.6E-01	4.7E+00	6.5E-04	3.0E-06	1.2E-08	8.0E-05
16	5.7E-02	3.8E-03	4.9E-18	1.1E+01	6.7E+01	1.8E+00	8.6E-01	4.7E+00	6.5E-04	3.0E-06	1.2E-08	8.0E-05
21	3.4E-03	9.0E-02	1.2E-12	6.7E-01	1.5E+01	2.8E-01	9.5E-02	4.8E-01	1.1E-05	4.9E-08	8.5E-09	5.5E-09
22	1.5E-03	1.7E-03	2.1E-17	2.9E-01	5.3E+00	2.8E-01	9.5E-02	4.8E-01	1.1E-05	4.9E-08	3.5E-09	5.5E-09
23	5.8E-03	1.1E-01	2.2E-12	1.1E+00	6.3E+01	3.1E-01	2.6E-01	1.8E+01	3.8E-03	1.7E-05	7.1E-09	5.5E-09
24	5.8E-03	1.1E-01	2.2E-12	1.1E+00	6.3E+01	3.1E-01	2.6E-01	1.8E+01	3.8E-03	1.7E-05	7.1E-09	5.5E-09
29	3.4E-02	1.7E-03	2.4E-19	6.7E+00	3.8E+01	1.3E+00	5.2E-01	2.4E+00	3.5E-04	1.6E-06	1.1E-08	8.3E-06
30	5.3E-02	4.4E-01	6.1E-13	1.0E+01	9.9E+02	1.1E+01	7.6E+00	2.1E+02	1.8E-02	8.4E-05	1.4E-08	5.5E-09
31	3.4E-02	1.7E-03	2.4E-19	6.7E+00	3.8E+01	1.3E+00	5.2E-01	2.4E+00	3.5E-04	1.6E-06	1.1E-08	8.3E-06
32	7.7E-02	1.4E-01	2.2E-13	1.5E+01	5.6E+02	6.1E+00	4.5E+00	8.2E+01	9.5E-03	4.3E-05	1.1E-08	5.5E-09
35	1.3E-01	3.3E-01	6.6E-13	2.6E+01	1.1E+03	1.1E+01	8.7E+00	1.8E+02	2.0E-02	9.0E-05	2.1E-08	5.5E-09
36	5.1E-02	5.7E-02	5.9E-16	1.0E+01	6.0E+02	5.8E+00	4.6E+00	8.0E+01	1.0E-02	4.6E-05	1.9E-08	5.5E-09
37	7.7E-02	1.4E-01	2.2E-13	1.5E+01	5.6E+02	6.1E+00	4.5E+00	8.2E+01	9.5E-03	4.3E-05	1.1E-08	5.5E-09
38	5.8E-03	1.1E-01	2.2E-12	1.1E+00	6.3E+01	3.1E-01	2.6E-01	1.8E+01	3.8E-03	1.7E-05	7.1E-09	5.5E-09
39	7.2E-02	1.4E+00	1.8E-10	1.4E+01	7.7E+02	1.2E+01	7.4E+00	2.3E+02	1.6E-02	7.2E-05	5.9E-08	5.5E-09
40	3.3E-02	1.4E-01	1.4E-11	6.4E+00	1.8E+02	3.2E+00	1.8E+00	3.5E+01	3.6E-03	1.6E-05	1.1E-08	1.5E-05
41	3.3E-02	1.4E-01	1.4E-11	6.4E+00	1.8E+02	3.2E+00	1.8E+00	3.5E+01	3.6E-03	1.6E-05	1.1E-08	5.5E-09
42	2.0E-02	2.7E-03	3.0E-17	4.0E+00	2.9E+01	1.3E+00	4.6E-01	2.1E+00	4.2E-04	1.9E-06	6.2E-09	2.6E-05
43	5.8E-03	1.1E-01	2.2E-12	1.1E+00	6.3E+01	3.1E-01	2.6E-01	1.8E+01	3.8E-03	1.7E-05	7.1E-09	5.5E-09
48	3.3E-02	1.4E-01	1.4E-11	6.4E+00	1.8E+02	3.2E+00	1.8E+00	3.5E+01	3.6E-03	1.6E-05	1.1E-08	1.5E-05
49	3.3E-02	1.4E-01	1.4E-11	6.4E+00	1.8E+02	3.2E+00	1.8E+00	3.5E+01	3.6E-03	1.6E-05	1.1E-08	1.5E-05
50	3.3E-02	1.4E-01	1.4E-11	6.4E+00	1.8E+02	3.2E+00	1.8E+00	3.5E+01	3.6E-03	1.6E-05	1.1E-08	1.5E-05
51	3.3E-02	1.4E-01	1.4E-11	6.4E+00	1.8E+02	3.2E+00	1.8E+00	3.5E+01	3.6E-03	1.6E-05	1.1E-08	1.5E-05
Total	**1.0E+00**	**3.9E+00**	**2.8E-10**	**2.0E+02**	**6.5E+03**	**8.6E+01**	**5.6E+01**	**1.2E+03**	**1.3E-01**	**5.8E-04**	**3.2E-07**	**3.5E-04**

Table A-37. 2032 HTF Estimated Radionuclide Inventory (Ci) at Closure - continued

Tank	Rh106	Ru106	Sb125	Sb126	Sb126m	Se79	Sm151	Sn126	Sr90	Tc99	Te125m	Th229
9	1.7E-18	1.7E-18	2.3E-05	4.1E-02	2.9E-01	1.6E-01	2.2E+02	2.9E-01	2.5E+03	2.7E+00	5.6E-06	6.4E-06
10	3.3E-19	3.3E-19	2.8E-06	4.2E-03	3.0E-02	1.6E-02	2.3E+01	3.0E-02	2.6E+02	2.8E-01	6.9E-07	6.4E-06
11	2.4E-13	2.4E-13	4.9E-04	1.4E-02	9.8E-02	1.1E-01	2.1E+02	9.8E-02	3.1E+03	1.8E+00	1.2E-04	6.9E-06
12	8.0E-15	8.0E-15	2.1E-04	2.0E-02	1.4E-01	1.4E-01	2.5E+02	1.4E-01	3.6E+03	2.4E+00	5.2E-05	9.8E-05
13	3.8E-15	3.8E-15	1.1E-04	2.6E-02	1.9E-01	1.4E-01	2.3E+02	1.9E-01	3.1E+03	2.4E+00	2.7E-05	1.0E-04
14	4.8E-18	4.8E-18	1.2E-05	9.5E-03	6.8E-02	4.0E-02	5.9E+01	6.8E-02	7.1E+02	6.8E-01	2.8E-06	8.3E-06
15	2.1E-14	2.1E-14	2.2E-04	2.6E-02	1.9E-01	2.0E-01	3.5E+02	1.9E-01	5.1E+03	3.4E+00	5.4E-05	5.8E-05
16	2.1E-14	2.1E-14	2.2E-04	2.6E-02	1.9E-01	2.0E-01	3.5E+02	1.9E-01	5.1E+03	3.4E+00	5.4E-05	5.8E-05
21	1.7E-10	1.7E-10	2.6E-04	2.0E-03	1.4E-02	1.6E-02	3.9E+01	1.4E-02	5.7E+02	2.7E-01	6.3E-05	1.0E-05
22	8.6E-14	8.6E-14	9.2E-05	8.7E-04	6.2E-03	6.8E-03	1.5E+01	6.2E-03	2.3E+02	1.2E-01	2.2E-05	1.3E-05
23	7.7E-10	7.7E-10	5.4E-03	2.6E-03	1.9E-02	2.0E-02	1.2E+00	1.9E-02	9.0E+02	3.5E-01	1.3E-03	8.3E-06
24	7.7E-10	7.7E-10	5.4E-03	2.6E-03	1.9E-02	2.0E-02	5.9E+01	1.9E-02	9.0E+02	3.5E-01	1.3E-03	8.3E-06
29	3.8E-15	3.8E-15	1.1E-04	2.6E-02	1.9E-01	1.4E-01	2.3E+02	1.9E-01	3.1E+03	2.4E+00	2.7E-05	1.0E-04
30	5.7E-10	5.7E-10	2.2E-02	2.1E-02	1.5E-01	1.6E-01	4.5E+02	1.5E-01	6.9E+03	2.8E+00	5.4E-03	8.3E-06
31	3.8E-15	3.8E-15	1.1E-04	2.6E-02	1.9E-01	1.4E-01	2.3E+02	1.9E-01	3.1E+03	2.4E+00	2.7E-05	1.0E-04
32	1.8E-10	1.8E-10	7.3E-03	3.3E-02	2.3E-01	2.6E-01	5.7E+02	2.3E-01	8.5E+03	4.3E+00	1.8E-03	8.3E-06
35	4.1E-10	4.1E-10	1.7E-02	5.5E-02	3.9E-01	4.3E-01	1.0E+03	3.9E-01	1.5E+04	7.2E+00	4.1E-03	8.3E-06
36	2.8E-12	2.8E-12	3.1E-03	2.1E-02	1.5E-01	1.7E-01	3.9E+02	1.5E-01	5.8E+03	2.8E+00	7.5E-04	8.3E-06
37	1.8E-10	1.8E-10	7.3E-03	3.3E-02	2.3E-01	2.6E-01	5.7E+02	2.3E-01	8.5E+03	4.3E+00	1.8E-03	8.3E-06
38	7.7E-10	7.7E-10	5.4E-03	2.6E-03	1.9E-02	2.0E-02	5.9E+01	1.9E-02	9.0E+02	3.5E-01	1.3E-03	8.3E-06
39	3.3E-08	3.3E-08	6.4E-02	2.9E-02	2.0E-01	2.2E-01	6.3E+02	2.0E-01	9.5E+03	3.8E+00	1.6E-02	8.3E-06
40	2.5E-09	2.5E-09	6.6E-03	1.8E-02	1.3E-01	1.2E-01	2.8E+02	1.3E-01	3.6E+03	2.0E+00	1.6E-03	3.2E-05
41	2.5E-09	2.5E-09	6.6E-03	1.8E-02	1.3E-01	1.2E-01	2.8E+02	1.3E-01	3.6E+03	2.0E+00	1.6E-03	8.3E-06
42	1.1E-13	1.1E-13	1.5E-04	9.1E-03	6.5E-02	7.0E-02	1.2E+02	6.5E-02	1.8E+03	1.2E+00	3.7E-05	2.1E-05
43	7.7E-10	7.7E-10	5.4E-03	2.6E-03	1.9E-02	2.0E-02	5.9E+01	1.9E-02	9.0E+02	3.5E-01	1.3E-03	8.3E-06
48	2.5E-09	2.5E-09	6.6E-03	1.8E-02	1.3E-01	1.2E-01	2.8E+02	1.3E-01	3.6E+03	2.0E+00	1.6E-03	3.2E-05
49	2.5E-09	2.5E-09	6.6E-03	1.8E-02	1.3E-01	1.2E-01	2.8E+02	1.3E-01	3.6E+03	2.0E+00	1.6E-03	3.2E-05
50	2.5E-09	2.5E-09	6.6E-03	1.8E-02	1.3E-01	1.2E-01	2.8E+02	1.3E-01	3.6E+03	2.0E+00	1.6E-03	3.2E-05
51	2.5E-09	2.5E-09	6.6E-03	1.8E-02	1.3E-01	1.2E-01	2.8E+02	1.3E-01	3.6E+03	2.0E+00	1.6E-03	3.2E-05
Total	**5.3E-08**	**5.3E-08**	**1.8E-01**	**5.4E-01**	**3.8E+00**	**3.7E+00**	**7.8E+03**	**3.8E+00**	**1.1E+05**	**6.3E+01**	**4.5E-02**	**8.3E-04**

Table A-37. 2032 HTF Estimated Radionuclide Inventory (Ci) at Closure - continued

Tank	Th230	Th232	U232	U233	U234	U235	U236	U238	Y90	Zr93
9	8.6E-08	8.8E-08	8.7E-05	1.3E-02	1.8E-03	7.4E-05	3.8E-05	1.8E-03	2.5E+03	4.3E+00
10	8.6E-08	8.8E-08	8.9E-06	2.7E-03	3.6E-04	1.4E-05	7.3E-06	3.6E-04	2.6E+02	4.4E-01
11	5.5E-07	1.7E-05	1.1E-06	2.4E-03	2.0E-03	3.8E-05	3.1E-04	7.0E-05	3.1E+03	9.8E-01
12	8.9E-07	1.2E-03	9.4E-06	3.4E-02	3.2E-03	6.1E-05	2.6E-04	6.1E-04	3.6E+03	1.6E+00
13	1.4E-06	1.7E-04	3.5E-05	3.5E-02	4.9E-03	1.4E-04	5.1E-04	1.6E-03	3.1E+03	2.4E+00
14	1.1E-07	8.3E-05	1.9E-05	2.9E-03	4.0E-04	3.3E-05	3.5E-05	6.2E-04	7.1E+02	9.7E-01
15	1.5E-06	1.7E-03	1.4E-06	2.0E-02	5.6E-03	9.0E-05	5.3E-04	2.1E-06	5.1E+03	1.9E+00
16	1.5E-06	1.7E-03	1.4E-06	2.0E-02	5.6E-03	9.0E-05	5.3E-04	2.1E-06	5.1E+03	1.9E+00
21	1.1E-06	1.1E-07	1.4E-06	3.6E-03	3.8E-03	6.1E-05	7.0E-04	3.6E-04	5.7E+02	1.1E-01
22	4.4E-07	1.1E-07	1.4E-06	4.4E-03	1.6E-03	2.4E-05	2.6E-04	5.4E-04	2.3E+02	4.5E-02
23	8.8E-07	1.1E-07	1.4E-06	3.4E-03	3.2E-03	4.4E-05	6.1E-04	3.6E-06	9.0E+02	1.6E-01
24	8.8E-07	1.1E-07	1.4E-06	3.4E-03	3.2E-03	4.4E-05	6.1E-04	3.6E-06	9.0E+02	1.6E-01
29	1.4E-06	1.7E-04	3.5E-05	3.5E-02	4.9E-03	1.4E-04	5.1E-04	1.6E-03	3.1E+03	2.4E+00
30	1.8E-06	1.1E-07	1.4E-06	2.9E-03	6.4E-03	1.1E-04	1.2E-03	4.2E-06	6.9E+03	1.5E+00
31	1.4E-06	1.7E-04	3.5E-05	3.5E-02	4.9E-03	1.4E-04	5.1E-04	1.6E-03	3.1E+03	2.4E+00
32	1.3E-06	1.1E-07	1.4E-06	2.0E-03	4.8E-03	6.8E-05	1.1E-03	3.2E-05	8.5E+03	2.4E+00
35	2.6E-06	1.1E-07	1.4E-06	3.4E-03	9.5E-03	1.6E-04	2.8E-03	9.7E-05	1.5E+04	4.0E+00
36	2.4E-06	1.1E-07	1.4E-06	2.4E-03	8.6E-03	1.7E-04	3.0E-03	5.4E-05	5.8E+03	1.5E+00
37	1.3E-06	1.1E-07	1.4E-06	2.0E-03	4.8E-03	6.8E-05	1.1E-03	3.2E-05	8.5E+03	2.4E+00
38	8.8E-07	1.1E-07	1.4E-06	3.4E-03	3.2E-03	4.4E-05	6.1E-04	3.6E-06	9.0E+02	1.6E-01
39	7.3E-06	1.1E-07	1.4E-06	1.2E-02	2.7E-02	4.1E-04	4.5E-03	5.7E-05	9.5E+03	2.1E+00
40	1.4E-06	3.2E-04	9.6E-06	1.1E-02	5.0E-03	1.1E-04	7.9E-04	1.0E-03	3.6E+03	1.3E+00
41	1.4E-06	1.1E-07	1.4E-06	7.0E-03	5.0E-03	1.1E-04	7.9E-04	1.0E-03	3.6E+03	1.3E+00
42	7.6E-07	5.3E-04	1.4E-06	7.5E-03	2.8E-03	6.5E-05	3.7E-04	1.1E-03	1.8E+03	6.4E-01
43	8.8E-07	1.1E-07	1.4E-06	3.4E-03	3.2E-03	4.4E-05	6.1E-04	3.6E-06	9.0E+02	1.6E-01
48	1.4E-06	3.2E-04	9.6E-06	1.1E-02	5.0E-03	1.1E-04	7.9E-04	1.0E-03	3.6E+03	1.3E+00
49	1.4E-06	3.2E-04	9.6E-06	1.1E-02	5.0E-03	1.1E-04	7.9E-04	1.0E-03	3.6E+03	1.3E+00
50	1.4E-06	3.2E-04	9.6E-06	1.1E-02	5.0E-03	1.1E-04	7.9E-04	1.0E-03	3.6E+03	1.3E+00
51	1.4E-06	3.2E-04	9.6E-06	1.1E-02	5.0E-03	1.1E-04	7.9E-04	1.0E-03	3.6E+03	1.3E+00
Total	**4.0E-05**	**7.3E-03**	**3.0E-04**	**3.2E-01**	**1.5E-01**	**2.7E-03**	**2.5E-02**	**1.7E-02**	**1.1E+05**	**4.3E+01**

Table A-38. 2032 H-Area Tank Farm Ancillary Equipment Inventory

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Ac227	1.5E-08	Pa231	9.6E-08
Al26	5.2E-04	Pd107	4.3E-02
Am241	2.2E+01	Pm147	1.1E-01
Am242m	1.7E-02	Pr144	4.2E-12
Am243	8.8E-02	Pt193	8.3E+00
Ba137m	3.1E+02	Pu238	2.0+02
Bk249	2.2E-28	Pu239	2.7E+00
C14	2.3E-04	Pu240	1.7E+00
Ce144	4.2E-12	Pu241	3.1E+01
Cf249	3.8E-12	Pu242	4.2E-03
Cf251	1.3E-13	Pu244	2.0E-05
Cf252	6.1E-18	Ra226	1.7E-08
Cl36	6.6E-02	Ra228	9.9E-06
Cm242	1.3E-02	Rh106	9.1E-10
Cm243	2.0E-03	Ru106	9.1E-10
Cm244	2.4E+00	Sb125	4.7E-03
Cm245	6.4E-04	Sb126	2.7E-02
Cm247	6.9E-13	Sb126m	1.9E-01
Cm248	7.1E-13	Se79	1.7E-01
Co60	6.6E-01	Sm151	3.1E+02
Cs134	1.1E-04	Sn126	1.9E-01
Cs135	2.5E-03	Sr90	4.4E+03
Cs137	3.3E+02	Tc99	2.8E+00
Eu152	5.0E-01	Te125m	1.1E-03
Eu154	1.0E+01	Th229	6.9E-05
Eu155	5.5E-01	Th230	2.0E-06
H3	1.2E+00	Th232	2.1E-04
I129	1.2E-05	U232	2.4E-05
K40	8.3E-03	U233	2.5E-02
Na22	1.9E-06	U234	7.2E-03
Nb93m	2.2E+00	U235	1.5E-04
Nb94	6.1E-05	U236	1.1E-03
Ni59	2.7E-01	U238	1.2E-03
Ni63	2.0E+01	Y90	4.4E+03
Np237	1.1E-02	Zr93	2.2E+00

Table A-39. 2002 Radionuclide Inventory for the H-Area Inactive Process Sewer Lines

Radionuclide	Inventory (Ci)
H3	2.85E+03
Cs137	3.81E+01
Sr90	1.94E+01
I129	1.28E-01
Am241	2.07E-01
Cm244	2.72E-02
Co60	5.15E-01
Cs134	6.54E-02
Pu238	3.27E-01
Pu239	5.50E+00
Ra226	7.63E-02
Th232	7.38E-02
U234	1.91E-01
U238	1.91E-01

Table A-40. 2002 Radionuclide Inventory for H-Area Retention Basins (281-1H, -2H, -8H)

Radionuclide	281-1H Inventory (Ci)	281-2H Inventory (Ci)	281-8H Inventory (Ci)
Ac228	3.85E-03	1.96E-03	1.08E-01
Am241	1.23E-01	6.28E-02	3.47E+00
C14	9.57E-04	4.88E-04	2.69E-02
Cm242	2.89E-05	1.47E-05	8.14E-04
Cm244	7.74E-01	3.95E-01	2.18E+01
Cm245	1.58E-04	8.07E-05	4.46E-03
Co60	4.52E-03	2.30E-03	1.27E-01
Cs134	1.55E-05	7.91E-06	4.37E-04
Cs137	3.86E+01	1.97E+01	1.09E+03
Eu152	6.37E-05	3.25E-05	1.79E-03
Eu154	4.62E-02	2.36E-02	1.30E+00
Eu155	3.13E-05	1.59E-05	8.81E-04
K40	2.88E-03	1.47E-03	8.10E-02
Na22	1.01E-04	5.13E-05	2.83E-03
Np237	2.21E-04	1.13E-04	6.23E-03
Pb212	4.95E-03	2.52E-03	1.39E-01
Pm147	3.62E-03	1.84E-03	1.02E-01
Pu238	1.62E+00	8.26E-01	4.56E+01
Pu239	1.11E+00	5.67E-01	3.13E+01
Pu240	2.66E-01	1.36E-01	7.50E+00
Ra226	2.55E-03	1.30E-03	7.19E-02
Ra228	4.19E-03	2.13E-03	1.18E-01
Sr90	9.47E+00	4.83E+00	2.67E+02
Tc99	1.60E-02	8.16E-03	4.50E-01
Th228	1.30E-02	6.65E-03	3.67E-01
Th230	4.11E-03	2.10E-03	1.16E-01
Th232	1.10E-02	5.60E-03	3.09E-01
U233	5.53E-04	2.82E-04	1.56E-02
U234	4.70E-02	2.40E-02	1.32E+00
U235	2.22E-03	1.13E-03	6.26E-02
U238	4.03E-02	2.06E-02	1.14E+00

Table A-41. 2002 Radionuclide Inventory for the H-Area Seepage Basins

Radionuclide	Inventory (Ci)
Am241	3.93E-01
Cs137	1.89E+02
H3	3.66E+05
I129	1.54E+00
Pu238	1.16E+00
Pu239	4.06E+00
Sr90	5.35E+01
Tc99	6.31E-01
U234	2.10E+00
U235	1.45E+00
U238	1.85E+00

Table A-42. 2025 Radionuclide Inventory for HB Line

Radionuclide	Inventory (Ci)
Am241	8.5E+01
Np237	1.4E+01
Pu236	8.8E+00
Pu238	6.8E+02
Pu239	3.5E+01
Pu240	5.0E+01
Pu241	2.6E+03
U232	1.5E-05

Table A-43. 2025 Radionuclide Inventory for H-Canyon

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Ag110	7.6E-02	Pu239	1.4E+02
Am241	3.1E+03	Pu240	9.0E+02
Ba137m	2.8E+04	Pu241	7.4E+04
C14	4.8E-01	Pu242	5.3E+01
Ce144	1.3E+02	Rh106	6.4E+01
Co60	1.9E+02	Ru106	6.4E+01
Cs134	1.2E+03	Sb125	1.3E+02
Cs137	2.9E+04	Se79	1.3E-01
Eu155	2.0E+02	Sn123	8.5E-06
H3	1.1E+02	Sn126	1.5E-02
I129	8.8E-03	Sr90	9.7E+03
Ni59	5.7E-01	Tc99	4.8E+00
Np237	6.8E+02	U233	8.4E-08
Pa233	6.8E+02	U234	2.9E+00
Pm147	5.3E+03	U235	2.9E-02
Pr144	1.3E+02	U236	7.5E-01
Pr144m	1.3E+02	U238	2.1E-03
Pu238	1.8E+05	Y90	9.7E+03

Table A-44. 2007 Radionuclide Inventory for Receiving Basin for Offsite Fuels (RBOF)

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Am241	1.2E-03	Ni59	1.2E-04
Am242	0.0E+00	Ni63	8.5E-03
Am242m	1.4E-06	Np237	4.8E-06
Am243	3.6E-05	Pb212	8.9E-06
Ba137m	1.7E-04	Pu238	9.2E-03
Bi212	1.8E-05	Pu239	3.7E-03
Be7	0.0E+00	Pu240	0.0E+00
C14	2.6E-05	Pu241	5.8E-02
Cf249	1.0E-05	Pu242	1.7E-04
Cf251	8.1E-06	Ra226	2.1E-04
Cm242	9.4E-08	Sb124	5.5E-07
Cm243	6.8E-05	Sb125	2.1E-04
Cm245	6.6E-06	Se79	1.1E-06
Cm247	9.7E-06	Sn126	4.6E-05
Co57	8.7E-05	Sr90	1.7E-02
Co58	1.5E-06	Tc99	1.1E-03
Co60	1.5E-03	Th232	4.5E-09
Cr51	3.6E-05	Tl208	5.1E-06
Cs134	2.2E-04	U232	5.7E-06
Cs137	8.0E-02	U233	4.7E-05
Eu152	2.9E-04	U234	2.9E-05
Eu154	4.1E-04	U235	1.1E-06
Eu155	6.5E-05	U236	3.2E-07
H3	5.6E-02	U238	1.4E-05
I129	3.8E-07	Y88	6.7E-08
K40	1.0E-05	Y90	2.3E-05
Na22	5.7E-05	Zn65	2.7E-05
Nb94	6.2E-06	Zr95	2.0E-05

Table A-45. 2025 Radionuclide Inventory for H-Area Sand Filter

Radionuclide	Inventory (Ci)
Ru106	1.02E+01
Cs137	4.81E+03
Ce144	4.16E+00
Pu238	2.35E+01

Table A-46. Radionuclide Inventories for Original Tritium Facilities

Building 232-H	
Radionuclide	**2007 Inventory (Ci)**
H3	4.10E+05
HANM, Gas Separation, Purification, Mixing, and Loading Facility	
Radionuclide	**2025 Inventory (Ci)**
H3	1.45E+06
HAOM, Receiving, Finishing and Packaging Facilities	
Radionuclide	**2025 Inventory (Ci)**
H3	8.70E+04 [1]
Pu238	8.55E-01 [1]
Vault 217-H, Long Term Storage Vault	
Radionuclide	**2025 Inventory (Ci)**
H3	3.87E+05 [1]
Building 234-7H, Materials Test Facility	
Radionuclide	**2025 Inventory (Ci)**
H3	1.93E+04 [1]
Building 236-H, Byproduct Purification Facility	
Radionuclide	**2025 Inventory (Ci)**
H3	2.90E+03
Building 237/238-H, Reclamation and Empty Reservoir Storage Facility	
Radionuclide	**2025 Inventory (Ci)**
H3	2.90E+02 [1]

[1] The inventory was corrected from that in Hiergesell et al. (2008) for consistency with the projected "end state" presented in Phifer et al. 2008. The Hiergesell et al. (2008) inventory for these facilities was based upon an ISD "end state" configuration rather than the appropriate DTCS "end state"; therefore, the Hiergesell et al. (2008) inventory has been reduced by a factor of 10. Per Table 11-2, item 14 in Volume I, Hiergesell et al. (2008) will be revised.

Table A-47. 2025 Radionuclide Inventories for Tritium Extraction Facility and Remote Handling Building (264-2H)

Tritium Extraction Facility		264-2H Remote Handling Building	
Radionuclide	**Inventory (Ci)**	**Radionuclide**	**Inventory (Ci)**
H3	2.10E+05	H3	7.26E+05
TPBAR Activation Products in 264-2H Remote Handling Building			
Radionuclide	**Inventory (Ci)**	**Radionuclide**	**Inventory (Ci)**
Ar37	2.2E-01	Ni63	6.8E+01
Ar39	2.8E-02	Ni66	4.8E-15
As76	4.3E-23	Os191	3.5E-06
Ba131	8.4E-04	P32	5.5E-02
Ba133	2.2E-03	Re186	2.3E-08
Ba133m	4.2E-20	Re188	2.2E-02
Ba135m	1.1E-24	Ru103	2.2E-03
Br82	3.5E-20	S35	2.1E-02
C14	4.3E-03	Sb122	1.8E-10
Ca41	2.3E-04	Sb124	2.1E-02
Ca45	6.6E-01	Sb125	4.8E+00
Ca47	1.5E-09	Sb126	1.6E-03
Cd115	5.3E-15	Sc46	1.2E-02
Cd115m	1.5E-04	Sc47	5.6E-09
Co58	3.6E+02	Se75	1.6E+00
Co60	1.0E+02	Sn113	2.4E+00
Cr51	3.6E+02	Sn117m	4.1E-01
Cs131	3.5E-03	Sn119m	2.0E+01
Cu66	0.0E+00	Sn121	9.4E-24
Fe55	4.8E-15	Sn121m	1.7E-03
Fe59	6.1E+02	Sn123	9.2E-01
Hf175	1.7E+01	Sn125	1.7E-02
Hf181	4.3E-02	Sr89	7.2E-02
In113m	6.8E-01	Ta182	2.0E+01
In114	2.4E+00	Ta183	9.6E-04
In114m	1.2E-01	Tc99	1.3E-04
K42	1.2E-01	Te123m	5.6E-03
La140	2.5E-11	Te125m	1.1E+00
Lu177	1.8E-08	W181	1.1E-02

Table A-47. 2025 Radionuclide Inventories for Tritium Extraction Facility and Remote Handling Building (264-2H) - continued

TPBAR Activation Products in 264-2H Remote Handling Building			
Radionuclide	**Inventory (Ci)**	**Radionuclide**	**Inventory (Ci)**
Mn54	4.7E-06	W185	2.9E-01
Mo93	1.0E+02	W187	6.5E-27
Mo99	3.1E-03	W188	2.2E-02
Nb92	4.1E-08	Y89m	3.7E-11
Nb93m	0.0E+00	Y90	4.1E-06
Nb94	3.2E-03	Y91	2.2E-01
Nb95	1.5E-05	Zn65	9.8E-03
Nb95m	1.4E-03	Zr89	3.8E-11
Nb96	1.3E+02	Zr93	3.4E-04
Ni59	5.9E-01	Zr95	8.0E+01

A.6 N-AREA FACILITIES

Table A-48. 2002 Radionuclide Inventory for Ford Building Seepage Basin

Radionuclide	Inventory (Ci)
Cs137	6.00E-03

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A.7 REACTOR AREAS

Table A-49. 2008 Radionuclide Inventory for 105-P Reactor Vessel Volumetrically Contaminated 304 Stainless Steel

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
H3	7.79E-03	Nb95m	6.89E-36	I129	1.20E-06
Be10	1.69E-05	Zr93	4.05E-03	Ba133	7.92E-09
C14	2.48E+02	Zr95	1.67E-32	Ba137m	1.74E+00
Na22	3.12E-08	Mo93	3.75E+00	Cs134	5.27E-04
Si32	5.26E-06	Tc97	2.99E-14	Cs135	4.23E-05
P32	5.26E-06	Tc97m	2.24E-33	Cs137	1.84E+00
S35	1.19E-23	Tc98	4.61E-09	La137	4.19E-10
Cl36	3.31E-04	Tc99	6.61E-01	La138	7.91E-15
Ar39	5.66E-02	Rh101	1.39E-13	Ce139	2.68E-23
Ar42	9.18E-12	Rh102	1.65E-14	Ce144	5.27E-08
K40	1.41E-05	Rh102m	5.71E-12	Pm143	2.18E-23
K42	9.18E-12	Rh106	2.34E-06	Pm144	5.01E-19
Ca41	1.25E-01	Ru106	2.34E-06	Pm145	2.51E-11
Ca45	7.05E-13	Pd107	5.95E-06	Pm146	1.31E-09
Sc46	1.48E-27	Ag108	1.52E+01	Pm147	5.55E-03
V49	3.74E-11	Ag108m	1.74E+02	Pr144	5.27E-08
V50	1.81E-11	Ag109m	9.78E-04	Pr144m	7.38E-10
Mn54	2.62E-04	Ag110	3.33E-08	Nd144	6.58E-14
Fe55	9.29E+03	Ag110m	2.45E-06	Sm145	2.56E-19
Co58	1.61E-27	Cd109	9.78E-04	Sm146	5.16E-13
Co60	4.44E+04	Cd113	2.97E-10	Sm147	3.62E-10
Ni59	1.51E+03	Cd113m	3.14E+00	Sm148	8.34E-16
Ni63	1.54E+05	In113m	6.09E-18	Sm151	1.55E-02
Zn65	1.27E-07	In115	5.04E-07	Eu149	9.35E-37
Ga68	5.21E-29	Sn113	6.09E-18	Eu152	5.03E-04
Ge68	5.21E-29	Sn119m	1.88E-05	Eu154	1.37E-02
Se75	1.31E-28	Sn121	3.78E-01	Eu155	3.28E-03
Se79	1.27E-05	Sn121m	4.87E-01	Gd151	4.11E-28
Kr81	1.16E-11	Sn123	3.40E-18	Gd152	2.54E-16
Kr85	3.42E-02	Sn126	2.56E-05	Gd153	6.20E-13
Rb83	1.83E-33	Sb124	1.48E-05	Tb157	1.45E-10
Rb87	7.95E-10	Sb125	1.55E-02	Tb158	1.39E-09
Y88	5.90E-28	Sb126	3.58E-06	Tb160	2.55E-33
Y90	1.10E+00	Sb126m	2.56E-05	Dy159	9.67E-25
Y91	2.12E-37	Te121	5.62E-25	Ho163	1.95E-12
Sr90	1.09E+00	Te121m	5.66E-25	Ho166m	1.08E-08

Table A-49. 2008 Radionuclide Inventory for 105-P Reactor Vessel Volumetrically Contaminated 304 Stainless Steel - continued

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Nb91	9.11E-09	Te123	2.24E-10	Tm168	1.19E-35
Nb92	1.21E-06	Te123m	1.06E-16	Tm170	5.56E-24
Nb93m	2.52E+00	Te125m	3.77E-03	Tm171	1.04E-10
Nb94	6.92E+00	Te127	6.06E-22	Tl204	6.34E-07
Nb95	3.70E-32	Te127m	6.19E-22	Tl206	3.10E-04
Tl207	7.69E-09	Ra225	8.28E-12	Pu236	1.58E-10
Tl208	8.67E-08	Ra226	1.53E-09	Pu238	1.18E-01
Tl209	1.74E-13	Ra228	1.37E-14	Pu239	2.44E-02
Pb205	1.22E-04	Ac225	8.28E-12	Pu240	3.79E-02
Pb209	8.28E-12	Ac227	7.70E-09	Pu241	1.05E+00
Pb210	5.88E-10	Ac228	1.37E-14	Pu242	1.16E-04
Pb211	7.71E-09	Th227	7.60E-09	Pu243	1.28E-13
Pb212	2.41E-07	Th228	2.41E-07	Pu244	4.55E-13
Pb214	1.53E-09	Th229	8.28E-12	Pu246	3.74E-21
Bi208	6.44E-05	Th230	1.34E-07	Am241	9.51E-02
Bi210m	5.88E-10	Th231	1.19E-05	Am242m	1.81E-03
Bi210m	3.10E-04	Th232	1.70E-14	Am242	1.81E-03
Bi211	7.71E-09	Th234	2.98E-04	Am243	4.72E-04
Bi212	2.41E-07	Pa231	1.47E-08	Am245	1.66E-22
Bi213	8.28E-12	Pa233	2.09E-06	Am246	3.74E-21
Bi214	1.53E-09	Pa234m	2.98E-04	Cm242	1.50E-03
Po210	5.88E-10	Pa234	3.87E-07	Cm243	1.73E-04
Po211	2.12E-11	U232	2.34E-07	Cm244	8.57E-03
Po212	1.55E-07	U233	2.73E-10	Cm245	3.64E-07
Po213	8.11E-12	U234	2.77E-04	Cm246	1.50E-07
Po214	1.53E-09	U235	1.19E-05	Cm247	1.28E-13
Po215	7.71E-09	U236	8.09E-06	Cm248	2.99E-13
Po216	2.41E-07	U237	2.51E-05	Cm250	1.50E-20
Po218	1.53E-09	U238	2.98E-04	Bk249	1.15E-17
At217	8.28E-12	U240	4.54E-13	Bk250	2.09E-21
Rn219	7.71E-09	Np235	1.73E-16	Cf249	5.68E-13
Rn220	2.41E-07	Np236	3.41E-13	Cf250	1.60E-12
Rn222	1.53E-09	Np237	2.09E-06	Cf251	1.67E-14
Fr221	8.28E-12	Np238	8.18E-06	Cf252	1.69E-14
Fr223	1.06E-10	Np239	4.72E-04	Es254	1.60E-24
Ra223	7.71E-09	Np240m	4.54E-13		
Ra224	2.41E-07	Np240	5.45E-16		

Notes: Total 304 stainless steel mass = 3.740E+08 g.
Inventory to be applied to the 105-C, KAC, and LAC buildings also.

Table A-50. 2008 Radionuclide Inventory for 105-P Reactor Building Volumetrically Contaminated Aluminum

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
H3	2.54E-04	Ar42	1.42E-10	Co58	6.63E-35
Be10	2.20E-06	K42	1.42E-10	Co60	7.96E+00
C14	3.23E-03	Ca41	1.75E-18	Ni59	6.71E-11
Na22	2.50E-13	Ca45	1.72E-14	Ni63	2.42E+00
Si32	4.91E-06	Sc46	1.44E-25	Zn65	2.03E-05
P32	4.91E-06	V49	4.54E-18	Se79	1.43E-18
S35	3.61E-33	V50	2.09E-21		
Cl36	3.13E-15	Mn54	1.06E-05		
Ar39	7.00E-15	Fe55	1.16E+02		

Notes: Total aluminum mass = 2.567E+06 g.
Inventory to be applied to the 105-C, KAC, and LAC buildings also.

Table A-51. 2008 Radionuclide Inventory for 105-P Reactor Building Volumetrically Contaminated Concrete

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
H3	1.68E+04	Nb94	8.35E-03	I129	7.18E-05
Be10	1.62E-08	Zr93	2.69E-06	Ba133	2.58E+02
C14	2.43E-01	Mo93	5.94E-03	Cs134	1.23E-03
Na22	8.53E-08	Tc98	1.62E-15	Cs137	1.10E+01
Si32	1.79E-12	Tc99	1.41E-03	Eu152	1.80E+01
P32	1.79E-12	Ru106	2.70E-30	Eu154	1.80E+00
S35	1.67E-25	Rh102	1.25E-28	Tl204	2.46E-10
Cl36	6.42E-05	Rh106	3.73E-11	Tl206	2.80E-07
Ar39	1.38E-01	Pd107	6.28E-12	Pb205	2.92E-08
Ar42	4.24E-13	Ag108	1.07E-02	Bi208	1.18E-07
K40	2.52E-03	Ag108m	1.23E-01	Bi210m	2.80E-07
K42	4.24E-13	Ag109m	1.38E-07	Po210	5.29E-18
Ca41	2.73E-01	Ag110	3.55E-11	Th232	2.95E-03
Ca45	1.79E-12	Ag110m	2.61E-09	U232	1.50E-04
Sc46	6.24E-30	Cd109	1.38E-07	U233	3.84E-02
V49	8.53E-14	Cd113	1.07E-09	U234	3.58E-02
V50	3.38E-11	Cd113m	3.26E-02	U235	6.66E-04
Mn54	2.96E-07	In113m	8.01E-21	U238	4.55E-02
Fe55	2.89E+00	In115	6.18E-07	Np237	1.75E-05
Co58	2.59E-30	Sn113	8.01E-21	Pu238	5.64E-04
Co60	2.22E+01	Sn119m	2.94E-08	Pu239	1.45E-04
Ni59	7.83E-01	Sn121	4.44E-04	Pu240	8.80E-05
Ni63	7.24E+01	Sn121m	5.72E-04	Pu241	1.11E-03
Zn65	2.53E-15	Sn123	2.29E-21	Am241	9.76E-04
Se79	9.71E-04	Sb124	1.21E-03	Am243	2.67E-04
Sr90	8.29E-01	Sb125	2.79E-03		
Y90	5.62E-14	Te123	8.77E-17		
Nb91	4.83E-15	Te123m	1.34E-23		
Nb93m	4.02E-03	Te125m	3.71E-06		

Note: Inventory to be applied to the 105-C, KAC, and LAC buildings also.

Table A-52. 2008 Radionuclide Inventory for 105-P Reactor Building Surface Contamination

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
H3	2.37E+03	Tc99	3.99E-02	Cs137	6.14E+00
Be10	9.91E-08	Ru103	1.98E-03	Ce141	2.17E-03
C14	2.05E+00	Ru106	2.12E-02	Ce144	4.77E-03
Na22	7.32E-03	Rh102	7.31E-19	Pm144	1.07E-03
Si32	4.91E-05	Rh106	7.91E-18	Pm146	1.57E-03
P32	4.91E-05	Pd107	4.90E-09	Eu152	5.10E-02
S35	3.49E-22	Ag108	1.01E-01	Eu154	4.41E-02
Cl36	2.84E+01	Ag108m	1.16E+00	Eu155	5.03E-03
Ar39	3.62E-01	Ag109m	8.88E-03	Pb212	2.70E-03
Ar42	4.66E-07	Ag110	5.53E-10	Ac228	1.16E-02
K40	5.96E-03	Ag110m	4.07E-08	U233	2.73E-02
K42	4.66E-07	Cd109	8.88E-03	U234	2.38E-02
Ca40	1.27E-02	Cd113	3.06E-14	U235	2.19E-02
Ca41	4.19E-01	Cd113m	3.65E+00	U236	2.20E-02
Ca45	2.75E-11	In113m	5.24E-26	U238	2.30E-02
Sc46	3.37E-24	In115	7.37E-11	Np237	2.26E-02
V49	8.99E-13	Sn113	1.54E-03	Np239	4.09E-03
V50	3.43E-16	Sn119m	1.53E-08	Pu238	1.82E+00
Mn54	3.00E-03	Sn121	3.53E-10	Pu239	4.66E-01
Fe55	1.24E+02	Sn121m	4.55E-10	Pu240	4.47E-01
Co57	8.26E-03	Sn123	4.45E-31	Pu241	1.06E+00
Co58	1.76E-03	Sn126	4.02E-02	Pu242	2.46E-02
Co60	1.54E+01	Sb124	4.52E-03	Am241	9.69E-02
Ni59	3.67E+00	Sb125	4.59E-03	Am243	2.12E-02
Ni63	1.83E+02	Te123m	7.86E-28	Cm243	1.11E-02
Zn65	6.13E-03	Te125m	7.62E-18	Cm244	1.11E-02
Se79	1.25E-01	Te127	4.82E-37	Cm245	2.27E-02
Y88	6.29E-04	I129	3.72E-02	Cm246	3.59E-02
Y90	1.43E-01	Ba133	1.24E-03	Cf249	1.19E-03
Sr90	4.82E+00	Ba137m	2.53E-01	Cf251	1.06E-02
Nb94	3.17E-03	Cs134	2.03E-03		
Zr95	2.93E-03	Cs135	3.66E-03		

Note: Inventory to be applied to the 105-C, KAC, and LAC buildings also.

Table A-53. 2008 Radionuclide Inventory for 105-R Reactor Vessel Volumetrically Contaminated 304 Stainless Steel

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
H3	1.51E-03	Nb95m	1.84E-37	I129	7.08E-07
Be10	7.77E-06	Zr93	1.56E-03	Ba133	5.58E-10
C14	9.36E+01	Zr95	3.43E-34	Cs134	1.42E-05
Na22	3.04E-10	Mo93	1.55E+00	Cs135	2.57E-05
Si32	2.31E-06	Tc97	1.65E-14	Cs137	1.06E+00
P32	2.31E-06	Tc97m	5.55E-35	Ba137m	1.00E+00
S35	2.42E-25	Tc98	9.49E-10	La137m	2.41E-10
Cl36	6.06E-05	Tc99	2.76E-01	La138	5.65E-15
Ar39	2.51E-02	Ru106	5.73E-08	Ce139	6.74E-25
Ar42	3.80E-12	Rh101	7.29E-15	Ce144	9.28E-10
K40	1.30E-05	Rh102	5.94E-16	Pm143	5.80E-25
K42	3.80E-12	Rh102m	2.05E-13	Pm144	9.28E-10
Ca41	5.20E-02	Rh106	5.73E-08	Pm144m	1.30E-11
Ca45	2.56E-18	Pd107	3.00E-06	Nd144	4.17E-14
Sc46	2.00E-29	Ag108	5.94E+00	Pm144	1.28E-20
V49	6.03E-13	Ag108m	6.83E+01	Pm145	9.60E-12
V50	1.81E-11	Ag109m	2.49E-05	Sm145	6.82E-21
Mn54	2.84E-06	Ag110	5.94E-10	Pm146	6.85E-10
Fe55	2.07E+02	Ag110m	4.37E-08	Sm146	1.54E-13
Co58	2.64E-29	Cd109	2.49E-05	Pm147	1.51E-04
Co60	3.99E+03	Cd113	4.04E-10	Sm147	2.73E-10
Ni59	6.18E+02	Cd113m	1.18E+00	Sm148	3.35E-16
Ni63	5.70E+04	In113m	1.27E-19	Eu149	2.49E-38
Zn65	3.29E-09	In115	5.20E-07	Sm151	1.06E-02
Ga68	1.37E-30	Sn113	1.27E-19	Gd151	9.64E-30
Ge68	1.37E-30	Sn119m	3.94E-07	Eu152	2.77E-05
Se75	3.43E-30	Sn121	1.35E-01	Gd152	5.88E-17
Se79	2.66E-06	Sn121m	1.75E-01	Gd153	1.50E-14
Kr81	6.13E-12	Sn123	6.95E-20	Eu154	1.46E-03
Kr85	1.51E-02	Sn126	1.59E-05	Eu155	1.69E-04
Rb83	4.53E-35	Sb124	5.91E-06	Tb157	1.39E-10
Rb87	6.18E-10	Sb125	4.91E-04	Tb158	1.50E-09
Y88	1.48E-29	Sb126	2.22E-06	Dy159	2.56E-26
Y90	7.21E-01	Sb126m	1.59E-05	Tb160	6.78E-35
Y91	5.69E-13	Te121	1.41E-26	Ho163	1.16E-12
Sr90	7.21E-01	Te121m	1.42E-26	Ho166m	1.94E-09
Nb91	1.85E-09	Te123	5.86E-12	Tm168	3.17E-37
Nb92	5.29E-07	Te123m	4.95E-11	Tm170	1.48E-25
Nb93m	1.11E+00	Te125m	1.21E-04	Tm171	2.66E-12

Table A-53. 2008 Radionuclide Inventory for 105-R Reactor Vessel Volumetrically Contaminated 304 Stainless Steel - continued

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Nb94	2.71E+00	Te127	1.36E-23	Tl204	3.25E-08
Nb95	7.56E-34	Te127m	1.39E-23	Tl206	1.19E-04
Tl207	8.09E-09	Ra225	1.60E-12	Pu236	4.87E-12
Tl208	2.61E-08	Ra226	1.65E-09	Pu238	1.13E-02
Tl209	3.35E-14	Ra228	1.39E-14	Pu239	2.19E-02
Pb205	4.88E-05	Ac225	1.60E-12	Pu240	3.16E-02
Pb209	1.60E-12	Ac227	8.11E-09	Pu241	2.93E-01
Pb210	6.35E-10	Ac228	1.39E-14	Pu242	2.37E-05
Pb211	8.11E-09	Th227	8.01E-09	Pu243	3.64E-15
Pb212	7.25E-08	Th228	7.25E-08	Pu244	5.92E-14
Pb214	1.65E-09	Th229	1.60E-12	Pu246	1.02E-22
Bi208	2.80E-05	Th230	1.43E-07	Am241	6.83E-02
Bi210	1.13E-04	Th231	1.31E-05	Am242m	2.58E-04
Bi210m	6.31E-06	Th232	1.68E-14	Am242	2.56E-04
Bi211	8.11E-09	Th234	3.18E-04	Am243	4.20E-05
Bi212	7.25E-08	Pa231	1.54E-08	Am245	4.42E-24
Bi213	1.60E-12	Pa233	1.59E-06	Am246	1.02E-22
Bi214	1.65E-09	Pa234m	3.18E-04	Cm242	2.11E-04
Po210	6.35E-10	Pa234	4.14E-07	Cm243	2.15E-05
Po211	2.23E-11	U232	7.05E-08	Cm244	3.26E-04
Po212	4.65E-08	U233	2.50E-10	Cm245	1.62E-08
Po213	1.57E-12	U234	2.90E-04	Cm246	4.65E-09
Po214	1.65E-09	U235	1.31E-05	Cm247	3.64E-15
Po215	8.11E-09	U236	7.15E-06	Cm248	8.13E-15
Po216	7.25E-08	U237	7.00E-06	Cm250	4.07E-22
Po218	1.65E-09	U238	3.18E-04	Bk249	3.06E-19
At217	1.60E-12	U240	5.92E-14	Bk250	5.69E-23
Rn219	8.11E-09	Np235	4.57E-18	Cf249	1.53E-14
Rn220	7.25E-08	Np236	2.80E-13	Cf250	4.29E-14
Rn222	1.65E-09	Np237	1.59E-06	Cf251	4.50E-16
Fr221	1.60E-12	Np238	1.16E-06	Cf252	4.50E-16
Fr223	1.12E-10	Np239	4.20E-05	Es254	4.28E-26
Ra223	8.11E-09	Np240m	5.92E-14		
Ra224	7.25E-08	Np240	7.10E-17		

Note: Total 304 stainless steel mass = 3.740E+08 g.

Table A-54. 2008 Radionuclide Inventory for 105-R Reactor Vessel Volumetrically Contaminated Aluminum

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
H3	2.22E-04	Ar42	2.65E-10	Co60	1.04E+00
Be10	3.75E-06	K42	2.65E-10	Ni59	2.33E-10
C14	5.48E-03	Ca41	8.19E-18	Ni63	3.05E+00
Na22	6.89E-16	Ca45	3.36E-30	Zn65	7.21E-16
Si32	1.25E-05	V49	4.21E-25	Se79	1.43E-16
P32	1.25E-05	V50	5.32E-21		
Cl36	6.24E-14	Mn54	9.12E-14		
Ar39	1.99E-14	Fe55	5.15E-01		

Note: Total aluminum mass = 2.567E+06 g.

Table A-55. 2008 Radionuclide Inventory for 105-R Reactor Building Volumetrically Contaminated Concrete

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
H3	1.19E+04	Nb91	1.33E-15	Te125m	4.06E-08
Be10	7.39E-09	Nb92	7.85E-10	I129	3.39E-05
C14	1.61E-01	Nb93m	1.85E-03	Ba133	2.29E+01
Na22	6.96E-10	Nb94	3.58E-03	Cs137	1.02E+01
Si32	8.49E-13	Zr93	1.14E-06	Eu152	1.60E+00
P32	8.50E-13	Zr95	2.91E-05	Eu154	1.60E-01
Cl36	2.61E-05	Mo93	2.59E-03	Tl204	1.30E-11
Ar39	6.00E-02	Tc98	3.52E-16	Tl206	1.20E-07
Ar42	1.74E-13	Tc99	8.16E-04	Pb205	1.24E-08
K40	3.10E-03	Ru106	4.48E-04	Pb212	1.14E-04
K42	1.74E-13	Pd107	1.99E-11	Bi208	5.52E-08
Ca41	1.09E-01	Ag108	4.36E-03	Bi210m	1.20E-07
Ca45	6.79E-28	Ag108m	5.00E-02	Th232	2.95E-03
V49	1.04E-20	Ag109m	1.71E-12	U232	1.50E-04
V50	3.38E-11	Ag110	6.62E-21	U233	3.58E-02
Mn54	8.62E-15	Ag110m	4.86E-19	U234	3.61E-02
Fe55	2.32E-02	Cd109	1.71E-12	U235	7.81E-04
Co58	1.81E-06	Cd113	1.07E-09	U238	4.56E-02
Co60	2.15E+00	Cd113m	9.53E-03	Np237	6.42E-06
Ni59	3.24E-01	In115	6.18E-07	Pu239	9.04E-05
Ni63	2.91E+01	Sn119m	2.13E-16	Pu240	8.80E-05
Zn65	9.75E-26	Sn121	1.76E-04	Pu242	2.18E-06
Se79	8.89E-23	Sn121m	2.27E-04	Am241	9.68E-04
Sr90	5.94E-01	Sn126	1.81E-05	Am243	2.59E-04
Y88	9.68E-06	Sb125	1.66E-07		
Y90	1.16E-14	Te123	1.69E-17		

Table A-56. 2008 Radionuclide Inventory for 105-R Reactor Building Surface Contamination

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
H3	3.31E+03	Ru103	7.67E-06	Ce144	1.85E-05
Be10	2.71E-07	Ru106	8.29E-05	Pm144	4.16E-06
C14	1.64E+01	Rh102	3.30E-31	Pm146	6.07E-06
Na22	3.30E-01	Rh106	6.38E-24	Eu152	1.59E-01
Si32	1.21E-04	Pd107	8.45E-09	Eu154	9.00E-01
P32	1.21E-04	Ag108	1.80E-01	Eu155	1.09E-01
Cl36	3.60E+01	Ag108m	2.07E+00	Pb212	1.44E-05
Ar39	4.50E-01	Ag109m	2.18E-08	Ac228	4.51E-05
Ar42	9.99E-07	Ag110	2.40E-20	U233	2.61E-02
K40	8.40E-05	Ag110m	1.77E-18	U234	2.28E-02
K42	9.99E-07	Cd109	2.18E-08	U235	2.18E-02
Ca40	6.83E-01	Cd113	3.24E-14	U236	2.18E-02
Ca45	5.42E-27	Cd113m	1.18E+00	U238	2.26E-02
V49	4.36E-20	In115	7.53E-11	Np237	2.18E-02
V50	4.30E-16	Sn113	5.95E-06	Np239	1.58E-05
Mn54	1.98E-05	Sn119m	9.84E-17	Pu238	1.52E-01
Fe55	5.34E-01	Sn121	3.21E-09	Pu239	4.39E-01
Co57	3.65E-05	Sn121m	4.13E-09	Pu240	2.36E-02
Co58	6.81E-06	Sn126	3.70E-02	Pu241	5.56E+01
Co60	1.19E+02	Sb124	1.98E-05	Pu242	2.22E-02
Ni59	4.43E+00	Sb125	2.93E-02	Am241	3.03E-01
Ni63	1.97E+02	Sb126	2.13E-11	Am243	8.21E-05
Zn65	1.91E-01	Te125m	2.63E-18	Cm243	4.32E-05
Se79	3.74E-02	I129	6.39E-02	Cm244	4.32E-05
Sr90	5.47E+01	Ba133	4.82E-06	Cm245	8.79E-05
Y88	1.00E-01	Cs134	7.86E-06	Cm246	1.39E-04
Y90	1.43E-01	Cs135	2.14E-05	Cf249	9.58E-06
Nb94	1.97E-05	Cs137	5.49E+01	Cf251	4.53E-05
Nb95	1.13E-01	Ba137m	2.53E-01		
Tc99	9.98E-02	Ce141	3.42E-02		

Table A-57. 1998 Radionuclide Inventory for R-Reactor Seepage Basin (RRSB)

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
C14	4.00E-01	Cs137	3.37E+02	Pu240	2.00E+00
Na22	1.25E-01	Pm147	6.45E-05	Am241	6.17E+01
K40	1.32E+00	Eu154	3.68E-01	Cm243	1.11E+00
Co60	1.32E-01	Th228	5.70E-01	Cm244	1.11E+00
Sr90	8.80E+01	Th230	3.41E-01	Cm246	3.17E+01
Tc99	1.00E-01	U235	1.91E-02		
Ru103	8.42E-05	Np237	2.93E-02		
Ru106	8.42E-05	Pu238	4.00E-01		
I129	1.50E-04	Pu239	2.00E+00		

Table A-58. 2001 Radionuclide Inventory for P-Reactor Seepage Basins (PRSBs)

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
C14	1.17E+00	Pm147	8.08E-03	Pu238	4.80E-02
Na22	7.30E-04	Eu154	7.54E-03	Pu239	6.92E-02
K40	1.21E-01	Pb212	1.40E-01	Pu240	6.92E-02
Co60	6.97E-01	Bi214	4.53E-02	Am241	1.99E-02
Ni63	5.00E-01	Ac228	1.21E-01	Cm242	4.82E-04
Sr90	4.38E+00	Ra228	2.57E-02	Cm243	1.80E-03
Tc99	2.46E-02	Th228	3.95E-02	Cm244	1.80E-03
Sb125	2.95E-02	Th230	1.66E-02	Cm245	5.11E-04
Cs134	5.39E-03	Th232	3.52E-02	Cm246	5.11E-04
Cs137	9.28E+01	Np237	8.97E-03		

Table A-59. 2001 Radionuclide Inventory for K-Reactor Seepage Basin (KRSB)

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
C14	5.04E-02	Ra226	1.42E-02	Pu240	6.50E-02
Na22	5.01E-04	U233	1.30E-02	Am241	1.60E-02
Co60	6.10E-03	U234	1.30E-02	Am243	4.34E-03
Sr90	3.90E-01	U238	1.25E-02		
Cs137	4.70E+00	Pu238	3.51E-03		
Pm147	6.87E-05	Pu239	6.50E-02		

Table A-60. 2001 Radionuclide Inventory for L-Reactor Seepage Basin (LRSB)

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Na22	1.37E-02	Cs137	8.37E-01	U238	1.53E-01
K40	1.04E-01	Pm147	6.15E-01	Pu238	1.34E-01
Co57	2.88E-02	Pb212	1.50E-01	Pu239	9.37E-02
Co60	3.30E-01	Ra228	1.87E-01	Pu240	9.37E-02
Zn65	4.39E-02	Th230	7.59E-02		
Sr90	1.31E+00	Th234	1.76E-01		
Tc99	4.66E-02	U234	1.72E-01		

Table A-61. 1997 Radionuclide Inventory for C-Reactor Seepage Basins (CRSBs)

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
C14	1.48E+00	Sb124	3.40E-03	Ac228	8.14E-03
S35	7.70E-03	Sb125	3.40E-03	U235	3.55E-04
K40	8.95E-03	Cs134	1.50E-03	U238	3.25E-03
Cr51	1.60E-03	Cs137	1.20E+00	Pu238	4.96E-03
Co60	5.10E-02	Ce141	1.10E-03	Pu239	1.65E-02
Ni63	2.30E-01	Ce144	1.10E-03	Pu240	1.65E-02
Sr89	7.10E-06	Pm147	5.00E-03	Am241	1.34E-02
Sr90	2.50E-01	Eu152	3.86E-03	Am243	8.22E-04
Zr95 [1]	3.40E-04	Eu154	1.73E-03	Cm243	1.28E-03
Nb95	3.40E-04	Eu155	7.51E-04	Cm244	1.28E-03
Ru103	1.60E-04	Ra226	4.41E-03		
Ru106	1.60E-04	Ra228	8.86E-03		

[1] Hiergesell et al. 2008 Table 3-66 provided two inventory values for Zr95 (i.e., an inventory of 3.40E-04 Ci from WSRC 1997 and a lower calculated value of 2.32E-04 Ci based upon average soil concentrations from WSRC 1998). The lower calculated Zr95 inventory of 2.32E-04 Ci has been eliminated. Per Table 11-2, item 14 in Volume I, Hiergesell et al. (2008) will be revised.

Table A-62. 2007 Radionuclide Inventory for P-Reactor Cask Car Railroad Tracks

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
K40	9.45E-04	Pb214	6.30E-05
Co60	1.37E-05	Bi214	5.75E-05
Cs137	7.97E-03	Ac228	5.72E-05
Pb212	6.08E-05	U235	1.61E-05
Bi212	6.65E-05	Ra226	1.00E-04

Note: The inventories of the C-, K-, and L-Reactor cask car railroad tracks will be taken as that of the P-Reactor cask car railroad tracks.

Table A-63. 2007 Radionuclide Inventory for R-Reactor Cask Car Railroad Tracks

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
K40	2.32E-02	Th230	1.18E-03
Cs137	6.40E-01	U233	9.23E-04
Bi212	1.33E-03	U234	9.23E-04
Pb212	1.93E-03	U235	4.41E-03
Bi214	1.30E-03	U238	9.28E-04
Pb214	1.12E-03		

Table A-64. 2007 Radionuclide Inventory for Outfall P02

Radionuclide	Inventory (Ci)
Tl208	1.44E-04

Table A-65. 2007 Radionuclide Inventory for Outfall P007

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
K40	5.22E-02	Ra226	1.26E-02
Co60	7.81E-04	Th230	1.97E-02
Cs137	2.89E-01	U233	1.54E-02
Tl208	2.95E-03	U234	1.54E-02
Pb214	1.45E-02	U238	1.46E-02

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A.8 S-AREA FACILITIES

Table A-66. DWPF Component Expected End State and Residual Waste

DWPF Component	Expected End State (ES)	Assumed Residual at ES
221S Building	ISD (Grouting in place)	1000 gal sludge
LPPP	ISD (Grouting in place)	50 gal sludge
OWST	De-inventoried	None
GWSB#1	De-inventoried/Grouted	None
GWSB#2	De-inventoried/Grouted	None
GWSB#3	De-inventoried/Grouted	None
Sandfilter	ISD (Grouting in place)	10 gal sludge
FESV	De-inventoried	None
Used Melters	Off-Site Disposal	None
Canister Shipping Facility	De-inventoried/Grouted	None
DWPF Recycle Evaporator	Construction status undecided	None

Note: OWST – Organic Waste Storage Tank, GWSB – Glass Waste Storage Bldg., FESV – Failed Equipment Storage Vaults. These facilities will not have a significant amount of residual radionuclides. The Used Melters will be dispositioned offsite.

Table A-67. 2031 Radionuclide Inventory for DWPF

Radionuclide	221S Building Inventory (Ci)	LPPP Inventory (Ci)	Sandfilter Inventory (Ci)	Total Inventory (Ci)
Ag110m	1.7E-10	8.6E-12	1.7E-12	1.8E-10
Am241	1.8E+01	8.9E-01	1.8E-01	1.9E+01
Am242	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Am242m	2.2E-02	1.1E-03	2.2E-04	2.3E-02
Am243	9.9E-03	4.9E-04	9.9E-05	1.0E-02
Ba137m	1.5E+03	7.6E+01	1.5E+01	1.6E+03
C14	1.3E-06	6.6E-08	1.3E-08	1.4E-06
Cd113	6.5E-14	3.3E-15	6.5E-16	6.9E-14
Cd115m	7.7E-72	3.9E-73	7.7E-74	8.2E-72
Ce141	2.4E-96	1.2E-97	2.4E-98	2.6E-96
Ce142	1.6E-05	8.2E-07	1.6E-07	1.7E-05
Ce144	3.0E-06	1.5E-07	3.0E-08	3.2E-06
Cm242	5.3E-19	2.7E-20	5.3E-21	5.6E-19
Cm243	5.2E-03	2.6E-04	5.2E-05	5.5E-03
Cm244	1.1E-01	5.3E-03	1.1E-03	1.1E-01
Cm245	1.1E-05	5.7E-07	1.1E-07	1.2E-05
Cm246	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Cm247	1.1E-12	5.6E-14	1.1E-14	1.2E-12
Cm248	1.2E-12	5.8E-14	1.2E-14	1.2E-12
Co60	1.1E+01	5.3E-01	1.1E-01	1.1E+01
Cr51	9.2E-117	4.6E-118	9.2E-119	9.7E-117
Cs134	6.3E-02	3.2E-03	6.3E-04	6.7E-02
Cs135	5.3E-03	2.6E-04	5.3E-05	5.6E-03
Cs137	1.6E+03	8.0E+01	1.6E+01	1.7E+03
Eu152	1.7E+00	8.7E-02	1.7E-02	1.8E+00
Eu154	1.4E+02	7.0E+00	1.4E+00	1.5E+02
Eu155	2.1E+01	1.0E+00	2.1E-01	2.2E+01
H3	1.5E-02	7.7E-04	1.5E-04	1.6E-02
I129	1.2E-02	6.2E-04	1.2E-04	1.3E-02
Nb94	9.4E-04	4.7E-05	9.4E-06	1.0E-03
Nb95	1.7E-81	8.3E-83	1.7E-83	1.8E-81
Nb95m	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Nd144	8.3E-10	4.1E-11	8.3E-12	8.8E-10
Ni59	2.4E+00	1.2E-01	2.4E-02	2.5E+00
Ni63	2.5E+02	1.2E+01	2.5E+00	2.6E+02
Np236	3.0E-08	1.5E-09	3.0E-10	3.2E-08
Np237	1.5E-02	7.6E-04	1.5E-04	1.6E-02
Pd107	1.6E-02	7.8E-04	1.6E-04	1.7E-02
Pm147	5.3E+01	2.6E+00	5.3E-01	5.6E+01
Pm148	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Pm148m	9.9E-77	5.0E-78	9.9E-79	1.1E-76
Pr144	3.0E-06	1.5E-07	3.0E-08	3.2E-06
Pr144m	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table A-67. 2031 Radionuclide Inventory for DWPF - continued

Radionuclide	221S Building Inventory (Ci)	LPPP Inventory (Ci)	Sandfilter Inventory (Ci)	Total Inventory (Ci)
Pu236	2.3E-04	1.2E-05	2.3E-06	2.5E-04
Pu237	2.9E-73	1.5E-74	2.9E-75	3.1E-73
Pu238	1.1E+03	5.3E+01	1.1E+01	1.1E+03
Pu239	1.2E+01	6.0E-01	1.2E-01	1.3E+01
Pu240	7.7E+00	3.8E-01	7.7E-02	8.1E+00
Pu241	4.3E+02	2.1E+01	4.3E+00	4.5E+02
Pu242	1.1E-02	5.3E-04	1.1E-04	1.1E-02
Rb87	1.6E-05	7.8E-07	1.6E-07	1.7E-05
Rh103m	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Rh106	1.0E-04	5.0E-06	1.0E-06	1.1E-04
Ru103	3.9E-79	1.9E-80	3.9E-81	4.1E-79
Ru106	1.0E-04	5.0E-06	1.0E-06	1.1E-04
Sb124	8.0E-54	4.0E-55	8.0E-56	8.4E-54
Sb125	2.5E+00	1.3E-01	2.5E-02	2.7E+00
Sb126	3.6E-02	1.8E-03	3.6E-04	3.8E-02
Sb126m	2.6E-01	1.3E-02	2.6E-03	2.7E-01
Se79	2.3E-01	1.2E-02	2.3E-03	2.5E-01
Sm147	3.4E-06	1.7E-07	3.4E-08	3.6E-06
Sm148	9.8E-12	4.9E-13	9.8E-14	1.0E-11
Sm151	3.4E+02	1.7E+01	3.4E+00	3.7E+02
Sn121m	3.4E-02	1.7E-03	3.4E-04	3.6E-02
Sn123	1.5E-22	7.4E-24	1.5E-24	1.6E-22
Sn126	2.6E-01	1.3E-02	2.6E-03	2.7E-01
Sr89	1.1E-59	5.7E-61	1.1E-61	1.2E-59
Sr90	2.8E+04	1.4E+03	2.8E+02	3.0E+04
Tb160	7.7E-45	3.8E-46	7.7E-47	8.1E-45
Tc99	4.3E+00	2.1E-01	4.3E-02	4.5E+00
Te125m	5.7E-01	2.9E-02	5.7E-03	6.1E-01
Te127	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te127m	5.2E-27	2.6E-28	5.2E-29	5.5E-27
Te129	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Te129m	1.4E-94	6.9E-96	1.4E-96	1.5E-94
Tl208	0.0E+00	0.0E+00	0.0E+00	0.0E+00
U232	1.1E-01	5.7E-03	1.1E-03	1.2E-01
U233	1.7E-05	8.7E-07	1.7E-07	1.8E-05
U234	4.6E-01	2.3E-02	4.6E-03	4.9E-01
U235	1.5E-03	7.6E-05	1.5E-05	1.6E-03
U236	3.3E-02	1.7E-03	3.3E-04	3.5E-02
U238	8.5E-03	4.3E-04	8.5E-05	9.0E-03
Y90	2.8E+04	1.4E+03	2.8E+02	3.0E+04
Y91	5.8E-51	2.9E-52	5.8E-53	6.2E-51
Zr93	1.9E+00	9.7E-02	1.9E-02	2.1E+00
Zr95	7.6E-46	3.8E-47	7.6E-48	8.1E-46

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A.9 T-AREA (FORMER TNX AREA)

Table A-68. 1997 Radionuclide Inventory for the TNX Burial Ground

Radionuclide	Inventory (Ci)
Ac228	1.16E-02
Cs137	1.37E-04
Pb212	1.18E-02
K40	5.37E-02
Ra226	8.11E-03
Ra228	1.25E-02
Na22	3.04E-06
Th228	1.20E-02
Th230	1.08E-02
Th232	1.03E-02
Th234	2.47E-02
U233	9.47E-02
U234	9.47E-02
U235	5.22E-03
U238	9.48E-02

Table A-69. 1997 Radionuclide Inventory for the TNX OTSB /IPSL

Radionuclide	Inventory (Ci)
Ac228	2.04E-02
Cs137	1.85E-03
Co57	1.05E-05
Cm242	4.40E-05
Cm245	8.67E-06
Cm246	8.67E-06
Pb212	2.13E-02
Pu238	1.69E-04
Pu239	1.07E-04
Pu240	1.07E-04
K40	5.12E-02
Pm146	5.63E-05
Ra226	7.11E-03
Ra228	3.24E-02
Ru106	7.50E-05
Sr90	1.02E-03
Th228	3.03E-02
Th230	9.59E-03
Th232	2.90E-02
U233	1.14E-01
U234	1.14E-01
U235	6.99E-03
U238	1.37E-01

Table A-70. 1997 Radionuclide Inventory for TNX Discharge Gully

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Ac228	1.45E-03	Pm146	2.13E-06
Cs137	3.35E-05	Ra226	6.41E-04
Co57	1.43E-06	Ra228	1.76E-03
Cm242	1.89E-05	Sr90	6.09E-05
Cm243	9.13E-05	Th228	2.30E-03
Cm244	9.13E-05	Th230	1.31E-03
Cm245	4.02E-06	Th232	2.29E-03
Cm246	4.02E-06	Th234	1.56E-02
Pb212	1.50E-03	U233	1.95E-02
Pu238	4.65E-05	U234	1.95E-02
Pu239	1.77E-05	U235	9.32E-04
Pu240	1.77E-05	U238	2.02E-02
K40	4.04E-03		

Table A-71. 1997 Radionuclide Inventory for TNX Outfall Delta and Swamp

Radionuclide	Inventory (Ci)
Ac228	1.07E-01
Cs137	2.86E-02
Co60	1.23E-05
Cm242	6.62E-05
Cm243	2.19E-02
Cm244	2.19E-02
Cm245	3.07E-04
Cm246	3.07E-04
Pb212	1.07E-01
Pu238	1.11E-03
Pu239	1.39E-03
Pu240	1.39E-03
K40	4.64E-01
Pm146	8.60E-06
Ra226	7.33E-02
Ra228	1.11E-01
Ru106	2.18E-04
Sr90	1.10E-02
Th228	1.31E-01
Th230	6.82E-02
Th232	1.09E-01
Th234	1.87E-01
U233	1.51E-01
U234	1.51E-01
U235	1.01E-02
U238	1.61E-01

Table A-72. 2004 Radionuclide Inventory for Buildings 677-T and 678-T

Radionuclide	Inventory (Ci)
Ac228	9.08E-03
Ce144	N/A
Cs137	1.49E-04
Co58	N/A
Co60	2.58E-06
Cm242	1.39E-05
Cm243	9.78E-05
Cm244	9.78E-05
Cm245	6.61E-06
Cm246	6.61E-06
Pb212	8.17E-03
Pu238	4.04E-05
Pu239	1.93E-05
Pu240	1.93E-05
K40	3.00E-03
Pm146	1.80E-06
Ra226	5.35E-04
Ra228	9.62E-03
Ru106	4.58E-05
Sr90	8.31E-05
Th228	9.62E-03
Th230	1.29E-03
Th232	1.00E-02
Th234	1.76E-02
U233	1.11E-02
U234	1.12E-02
U235	8.65E-04
U236	1.73E-06
U238	1.82E-02
Zn65	N/A

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A.10 Z-AREA FACILITIES – SALTSTONE FACILITY

Table A-73. 2030 Radionuclide Inventory for Saltstone Vault 1

Radionuclide[1]	Inventory (Ci)	Radionuclide[1]	Inventory (Ci)
Am241	4.7E-04	Pu242	9.0E-04
Ba137m	4.1E+00	Ra226	6.4E-07
C14	1.3E+00	Rh106	1.5E-10
Cl36	7.6E-04[2]	Ru106	1.5E-10
Co60	8.2E-05	Sb125	1.6E-01
Cs137	4.3E+00	Sb126	1.4E-01
Eu152	1.8E-03	Sb126m	1.0E+00
Eu154	2.3E-04	Se79	3.0E-01
H3	6.1E+00	Sn126	1.0E+00
I129	1.1E-01	Sr90	6.9E-03
K40	7.6E-04[2]	Tc99	1.1E+02
Nb93m	2.5E-01	Te125m	3.8E-02
Nb94	2.5E-03	Th229	3.0E-01
Ni59	3.5E-02	Th230	4.1E-01
Ni63	7.8E-01	U233	2.8E-01
Np237	4.5E-03	U234	2.8E-01
Pd107	1.9E-03	U235	3.2E-03
Pt193	3.7E-01	U236	3.2E-03
Pu238	7.8E-03	U238	7.4E-03
Pu239	1.2E-02	Y90	6.9E-03
Pu240	1.2E-02	Zr93	2.5E-01
Pu241	9.8E-03		

1 Due to the smaller number of radionuclides analyzed, the Vault 1 inventory contains fewer radionuclides than the other disposal units.

2 These values have been updated from Hiergesell et al. (2008), to be consistent with the most recent estimated closure inventory for the SDF (SRNS 2009). Per Table 11-2, item 14 in Volume I, Hiergesell et al. (2008) will be revised.

Table A-74. 2030 Radionuclide Inventory for Saltstone Vault 4

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Ac227	1.6E-05	Pa231	9.3E-05
Al26	3.4E-01	Pd107	5.0E-02
Am241	5.6E+00	Pm147	5.5E-02
Am242m	2.8E-03	Pr144	7.2E-11
Am243	1.6E-02	Pt193	1.0E+01
Ba137m	2.8E+05	Pu238	9.1E+03
Bk249	1.8E-28	Pu239	3.8E+02
C14	2.7E+01	Pu240	1.2E+02
Ce144	7.2E-11	Pu241	2.4E+03
Cf249	6.5E-13	Pu242	8.1E-01
Cf251	2.4E-01	Pu244	3.6E-03
Cf252	1.8E-18	Ra226	3.6E-06
Cl36	3.0E-03[1]	Ra228	1.6E-06
Cm242	5.4E-19	Rh106	9.1E-07
Cm243	4.5E-02	Ru106	9.1E-07
Cm244	2.8E+00	Sb125	5.8E-01[1]
Cm245	4.2E-03	Sb126	9.0E-01
Cm247	1.2E-13	Sb126m	6.4E+00
Cm248	1.2E-13	Se79	1.4E+00
Co60	4.6E-01	Sm151	4.2E+01
Cs134	5.2E-01	Sn126	6.4E+01[1]
Cs135	5.4E+00	Sr90	4.6E+04
Cs137	3.0E+05	Tc99	5.8E+02
Eu152	9.7E-02	Te125m	1.4E-01
Eu154	1.4E+00	Th229	4.8E+00[1]
Eu155	1.0E-01	Th230	7.4E+00[1]
H3	2.6E+02	Th232	2.7E-05
I129	2.8E-01	U232	8.8E-03
K40	3.0E-03[1]	U233	4.6E+00
Na22	1.5E-01	U234	5.1E+00
Nb93m	1.3E+00	U235	2.1E-01
Nb94	1.3E-02	U236	1.7E-01
Ni59	5.2E-02	U238	5.9E-01
Ni63	2.2E+01	Y90	4.6E+04
Np237	6.1E-01	Zr93	1.3E+00

[1] These values have been updated from Hiergesell et al. (2008), to be consistent with the most recent estimated closure inventory for the SDF (SRNS 2009). Per Table 11-2, item 14 in Volume I, Hiergesell et al. (2008) will be revised.

Table A-75. 2030 Radionuclide Inventory for 64 Saltstone Disposal Cells – Vaults 1 and 4 excluded

Radionuclides	Inventory[1] (Ci)	Radionuclides	Inventory[1] (Ci)
Ac227	1.1E-05	Pa231	6.3E-05
Al26	1.2E+01	Pd107	3.6E-01
Am241	9.0E+01[2]	Pm147	4.9E+00[2]
Am242m	3.8E-02	Pr144	2.3E-08
Am243	2.4E+00	Pt193	7.0E+01[2]
Ba137m	1.4E+03	Pu238	1.1E+04
Bk249	1.2E-26	Pu239	9.6E+02[2]
C14	1.3E+02	Pu240	2.6E+02
Ce144	2.3E-08	Pu241	2.7E+03
Cf249	4.3E-11	Pu242	2.5E-01
Cf251	1.5E-12	Pu244	1.0E-03
Cf252	1.2E-16	Ra226	5.0E-05
Cl36	2.7E-02[2]	Ra228	5.6E-03
Cm242	4.0E-17	Rh106	7.7E-05[2]
Cm243	1.3E-02[2]	Ru106	7.7E-05[2]
Cm244	6.1E+01	Sb125	1.5E+01
Cm245	1.5E-02	Sb126	7.7E+01[2]
Cm247	4.5E-12	Sb126m	5.2E+02[2]
Cm248	4.7E-12	Se79	9.0E+01[2]
Co60	3.5E+00[2]	Sm151	3.8E+03
Cs134	9.6E-04[2]	Sn126	5.2E+02[2]
Cs135	8.3E-03[2]	Sr90	2.4E+03
Cs137	1.5E+03	Tc99	3.5E+04[2]
Eu152	6.3E+00	Te125m	3.7E+00
Eu154	1.2E+02	Th229	2.5E+00[2]
Eu155	8.3E+00	Th230	1.2E+01[2]
H3	1.9E+03	Th232	9.0E-02
I129	2.4E+01	U232	2.0E-02
K40	2.7E-02[2]	U233	2.4E+00
Na22	4.4E+00	U234	8.3E+00
Nb93m	2.4E+01	U235	1.9E-01
Nb94	2.4E-01	U236	1.0E+00
Ni59	5.4E+00	U238	6.4E+00[2]
Ni63	1.5E+02	Y90	2.4E+03
Np237	3.2E+00	Zr93	2.4E+01

1 The estimated closure inventory is for all 64 projected cells; therefore, the per-cell values in SRNS (2009) were multiplied by 64 for this table.

2 These values have been updated from Hiergesell et al. (2008), to be consistent with the most recent estimated closure inventory for the SDF (SRNS 2009). Per Table 11-2, item 14 in Volume I, Hiergesell et al. (2008) will be revised.

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A.11 GENERAL SITE AREAS

Table A-76. 2002 Radionuclide Inventory for Integrator Operable Units (IOU)

Integrator Operable Unit (IOU)	Cs137 Inventory (Ci)
Fourmile Branch IOU	1.54E+02
Lower Three Runs IOU	6.44E+02
Pen Branch IOU	1.30E+01
SR and Floodplain Swamp IOU	4.30E+02
Steel Creek IOU	1.20E+02
Upper Three Runs IOU	4.10E+01

Table A-77. 2002 Radionuclide Inventory for the Groundwater Operable Units (OU)

Groundwater Operable Unit (OU)	H3 Inventory (Ci)
C-Area Groundwater OU	1.1E+03
D-Area Groundwater OU	3.6E+02
F-Area Groundwater OU	9.0E+02
H-Area Groundwater OU	2.4E+01
K-Area Reactor Groundwater OU	2.4E+05
L-Area Northern and Southern Groundwater OU (combined)	3.58E+03[1]
MWMF Groundwater OU	4.16E+05[2]
P-Area Reactor Groundwater OU	1.6E+03
R-Area Reactor Groundwater OU	1.6E+00
T-Area (TNX) Groundwater OU	2.0E-01

[1] The L-Area Northern and Southern Groundwater OUs have been combined and the value has been updated from Hiergesell et al. (2008) to be consistent with the most recent groundwater tritium inventory (WSRC 2006).

[2] Tritium in the MWMF OU was inadvertently omitted from Hiergesell et al. (2008). The MWMF Groundwater OU tritium inventory was obtained from ERD 2001. The MWMF Groundwater OU tritium inventory includes contamination originating from the MWMF, LLRWDF, and ORWBG waste sites.

Table A-78. 1997 Radionuclide Inventory Associated with Identified Spills

Radio-nuclide	FTF Spill Inventory (Ci)			HTF Spill Inventory (Ci)				
	Spill at Tank 3	Spill at Tank 8	Spill at 281-3F	Spill at Tank 13	Spill at Tank 9	Spill at Tank 16	Spill at Tank 37	Spill at 281-3H
H3	---	---	---	---	---	5.00E-02	8.41E-02	---
Se79	1.73E-04	1.24E-02	4.65E-05	1.46E-03	2.57E-04	3.86E-03	6.51E-03	1.39E-04
Sr90	2.64E+01	1.89E+03	7.09E+00	2.23E+02	3.92E+01	2.00E+00	3.36E+00	2.13E+01
Zr93	8.21E-04	5.90E-02	2.21E-04	6.96E-03	1.22E-03	1.82E-02	3.09E-02	6.62E-04
Tc99	6.21E-03	4.46E-01	1.67E-03	5.26E-02	9.24E-03	1.39E-01	2.34E-01	5.01E-03
Ru106	2.39E-01	1.72E+01	6.43E-02	2.02E+00	3.55E-01	5.33E+00	9.00E+00	1.93E-01
Pd107	4.78E-05	3.43E-03	1.29E-05	4.05E-04	7.11E-05	1.07E-03	1.80E-03	3.86E-05
Ag110m	7.39E-05	5.30E-03	1.99E-05	6.26E-04	1.10E-04	1.65E-03	2.78E-03	5.96E-05
Sn126	2.37E-04	1.70E-02	6.38E-05	2.01E-03	3.53E-04	5.29E-03	8.93E-03	1.91E-04
Sb125	3.04E-05	2.18E-03	8.18E-06	2.58E-04	4.52E-05	6.79E-04	1.14E-03	2.45E-05
Te125m	1.22E-01	8.76E+00	3.28E-02	1.03E+00	1.82E-01	2.72E+00	4.60E+00	9.85E-02
I129	1.61E-04	1.16E-02	4.33E-05	1.37E-03	2.40E-04	3.60E-03	6.07E-03	1.30E-04
Cs134	3.64E+00	2.61E+02	9.79E-01	3.08E+01	5.41E+00	8.13E+01	1.37E+02	2.94E+00
Cs135	1.24E-04	8.92E-03	3.34E-05	1.05E-03	1.85E-04	2.77E-03	4.68E-03	1.00E-04
Cs137	3.72E+01	2.67E+03	1.00E+01	3.15E+02	5.53E+01	8.30E+02	1.40E+03	3.00E+01
Ce144	6.56E-02	4.71E+00	1.76E-02	5.56E-01	9.76E-02	1.46E+00	2.47E+00	5.29E-02
Pm147	3.37E+00	2.42E+02	9.05E-01	2.85E+01	5.01E+00	7.51E+01	1.27E+02	2.72E+00
Sm151	5.04E-01	3.62E+01	1.36E-01	4.27E+00	7.49E-01	1.12E+01	1.90E+01	4.07E-01
Eu154	1.97E+00	1.41E+02	5.29E-01	1.67E+01	2.93E+00	4.39E+01	7.41E+01	1.59E+00
Eu155	7.08E-02	5.08E+00	1.90E-02	6.00E-01	1.05E-01	1.58E+00	2.67E+00	5.71E-02
Pu239	---	---	---	---	---	2.00E-01	3.36E-01	---

A.12 FUTURE SRS OPERATIONAL FACILITIES

Table A-79. 2031 Radionuclide Inventory for Salt Waste Processing Facility

Radio-nuclide	Alpha Sorption Tank-A (AST-A) Inventory (Ci)	Filter Feed Tank-A (FFT-A) Inventory (Ci)	Sludge Solids Receipt Tank (SSRT) Inventory (Ci)	Salt Solution Feed Tank (SSFT) Inventory (Ci)	Strip Effluent Hold Tank Inventory (Ci)	Filter Feed Tank-B (FFT-B) Inventory (Ci)	Total Inventory (Ci)
Al26	5.20E-06	5.20E-06	4.20E-08	9.40E-07	--	--	1.14E-05
Am241	1.70E-06	1.70E-06	1.20E-08	3.10E-07	--	3.60E-04	3.64E-04
Am242m	2.70E-06	2.70E-06	2.00E-08	5.00E-07	--	5.90E-04	5.96E-04
Am243	2.90E-06	2.90E-06	2.00E-08	5.20E-07	--	6.30E-04	6.36E-04
Ba137m	1.40E+01	1.40E+01	6.50E-02	2.50E+00	1.80E+01	--	4.86E+01
C14	1.20E-05	1.20E-05	1.00E-07	2.20E-06	--	--	2.63E-05
Ce144	2.90E-14	2.90E-14	2.40E-16	5.30E-15	--	--	6.35E-14
Cf249	1.20E-05	1.20E-05	9.50E-08	2.10E-06	--	--	2.62E-05
Cf251	1.20E-05	1.20E-05	9.80E-08	2.20E-06	--	--	2.63E-05
Cf252	1.20E-10	1.20E-10	9.50E-13	2.10E-11	--	--	2.62E-10
Cm242	3.90E-24	3.90E-24	3.20E-26	7.10E-25	--	--	8.54E-24
Cm243	8.70E-06	8.70E-06	7.30E-08	1.60E-06	--	--	1.91E-05
Cm244	6.60E-05	6.60E-05	5.20E-07	1.20E-05	--	--	1.45E-04
Co60	3.70E-07	3.70E-07	3.00E-09	6.70E-08	--	--	8.10E-07
Cs137	1.50E+01	1.50E+01	6.90E-02	2.60E+00	2.00E+01	--	5.27E+01
Eu152	1.90E-05	1.90E-05	1.60E-07	3.50E-06	--	--	4.17E-05
Eu154	1.20E-05	1.20E-05	9.40E-08	2.10E-06	--	--	2.62E-05
Eu155	9.60E-07	9.60E-07	7.90E-09	1.80E-07	--	--	2.11E-06
H3	1.00E-03	1.00E-03	8.30E-06	1.80E-04	--	--	2.19E-03
I129	4.30E-06	4.30E-06	3.40E-08	7.80E-07	--	--	9.41E-06
Nb94	9.90E-06	9.90E-06	8.10E-08	1.80E-06	--	--	2.17E-05
Ni59	1.40E-05	1.40E-05	1.20E-07	2.60E-06	--	--	3.07E-05
Ni63	1.20E-04	1.20E-04	1.00E-06	2.20E-05	--	--	2.63E-04
Np237	1.80E-06	1.80E-06	1.40E-08	3.20E-07	--	1.30E-04	1.34E-04
Pa231	1.90E-04	1.90E-04	1.50E-06	3.40E-05	--	--	4.16E-04
Pm147	3.50E-06	3.50E-06	2.80E-08	6.40E-07	--	--	7.67E-06
Pu238	4.80E-03	4.80E-03	2.80E-05	8.80E-04	--	2.30E+00	2.31E+00
Pu239	8.20E-05	8.20E-05	4.80E-07	1.50E-05	--	4.10E-02	4.12E-02
Pu240	8.20E-05	8.20E-05	4.80E-07	1.50E-05	--	4.10E-02	4.12E-02
Pr144	2.90E-14	2.90E-14	2.40E-16	5.30E-15	--	--	6.35E-14
Rh106	2.00E-11	2.00E-11	1.60E-13	3.60E-12	--	--	4.38E-11
Ru106	2.00E-11	2.00E-11	1.60E-13	3.60E-12	--	--	4.38E-11
Sb125	9.50E-07	9.50E-07	7.70E-09	1.70E-07	--	--	2.08E-06
Sb126	3.80E-06	3.80E-06	3.10E-08	7.00E-07	--	--	8.33E-06
Se79	1.60E-04	1.60E-04	1.30E-06	3.00E-05	--	--	3.51E-04
Sm151	3.10E-03	3.10E-03	2.50E-05	5.70E-04	--	--	6.80E-03
Sn126	2.70E-05	2.70E-05	2.20E-07	5.00E-06	--	--	5.92E-05
Sr90	3.50E-04	3.50E-04	1.80E-06	6.40E-05	--	2.20E-01	2.21E-01
Tc99	5.30E-02	5.30E-02	4.30E-04	9.60E-03	--	--	1.16E-01
U232	4.30E-06	4.30E-06	3.10E-08	7.90E-07	--	6.50E-04	6.59E-04
U235	9.30E-08	9.30E-08	6.90E-10	1.70E-08	--	1.40E-05	1.42E-05
U238	2.70E-08	2.70E-08	2.00E-10	5.00E-09	--	4.10E-06	4.16E-06
Y90	3.50E-04	3.50E-04	1.80E-06	6.40E-05	--	2.20E-01	2.21E-01

Table A-80. 2025 Radionuclide Inventory for MOX Facility

Radionuclide	Inventory (Ci)	Inventory (Ci)	Inventory (Ci)	Inventory (Ci)
Pu236	2.29E+00	2.29E+00	2.29E+00	2.29E+00
Pu238	3.70E+01	3.70E+01	3.70E+01	3.70E+01
Pu239	2.47E+02	2.47E+02	2.47E+02	2.47E+02
Pu240	6.87E+01	6.87E+01	6.87E+01	6.87E+01
Pu241	4.45E+03	4.45E+03	4.45E+03	4.45E+03
Am241	1.49E+02	1.49E+02	1.49E+02	1.49E+02
U232	9.08E-06	9.08E-06	9.08E-06	9.08E-06
U234	6.63E-02	4.24E-02	3.09E-02	2.44E-02
U235	9.09E-04	5.81E-04	4.24E-04	3.35E-04
U236	2.70E-02	1.73E-02	1.26E-02	9.96E-03
U238	6.14E-02	3.93E-02	2.87E-02	2.26E-02

A.13 REFERENCES

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B. APPENDIX B: COMPOSITE ANALYSIS REVIEW CRITERIA MATRIX

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008)

Criteria	Where to Find (CA Report Sections)
Criterion IV.P.(3) from *Radioactive Waste Management Manual, Chapter 4 Low-Level Waste Requirements,* DOE M 435.1-1, 7/09/99	
IV. P.(3) Composite Analysis. For disposal facilities which received waste after September 26, 1988, a site-specific radiological composite analysis shall be prepared and maintained that accounts for all sources of radioactive material that may be left at the DOE site and may interact with the low-level waste disposal facility, contributing to the dose projected to a hypothetical member of the public from the existing or future disposal facilities. Performance measures shall be consistent with DOE requirements for protection of the public and environment and evaluated for a 1,000 year period following disposal facility closure. The composite analysis results shall be used for planning, radiation protection activities, and future use commitments to minimize the likelihood that current low-level waste disposal activities will result in the need for future corrective or remedial actions to adequately protect the public and the environment.	**Whole document** – Both volumes represent the results and documentation of the site-specific radiological composite analysis (CA) carried out for the Savannah River Site (SRS). This analysis accounts for sources of radioactive material potentially remaining at the SRS at closure which may contribute to projected dose to a hypothetical member of the public from existing and future disposal facilities. The performance measures are consistent with Department of Energy (DOE) requirements for a composite analysis, and are evaluated for a 1,000-year period following closure. The results are presented in a manner that will be used in planning for future use commitments of the site.
Criteria 3.3.1 through 3.3.11.1 from *Low-Level Waste Disposal Facility Federal Review Group Manual*, Revision 3, June 2008	
3.3.1 Site and Facility Characteristics	
3.3.1.1 The CA provides a coherent presentation of the relevant descriptive information concerning the disposal site, its location on the DOE site, and its proximity to other sources of radioactive material. The sources of radioactive material are described including relevant features that could influence radionuclide release and migration.	**3.1 SRS Facility Descriptions and Operational History** (principally) In this section of the CA, a historical overview of all facilities at the SRS is given, including the disposal operations that are of concern to both the Performance Assessment (PA) and CA process. This historical overview provides a geographical and operational perspective on the origination of the sources of radionuclides that are disposed in the facilities today, and on other residual radioactive material at the site. **3.4 Savannah River Site Performance Assessments** The discussion in this section provides facility and radionuclide information from SRS PAs that have been completed, are in progress, or are being considered. The sources of radioactive material and relevant features of the disposal facility that could influence release and migration of radionuclides are reviewed. **8.2 Radionuclide Source Locations and Inventory** This section provides descriptions of sources of radioactive material on the site considered potentially significant in the CA, including location, and provides estimated inventories for each source.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.2 Radioactive Sources/Release Mechanism	
3.3.2.1 The CA identifies all sources of radioactive material in the ground that could contribute to the potential future doses from the LLW disposal facility. Sources selected for the CA and the reasons for excluding any source are justified. Potential sources of radioactive material to be considered include wastes disposed of prior to 1988, other LLW disposal facilities, transuranic waste or alpha LLW disposal, buildings, tanks, cribs, spills, ditches, seepage basins, and leaks.	**8.1 Source Terms Approach** (principally) This CA section identifies all radionuclide source locations over the entire SRS site that are projected to have a remaining radionuclide inventory after all DOE site operations have ceased and that could interact with the waste disposal facilities covered by the PA process. The source locations identified include future facilities included in DOE long-range plans for the SRS. The justification for using a screening tool to evaluate the potential for contribution of radionuclides to groundwater for each source is given. **3.1 SRS Facility Descriptions and Operational History** The historical overview of all facilities at the SRS provided in this section contributes to the understanding of the origination of the sources of radionuclides that are disposed in the waste disposal facilities today, of the other residual radioactive material at the site, and of the potential relationship between the facilities and residual radioactive material.
3.3.2.2 The CA identifies and quantifies all radionuclides present in the LLW disposal facility and all other contributing sources of radioactive material that could contribute significantly to the total potential dose. Inventory estimates included in the analysis are justified. The estimates of radionuclide species and inventories in the sources selected for consideration are derived from referenced documentation or data summaries presented in the CA and are based on existing records, process knowledge, or site investigations (e.g., Remedial Investigations, Feasibility Studies). Any radionuclides that are screened from the analysis are identified and their exclusion justified as being insignificant contributors to the total dose estimated in the analysis.	**8.2 Radionuclide Source Locations and Inventory** (principally) This section provides descriptions of all sources of radioactive material on the site considered potentially significant in the CA, including location, provides estimated inventories for each source, and provides a justification for the estimates given. The estimates are based on existing knowledge, process knowledge, and site investigations. **7.2 Radionuclide Screening** This section identifies, as a result of screening calculations, the radionuclides potentially significant in dose calculations, beginning with a list of 849 isotopes. Justification for screening out radionuclides is given in this section. **8.1 Source Terms Approach** This CA section identifies all radionuclide source locations over the entire SRS site that are projected to have a remaining radionuclide inventory after all DOE site operations have ceased, that could interact with the waste disposal facilities covered by the PA process, and that could significantly contribute to the total potential dose.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.2 Radioactive Sources/Release Mechanism (continued)	
3.3.2.3 The known physical and chemical characteristics of the radioactive materials considered in the CA, the site characteristics, and the effects of the Comprehensive Environmental Responses, Compensation, and Liability Act (CERCLA) actions prescribed in the Record of Decisions (RODs) or similar binding agreements such as those associated with Decontamination and Decommissioning (D&D), are included in the generation of the source terms and the transport of the radionuclides. Extrapolations are made and justified from known data to estimate radionuclides and inventories where clear information does not exist.	**8.2 Radionuclide Source Locations and Inventory** (principally) This section provides descriptions of the sources of radioactive material on the site considered potentially significant in the CA in relation to how the contamination originated and to D&D activities under CERCLA and other regulatory drivers. Estimated inventories for each source are based on, extrapolated from, and justified on the basis of existing knowledge, process knowledge, and site investigations. **3.2 Savannah River Site and Facility Characteristics** Site characteristics, especially groundwater and surface water hydrology of the SRS, are provided in this section for the purpose of supporting the radionuclide transport analyses carried out in the CA. **9.3.1 Disposal Unit Module** This section describes how the site characteristics and source term information are used in evaluating radionuclide transport for the CA. Physical and chemical characteristics of radionuclides are addressed in the base case modeling in terms of availability for transport (source release) and sorption during transport. Use of available PA source terms for disposal facilities incorporates the PA considerations of physical and chemical characteristics of radionuclides in these facilities.
3.3.2.4 Source terms and flow and transport models in the CA are commensurate with the available data consistent with the PA, incorporate the important characteristics identified in the PA, and provide outputs consistent with the PA.	**9.5 Relationship to Performance Assessment Modeling** (principally) This section addresses these criteria directly, pointing to the ways in which the CA relies on source terms and flow and transport models used in the PAs that have been completed or are currently in progress. **8.2 Radionuclide Source Locations and Inventory** For the PA facilities listed in this section, the basis of the inventory was the same data source as used in the PAs. **9.1 Conceptual Model and Modeling Philosophy** The consistency between PA conceptual models and the CA conceptual model is noted in this section **9.2.1 Modeling Codes and Existing Models** The General Separations Area (GSA) flow model is described and referenced in this section, which serves as the basis for the subsurface flow modeling in the PAs as well as in the CA. **9.3.1 Disposal Unit Module** This section describes how the PA radionuclide fluxes to the water table were utilized in the CA analysis, thus assuring consistency between PAs and CA with respect to source release modeling.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.3 Performance Measures	
3.3.3.1 The CA presents an assessment using the time of 1,000 years for exposures to hypothetical members of the public with all disposal facilities closed, D&D completed, and operations at the DOE site terminated. The assessment establishes a realistic case for comparison with the dose constraint (30 mrem/yr) and dose limit (100 mrem/yr).	**4 Analysis Criteria** (principally) The assessment period (AP) of 1,000 years after facility closure, completion of D&D, and termination of all site operations at the SRS for the CA is established in this section. In consideration of current plans in place (land use, waste site remediation, D&D, etc.), the AP is realistically defined as beginning in 2025 for the CA. **10.1 Deterministic Base Case** (from Performance Evaluation Section) The comparison of the results of the deterministic base case modeling (representing the most realistic assessment case) to the dose constraint of 30 mrem/yr and dose limit of 100 mrem/yr is made in this section.
3.3.4 Point of Assessment	
3.3.4.1 The point of assessment is the publicly accessible point of maximum impact reasonably expected for future members of the public for the time period of assessment. The point of assessment is justified and is supported by land use plans or reasonably conservative assumptions that are justified.	**4.2 Points of Assessment** The justification for the selection of the points of assessment (POAs) for the CA is given in this section. Criteria considered as well as important elements of SRS land use plans (SRS Long Range Comprehensive Plan and SRS End State Vision) are provided to support the selection.
3.3.4.2 Changes in the point of assessment as a function of time are justified.	N/A – The points of assessment do not change as a function of time. **4.2 Points of Assessment** The publicly-accessible locations of the POAs, described in this section, do not change over time.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.5 Assumptions	
3.3.5.1 Assumptions incorporated into the analysis, including those related to the radionuclides to be considered, the inventories of radionuclides, the source term evaluation and the transport of radionuclides, are identified, justified, and consistent with the conceptual model of site behavior presented in the PA conducted on the LLW disposal facility.	**9.3.1 Disposal Unit Module** (principally) This section describes how the site characteristics and source term information are used in evaluating radionuclide transport for the CA. Assumptions pertinent to source release and transport are identified and justified. Use of available PA source terms for disposal facilities provides a consistency with PA assumptions with respect to source release and vadose zone transport. Sorption is addressed using K_d's and CDP's consistent with the source of these parameters used in the PAs (although a few values are updated based on ongoing work at the SRS). **9.3.5 Model Validation** (principally) The results of model validation work is presented in this section, which support the assumption that GoldSim™ modeling methodology, used in both CA and PAs, produces transport results that are reasonably consistent with results of a 1-D application of PORFLOW, a model which has been used and extensively tested in support of SRS PA applications. **7.2 Radionuclide Screening** Assumptions made in the process of screening out radionuclides are described and justified in this section. The screening process began with a list of radionuclides that was identical to that used for the F-Area Tank Farm (FTF) PA and that included all radionuclides considered in the E-Area Low Level Waste Facility (ELLWF) PA, and the same exposure and dose equations were utilized for corresponding exposure pathways in both PAs and CA. Comparison of CA screening results with those for the E-Area Low Level Waste Facility (ELLWF) and the F-Area Tank Farm (FTF) PAs are provided in this section. **8.2 Radionuclide Source Locations and Inventory** This section provides descriptions of all sources of radioactive material on the site considered potentially significant in the CA and provides justification for estimated inventories for each source, including an explanation of assumptions made in developing the estimates.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.5 Assumptions (continued)	
3.3.5.1 (Continued)	**9.1 Conceptual Model and Modeling Philosophy** The assumptions underlying the conceptual model are described in this section. The consistency between PA conceptual models and the CA conceptual model is also noted. **9.2.1 Modeling Codes and Existing Models** The General Separations Area (GSA) flow model is described and referenced in this section, which serves as the basis for the subsurface flow modeling in the PAs as well as in the CA. **9.5 Relationship to Performance Assessment Modeling** This section points to the ways in which the CA relies on source terms and flow and transport models used in the PAs that have been completed or are currently in progress, and thus on the same assumptions used in developing the source terms and implementing flow and transport models. **Appendix F Key Assumptions** Key assumptions relevant to SRS operations and modeling are described in this appendix. Where pertinent, consistencies with PA assumptions are noted.
3.3.5.2 The CA identifies results, objectives, constraints, or milestones of other DOE programs, Federal, state, or local statutes, or agreements (e.g., D&D programs, Formerly Utilized Sites Remedial Action Program (FUSRAP), CERCLA RODs) that may impact the analysis or conclusions of the CA.	**3.5 Related Documentation** (principally) This section provides a summary of the site-specific documentation, including regulatory documentation, which may impact the analysis or conclusions of the CA. In addition to completed PAs and specific CERCLA, RCRA, D&D, and high-level tank closure documentation, this includes the SRS End State Vision, SRS Long Range Comprehensive Plan, SRS Strategic Plan, SRS Land Use Control Assurance Plan, SRS Federal Facilities Agreement, DOE Closure Planning Guidance, SRS Groundwater Protection Program, SRS Related NEPA Documentation, SRS Safety Basis Documentation, and SRS Annual Environmental Monitoring Reports. The purpose of each type of documentation is discussed, along with guidelines or regulations arising from the documentation that may impact the CA. **2.2 Major Changes Necessitating a Composite Analysis Revision** The impact of DOE programs and plans, as well as CERCLA and D&D actions, on the need to revise the original CA is described in this section, thus showing the impact of these programs/statutes/agreements on the CA.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.6 Modeling	
3.3.6.1 The CA presents a reasonable methodology for estimating the transport of radionuclides to the point of assessment from all sources based on and consistent with the available site data.	**9.1 Conceptual Model and Modeling Philosophy** (principally) The CA conceptual model described in this section is consistent with the background SRS hydrogeologic information presented in CA Section 3. The model description elucidates how source information and site data were incorporated into the model (see Table 9-1). The base case conceptual model (Section 9.1.3) reasonably focuses computational efforts on those radionuclides and sources most important with respect to dose, using less detailed and more conservative methods (i.e., generic release model) for sources determined to contribute insignificantly to dose. **9.3.1 Disposal Unit Module** This section describes how the site characteristics and source term information are used in a computational sense in evaluating radionuclide transport from the source to the vadose zone for the CA base case model. **9.3.4 Input Data** The input data for the base case model are described in this section, demonstrating the heavy reliance on available site data for both assumed inventory and parameters that affect transport. The consistency of the data used in the base case model with PAs and site data are demonstrated. **9.2.1 Modeling Codes and Existing Models** The suitability of the GoldSim™ modeling approach in the CA is addressed in this section. Briefly, since the POAs are in surface water, and thus concentrations in groundwater were not required, radionuclide fluxes through the vadose zone and aquifer to the streams/river could be represented in one-dimension. Particle track modeling was used with existing 3-dimensional models for different regions of the SRS, such that flow path lengths and flow rates could be derived for all sources.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.6 Modeling (continued)	
3.3.6.2 Analytical and numerical models selected are documented and verified either in referenced publications or in the appendices of the CA.	For codes: **6.1 Comprehensive Composite Analysis Quality Assurance Plan** The software quality assurance plan for the GoldSim™ code is cited in this section. This plan provides a reference to the extensive verification and validation testing that is available for this code. **9.2.1 Modeling Codes and Existing Models** This section provides brief descriptions and references for verified flow models and codes used in developing one-dimensional flow input for the GoldSim™ model. **9.2.2 GoldSim™ Composite Analysis Model Overview** Both the transport and dose module applications using GoldSim™ are summarized in this section, along with the appropriate reference to the particular GoldSim™ version used. **9.3.5 Model Validation** This section provides validation results that demonstrate the comparability of GoldSim™ with PORFLOW in simulating radionuclide transport in aquifers containing varying degrees of sand and clay. For models: **9.3.1 Disposal Unit Module** The analytical models used in describing source release are documented in this section. Where appropriate, references are made to supporting publications. **9.3.2 Dose Module** The models used for estimating exposure and dose (and implemented in the GoldSim™ Dose Module) are documented in this section, making appropriate reference to supporting publications.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.6 Modeling (continued)	
3.3.6.3 Any analytical and numerical models used in the CA for analyzing the transport of radionuclides to the point of assessment are appropriate for the LLW disposal facility and all other contributing sources. The models used in the CA provide calculated results that are representative of the results calculated in the PA for similar wastes in similar disposal facilities.	For codes: **9.2.1 Modeling Codes and Existing Models** (principally) The suitability of the GoldSim™ modeling approach in the CA is addressed in this section. Briefly, since the POAs are in surface water, and thus concentrations in groundwater were not required, radionuclide fluxes through the vadose zone and aquifer to the streams/river could be represented in one-dimension. Particle track modeling was used with existing 3-dimensional models for different regions of the SRS, such that flow path lengths and flow rates could be derived for all sources. **9.5 Relationship to Performance Assessment Modeling** (principally) The GoldSim™ model was used in the ELLWF, FTF and Saltstone Disposal Facility (SDF) PAs, and thus has been determined to be appropriate for the LLW disposal facilities **9.3.5 Model Validation** This section provides validation results that demonstrate the comparability of the GoldSim™ model and PORFLOW in simulating radionuclide transport in aquifers containing varying degrees of sand and clay. PORFLOW has been used in simulating 3-dimensional groundwater flow for PAs. For models: **9.3.1 Disposal Unit Module** The analytical models used in describing source release utilize site data that were utilized in the PAs. **9.3.2 Dose Module** The models used for estimating exposure and dose (and implemented in the GoldSim™ Dose Module) are identical to those used for PA. **9.5 Relationship to Performance Assessment Modeling** This section points to the ways in which the CA relies on source terms and flow and transport models used in the PAs that have been completed or are currently in progress, and thus on the same assumptions used in developing the source terms and implementing flow and transport models. The PA-modeled fluxes of each CA radionuclide into the aquifer at each time step for the E-Area IL Vault, the E-Area LAW Vault, and the SDF Vault 2 units were collected and used as direct input into the CA model and therefore represent the PA results exactly to that point.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.6 Modeling (continued)	
3.3.6.4 Credits for CERCLA/Resource Conservation & Recovery Act of 1976 (RCRA) actions or other actions (e.g., D&D, tank closures) are represented in the conceptual models used in the CA, and are justified by supporting or referenced documentation.	**9.3.1 Disposal Unit Module** (principally) This section addresses CERCLA/RCRA actions and other closure actions by incorporating the inventory information from Section 8, which includes consideration of these actions. These actions are also addressed in the infiltration estimates that account for the SRS end state for each radionuclide source at closure. **8.2 Radionuclide Source Locations and Inventory** This section considers the SRS end state, after CERCLA/RCRA and closure actions have occurred, in identifying which facilities will potentially retain a significant inventory. In addition, these actions are also considered in developing the inventory estimate for each of the identified facilities. Justification for these estimates, based on supporting documentation, is provided. **9.1.3 Base Case Conceptual Model** Table 9-1 in this section lists ways in which the source release flux relies on the factors associated with closure actions that affect infiltration, such as closure caps and cementitious enclosures or slabs.
3.3.6.5 The input data to the models are based on field data from the site, laboratory data interpreted for field applications, referenced literature sources which are applicable to the site, or related analyses performed for the PA. Any assumptions used to formulate input data are justified and have a defensible technical basis.	**9.3.4 Input Data (**principally) This section describes the source of input data, site-specific when available, and points to the tabulation of the data, much of which is available in Appendix C. Justification for selection of values is provided largely in the supporting documentation cited for each type of input. Input data for the CA model were taken from referenced sources based on site-specific field and laboratory studies and literature sources. Most of the data for facilities for which a PA has been done were taken from the same sources as those used for that PA or were compiled from PA input data sets. **9.4.2 Sensitivity Analysis** The basis of the data required for each sensitivity scenario is provided in this section. **9.4.3.2 Parameter Distributions** the sources of parameter distributions are discussed in this section, and supporting documentation is referenced. **7.2 Radionuclide Screening** The input data used in the screening calculations are described in this section, and appropriate supporting documentation for the values used is referenced.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.6 Modeling (continued)	
3.3.6.6 Intermediate calculations are performed, and the results are presented to demonstrate the CA calculations are representative of the site and are consistent with results presented in the PA for similar situations.	**9.5 Relationship to Performance Assessment Modeling** (principally) With the exception of the NRCDA 643-7E, isotope fluxes to the aquifer taken directly from the PA calculations were used in the CA model for PA facilities requiring source release modeling. This method ensures that the CA calculation is consistent with the PA as much as possible. Use of the PA flux accounts for source release factors in the CA without specific modeling, because the source release has already been taken into account when calculating the PA fluxes. **9.3.4.4 Aquifer** (Input Data Section) One-dimensional aquifer input data (travel distance and flow rates) were obtained from the aquifer flow analysis. This analysis involved using existing models (3-dimensional, or 3D) of groundwater flow within the SRS boundary to determine the path contaminants would follow from each of the CA sources to stream outcrops. These intermediate calculations provide data for the GoldSim™ model that are representative of the site in the various regions of interest, and provide consistency with the PA models in the General Separations Area for which the 3D models were developed. **9.3.5 Model Validation** This section provides validation results that demonstrate the comparability of the GoldSim™ model and PORFLOW in simulating radionuclide transport in aquifers containing varying degrees of sand and clay. PORFLOW has been used in simulating 3D groundwater flow for PAs.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.6 Modeling (continued)	
3.3.6.7 The conceptual model used for the CA is consistent with the representation of the conceptual model used in the PA, and includes the major mechanisms affecting the transport of radionuclides at the DOE site. The components of the conceptual model for the CA are reasonably represented in the analysis of the LLW disposal facility and other contributing sources.	**9.5 Relationship to Performance Assessment Modeling** (principally) This section points to the ways in which the CA relies on source terms and flow and transport models used in the PAs that have been completed or are currently in progress, and thus on the same assumptions used in developing the source terms and implementing flow and transport models. The PA-modeled fluxes of each CA radionuclide into the aquifer at each time step for the E-Area IL Vault, the E-Area LAW Vault, and the SDF Vault 2 units were collected and used as direct input into the CA model and therefore represent the PA results exactly to that point. The conceptual models for each of the facilities for which a PA has been completed have been appropriately considered within the CA model. Transport within the vadose and saturated zones for both the PAs and the CA is defined as K_d-mediated transport. **9.1 Conceptual Model and Modeling Philosophy** A brief discussion of the ELLWF, FTF, and SDF PA conceptual models can be found in Sections 3.4.1.5, 3.4.2.5, and 3.4.4.5, respectively, for comparison to that of the CA discussed within Section 9.1.3, and outlined in Table 9-1. These descriptions include the radionuclides analyzed, release mechanisms, the basis for the vadose zone and aquifer conceptual models, and the groundwater discharge point (i.e., either Upper Three Runs or Fourmile Branch).
3.3.7 Exposure Pathways and Dose Analysis	
3.3.7.1 The CA provides a complete discussion of all important exposure pathways for the evaluation of potential doses to a hypothetical, individual member of the public at the point of exposure for any time during the period of assessment. The exposure pathways identified in the CA should be consistent with the exposure pathways in the PA. The exposure pathways considered in the CA include only those pathways that are related to the exposure of individual members of the public at the point of assessment and are justified.	**7.1.3 Exposure Pathway Screening** (principally) Identification of the important exposure pathways for the CA, with respect to the POAs identified in Section 4 is provided in this section. The exposure pathways are consistent with the PA, with the exception that groundwater-related pathways are not addressed in the CA, due to the lack of a POA affected by groundwater use. The CA exposure pathways consider only the pathways related to exposure to individual members of the public at the POA. **9.5 Relationship to Performance Assessment Modeling** In this section it is noted that the CA and the PAs both use contaminated water as an exposure medium. In the CA surface water is the source, while the PA uses groundwater. However, many of the ways in which the water is used, such as drinking and irrigation, are the same in the two analyses.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.7 Exposure Pathways and Dose Analysis (continued)	
3.3.7.2 The dose analysis performed for the CA is consistent with that performed for the PA for similar exposure pathways and similar exposure scenarios.	**9.3.2 Dose Module** The equations described in this section for estimating exposure and dose (and implemented in the GoldSim™ Dose Module) are identical to those used for PAs. **9.5 Relationship to Performance Assessment Modeling** The dose module for the CA was developed based upon the dose modules utilized within the ELLWF PA and FTF PA, and the CA radionuclide screening (Section 7.2). The dose module developed for use in the CA has also been produced to support a variety of PA efforts involving multiple facilities at the SRS. This approach assures consistency in dose calculations between the PA and CA programs for similar pathways and scenarios.
3.3.8 Sensitivity/Uncertainty	
3.3.8.1 The sensitivity and uncertainty analysis considers factors such as alternative land use plans, CERCLA/RCRA actions or other actions (e.g., D&D, tank closures), radionuclide inventories, site and facility characteristics, and transport parameters to provide reasonable estimates of potential doses at the point of assessment for the period of the assessment. The maximum projected dose over the period of the assessment (at least 1,000 years) is presented at the point of assessment.	**9.4 Sensitivity and Uncertainty** The sensitivity scenarios for the CA were selected to consider factors such as release rates, radionuclide inventories, alternative points of assessment, groundwater divides, stream flow variation, and alternative disposal actions, as described in this section. The uncertainty in transport parameters (K_d's in particular, as well as aquifer flow rates) were considered in a probabilistic analysis. The maximum projected doses over the 1,000-yr period of assessment, a 10,025-yr period of assessment, and a 102,050-yr period of assessment are also provided in this section.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.8 Sensitivity/Uncertainty (continued)	
3.3.8.2 The calculated results and the sensitivity or uncertainty analysis results are used to evaluate meeting the dose constraint of 30 mrem/year and the dose limit of 100 mrem/year at the point of assessment over the period of assessment.	**10.1 Deterministic Base Case** (Performance Evaluation Section) (principally) The deterministic base case doses are shown to all be well below the primary dose limit of 100 mrem/yr, and the dose constraint of 30 mrem/yr in this section. **10.2 Sensitivity and Uncertainty Results** (Performance Evaluation Section) (principally) The results of the sensitivity analyses summarized in this section are compared to the 30 mrem/yr dose constraint for each scenario evaluated. The results of the probabilistic analysis are also compared to the 30 mrem/yr dose constraint.
3.3.9 ALARA & Options Analysis	
3.3.9.1 For analyses that exceed the dose constraint of 30 mrem/year but are less than the dose limit of 100 mrem/year, an options analysis is provided which identifies alternatives that could be conducted to reduce the dose to less than the dose constraint. The options analysis, using the ALARA process, considers alternatives which are technically feasible and demonstrated to be effective in reducing doses to the public at the point of assessment over the period of the assessment.	**10.3 Options Analysis** Based on CA results indicating no doses exceeding the dose constraint of 30 mrem/yr, an options analysis was not required; thus, the ALARA process was not required to be implemented in a consideration of alternatives.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.9 ALARA & Options Analysis (continued)	
3.3.9.2 For analyses that exceed the dose limit of 100 mrem/year, an options analysis, using the ALARA process, is provided which identifies alternatives that should be conducted to reduce the dose to less than the limit. The options analysis, using the ALARA process, considers alternatives which are technically feasible and demonstrated to be effective in reducing doses to the public at the point of assessment over the period of the assessment.	**10.3 Options Analysis** Based on CA results indicating no doses exceeding the dose limit of 100 mrem/yr, an options analysis was not required; thus, the ALARA process was not required to be implemented in a consideration of alternatives.
3.3.9.3 The ALARA process uses a cost-benefit analysis based on the cost of dose-reduction in accordance with DOE O 5400.5.	**10.3 Options Analysis** Based on CA results indicating no doses exceeding the dose constraint of 30 mrem/yr or the dose limit of 100 mrem/yr, the ALARA process was not required to be implemented in consideration of cost-benefit.
3.3.10 Results Integration	
3.3.10.1 The results of the analysis for the source terms and transport of radionuclides, dose analysis, available site monitoring data, supporting field investigations, sensitivity or uncertainty analysis and options analysis are reasonable representations of the existing knowledge of the site, disposal facility, and contributing sources.	**9.1 Conceptual Model and Modeling Philosophy** (principally) This section describes the base case modeling approach adopted for the CA, which was to develop a reasonably-conservative best estimate of the dose at each POA. The results of the sensitivity and uncertainty analyses aided in appropriate interpretation of the base case modeling results. This section also discusses the conceptual model's consistency with existing knowledge of the site and disposal facilities (partially accomplished through consistency with the PAs); consistency with SRS groundwater models, including those available for regions of the site not impacted by PA facilities; consistency with the available knowledge of source locations and inventory at the site; and consistency with the site's land use plan. The results arising from implementation of this conceptual model are considered to be reasonable representations of the existing knowledge. **3.1 SRS Facility Descriptions and Operational History** A comprehensive review of the SRS operational history relevant to facilities and areas of concern with respect to the CA is provided in this section. This historical information is important in identifying potential source locations in addition to PA facilities, and in developing reasonable estimates of inventories for these source locations.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.10 Results Integration (continued)	
3.3.10.1 (continued)	**3.2 Savannah River Site and Facility Characteristics** Available information about the physical, chemical, and biological environment at the SRS is provided in this section, with special emphasis on the surface and groundwater hydrology. Available monitoring data and other data arising from field investigations are summarized for all of the areas of the SRS in which residual sources of radionuclides will potentially exist at site closure. This information was also important in describing sources and inventories, and providing a perspective of the extensive body of scientific knowledge that has been developed for the SRS. **8.2 Radionuclide Source Locations and Inventory** This section identifies potentially significant sources of radionuclides, and the estimated inventories that provide a source term for the model. The information provided here is a reasonable, and most current, representation of the existing knowledge of the source terms at the site. **9.3.8 Composite Analysis Base Case Deterministic Results** The results provided in this section represent a reasonably-conservative best estimate of the maximum dose, in mrem/yr, for each source, and the total dose as a function of time for each exposure scenario and POA. **9.4.1 Sensitivity Scenario Selection** The selection of scenarios for which sensitivity analyses were carried out, described in this section, was based on consideration of how to provide additional assurance of the reasonableness of the deterministic base case model. **9.4.2 Sensitivity Analysis** The results of the sensitivity analyses are discussed here to provide additional interpretive information for the base case results. These results provided considerable confidence that the base case model results are reasonable. **10 Performance Evaluation and Results Integration** The discussion in this section focuses on interpretation of the base case results in light of the sensitivity and uncertainty analyses, concluding that these results provide great confidence that the dose limit and dose constraint will not be exceeded under any reasonable conditions, at any of the POAs.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.10 Results Integration (continued)	
3.3.10.2 The analysis, results, and conclusions of the CA provide a reasonable representation of the disposal facility and other contributing sources for determining the appropriate actions to be taken for the protection of public health and environment. The analysis and results of the CA are consistent with comparable results of the PA and provide a defensible and complete basis for an acceptable decision by DOE.	**9.5 Relationship to Performance Assessment Modeling** (principally) This section points out the many consistencies between the CA and PAs, including, but not limited to, the use of the same radionuclide fluxes to the water table for some PA facilities, the use of the same aquifer flow model for the GSA, the use of the same exposure and dose models, and the use of the same input data for most parameters. By using the PA fluxes to the water table from the PA for facilities with releases potentially contributing more that 0.1 mrem/yr to the total dose, and using the same source of aquifer flow parameters, the results are consistent. Final doses cannot be compared between the CA and PA due to the different POAs of the analyses. **10 Performance Evaluation and Results Integration** (principally) The discussion in this section focuses on interpretation of the base case results in light of the sensitivity and uncertainty analyses, concluding that these results provide great confidence that the dose limit and dose constraint will not be exceeded under any reasonable conditions, at any of the POAs. **9.3.9 Summary of Base Case Deterministic Modeling (**principally) The summarized base case modeling results presented in this section provide insight into which sources, POAs, and pathways are most important to the doses calculated, and thus most important to consider in protection of public health. **9.1 Conceptual Model and Modeling Philosophy** This section describes the base case modeling approach adopted for the CA, which was to develop a reasonably-conservative best estimate of the dose at each POA. The results of the sensitivity and uncertainty analyses aided in appropriate interpretation of the base case modeling results. This section also discusses the conceptual model's consistency with existing knowledge of the site and disposal facilities (partially accomplished through consistency with the PAs); consistency with SRS groundwater models, including those available for regions of the site not impacted by PA facilities; consistency with the available knowledge of source locations and inventory at the site; and consistency with the site's land use plan. The results arising from implementation of this conceptual model are considered to be reasonable representations of the existing knowledge. **9.3.1 Disposal Unit Module** The analytical models used in describing source release utilize site data largely consistent with data utilized in the PAs. These models also relied heavily on process descriptions and input data derived from the large body of site knowledge available, thus providing a defensible basis for calculating releases of all contributing sources.

Table B-1. CA Review Criteria Matrix (from Phifer et al. 2008) - continued	
Criteria	**Where to Find (CA Report Sections)**
3.3.10 Results Integration (continued)	
3.3.10.2 (continued)	**9.4.2 Sensitivity Analysis** The sensitivity analyses provide confidence that the base case results are reasonable, and not likely to underestimate the doses at any of the POAs.
3.3.10.3 The conclusions of the CA address and incorporate any constraints resulting from other DOE programs or from any Federal, state, and local statutes or regulations or agreements that would influence the calculated results or the options analysis.	N/A
3.3.10.4 Implementation of the conclusions from the options analysis can be reasonably accomplished at the disposal facility or the other contributing sources.	**10.3** – N/A (Based on CA results indicating no doses exceeding the dose constraint of 30 mrem/yr or the dose limit of 100 mrem/yr, an options analysis was not required)
3.3.11 QA	
3.3.11.1 The CA discusses QA measures applied to the preparation of the analysis and its documentation. The CA includes appendices or references to published documents that provide a basis for the discussions in the CA.	**6 Quality Assurance and Quality Control** The quality assurance controls applied in production of the CA are described in detail in this section, and the published document from which this information was derived is cited. In addition to the QA measures from this document, addition QA measures applicable to the CA-associated C14 geochemical sorption studies and the dose module development are described. **12 References** (and reference lists in individual sections) This section provides a comprehensive list of all the references cited in the CA that were used to provide a basis for the discussions. Each section also lists references cited in that particular section; references which are therefore included in Section 12. **Appendix A Inventory** Appendix A lists the inventory values that were used in CA calculations, and introduced in Section 8. **Appendix B Composite Analysis Review Criteria Matrix** Table B-1 provides information on where in the CA document review criteria are addressed **Appendix C Input Data** Appendix C provides tables of input data for both the deterministic base case model, and the sensitivity and probabilistic analyses **Appendix D Aquifer Flow Path Data and Figures** Appendix D provides data and figures representing the aquifer flow path information used in the base case modeling **Appendix E Modeling Results Figures** Appendix E provides figures representing a subset of the base case modeling results (a complete set of figures is provided in the cited reference), and figures representing the deterministic modeling results for years beyond the 1,000-yr AP. **Appendix F Key Assumptions** A listing of key assumptions used in carrying out the analyses in support of the CA results is provided in Appendix F.

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C. APPENDIX C: COMPOSITE ANALYSIS INPUT DATA TABLES

C.1 INPUT DATA TABLES

Table C-1. Source Naming, Related SRS Areas, Related POAs, and Source Information used in Composite Analysis Modeling

Table C-1. Source Naming, Related SRS Areas, Related POAs, and Source Information used in Composite Analysis Modeling

Source #	SRS Area	CA Base Case Abbreviated Name [1]	Source Identification [1]	Aquifer Flow Path Identification [2]	Related POA [3]	Appendix A Inventory Table	Inventory Date	End State[4]
1	SRNL	A-Area Seep	Savannah River Lab Seepage Basins	A_SeepBasins	UTR	A-3	2002	SB
2		773-A	773-A Main Laboratory	A_773	SAV	A-4	2025	DTCS
3		776-A	776-1A – 776-6A Radioactive Liquid Waste Handling Facilities	A_776_Complex	UTR	A-5	2025	DTCS
4		778-A	778-1A and 778-2A Solid Waste Handling and Staging Facility	A_778_Complex	SAV	A-6	2025	DTCS
5	C-Area	105-C SS	C-Area Reactor Building Stainless Steel	C_Reactor_Vessel	FMB	A-49	2008	ISD
6		105-C Al	C-Area Reactor Building Aluminum	C_Reactor_Vessel	FMB	A-50	2008	ISD
7		105-C Con	C-Area Reactor Building Concrete	C_Reactor_Bld	FMB	A-51	2008	ISD
8		105-C Surf	C-Area Reactor Building Surface Contamination	C_Reactor_Bld	FMB	A-52	2008	ISD
9		C-Area Seep	C-Area Seepage Basin	C_SeepBasins	FMB	A-61	1997	BCG / SB
10		C-Area Dirt	C-Area Reactor Cask Car Railroad Tracks	C_Reactor_Bld	FMB	A-62	2007	Removal to 105-C
11	D-Area	420-D	420-D Concentrator Building	D_420	SAV	A-8	2006	DTCS
12		420-2D	420-2D Rework Handling Facility	D_420-2	SAV	A-9	2005	DTCS
13		421-D	421-D Finishing Building	D_421	SAV	A-10	2005	DTCS
14		421-2D	421-2D Moderator Handling and Storage Facility	D_421-2	SAV	A-11	2004	DTCS
15		772-D	772-D Control Laboratory	D_772	SAV	A-12	2004	DTCS
16	E-Area	E-Area CIG	Component-In-Grout Disposal Units	E_CIG	UTR	A-13	2025	GCC
17		E-Area ET	Engineered Trenches	E_ET	UTR	A-14	2025	GCC
18		E-Area ILV	Intermediate Level Vault	E_ILV	UTR	A-15	2025	GCC
19		E-Area LAWV	Low Activity Waste Vaults	E_LAWV	UTR	A-16	2025	GCC
20		NR Pad 1 Part 1	Naval Reactor Components Disposal Areas 643-7E Metal Part 1	E_NR_Pad1	UTR	A-17	2025	GCC

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Source #	SRS Area	CA Base Case Abbreviated Name [1]	Source Identification [1]	Aquifer Flow Path Identification [2]	Related POA [3]	Appendix A Inventory Table	Inventory Date	End State[4]
21	E-Area	NR Pad 1 Part 2	Naval Reactor Components Disposal Areas 643-7E Metal Part 2	E_NR_Pad1	UTR	A-17	2025	GCC
22		NR Pad 1 Part 3	Naval Reactor Components Disposal Areas 643-7E Metal Part 3	E_NR_Pad1	UTR	A-17	2025	GCC
23		NR Pad 1 Part 4	Naval Reactor Components Disposal Areas 643-7E Metal Part 4	E_NR_Pad1	UTR	A-17	2025	GCC
24		NR Pad 1 Part 5	Naval Reactor Components Disposal Areas 643-7E Metal Part 5	E_NR_Pad1	UTR	A-17	2025	GCC
25		NR Pad 1 Part 6	Naval Reactor Components Disposal Areas 643-7E Metal Part 6	E_NR_Pad1	UTR	A-17	2025	GCC
26		NR Pad 1 Crud	Naval Reactor Components Disposal Areas 643-7E Liquid Crud	E_NR_Pad1	UTR	A-17	2025	GCC
27		NR Pad 2	Naval Reactor Components Disposal Areas 643-26E	E_NR_Pad2	UTR	A-18	2025	GCC
28		E-Area ST East	Slit Trenches East	E_ST_East	UTR	A-19	2025	GCC
29		E-Area ST Cent	Slit Trenches Central	E_ST_Center	UTR	A-20	2025	GCC
30		E-Area ST West	Slit Trenches West	E_ST_West	UTR	A-21	2025	GCC
31		TRU Pad	TRU Pad 1	E_TRU_Pad	UTR	A-22	1972	OSR or ISD
32		LLRWDF FMB	Low-Level Radioactive Waste Disposal Facility Path to Fourmile Branch	E_MWMF_FMB	FMB	A-23	1990	GCC
33		LLRWDF UTR	Low-Level Radioactive Waste Disposal Facility Path to Upper Three Runs	E_MWMF_UTR	UTR	A-23	1990	GCC
34		MWMF	Mixed Waste Management Facility	MWMF	UTR	A-23	1979	CCC
35		ORWBG	Old Radioactive Waste Burial Grounds w/o I129 on Berl Saddles	E_OBG	FMB	A-24	1962	GCC
36		ORWBG AgI	Old Radioactive Waste Burial Grounds I129 on Berl Saddles	E_OBG	FMB	A-24	1962	GCC
37	F-Area	F-Area MS	F-Area Materials Storage Facility	F_FAMS	UTR	A-25	2003	ISD
38		F-Area IPSL UTR	F-Area Inactive Process Sewer Lines Path to Upper Three Runs	F_SewerLine_UTR	UTR	A-26	2002	GCC

Table C-1. Source Naming, Related SRS Areas, Related POAs, and Source Information used in Composite Analysis Modeling

Source #	SRS Area	CA Base Case Abbreviated Name [1]	Source Identification [1]	Aquifer Flow Path Identification [2]	Related POA [3]	Appendix A Inventory Table	Inventory Date	End State[4]
39	F-Area	F-Area IPSL FMB	F-Area Inactive Process Sewer Lines Path to Fourmile Branch	F_SewerLine_FMB	FMB	A-26	2002	GCC
40		221-F	F-Canyon and FB Line	F_Canyon	UTR	A-27, A-28, A-29	2007	ISD
41		211-F	Canyon Auxiliary Facilities 211-F	F_AuxFacilities	UTR	A-30	2006	DTCS
42		281-3F	F-Area Retention Basin 281-3F	F_RetentBasin	FMB[5]	A-31	2002	BCG / SB
43		F-Area Seep	F-Area Seepage Basins	F_SeepBasin	FMB	A-32	2002	CCC
44		FTF Type I	F-Area Tank Farm Type I Tanks	F_Tanks_Type1	UTR	A-33	2020	ISD / GCC
45		FTF Type IV	F-Area Tank Farm Type IV Tanks	F_Tanks_Type4	UTR	A-33	2020	ISD / GCC
46		FTF Type III	F-Area Tank Farm Type III Tanks	F_Tanks_Type3	UTR	A-33	2020	ISD / GCC
47		FTF Type IIIA UTR	F-Area Tank Farm Type IIIA Tanks Path to Upper Three Runs	F_Tanks_Type3A_UTR	UTR	A-33	2020	ISD / GCC
48		FTF Type IIIA FMB	F-Area Tank Farm Type IIIA Tanks Path to Fourmile Branch	F_Tanks_Type3A_FMB	FMB	A-33	2020	ISD / GCC
49		FTF Equip	FTF Ancillary Equipment	F_Equipment	UTR	A-34	2020	GCC
50		772-F	Central Laboratory Facility Building 772-F	F_Central_Labs1	UTR	A-35	2025	ISD
51		772-1F,-4F	Central Laboratory Facility Buildings 772-1F and 772-4F	F_Central_Labs2	UTR	A-35	2025	DTCS
52		F-Area Old Seep	Old F-Area Seepage Basin	F_Old_SeepBasin	UTR	A-36	2002	DTCS
53	H-Area	HTF Type I UTR E	H-Area Tank Farm Type I Tanks East Path to Upper Three Runs	H_Tanks_Type1_East	UTR	A-37	2032	ISD / GCC
54		HTF Type I UTR W	H-Area Tank Farm Type I Tanks West Path to Upper Three Runs	H_Tanks_Type1_West	UTR	A-37	2032	ISD / GCC
55		HTF Type II	H-Area Tank Farm Type II Tanks	H_Tanks_Type2	UTR	A-37	2032	ISD / GCC
56		HTF Type IV UTR	H-Area Tank Farm Type IV Tanks Path to Upper Three Runs	H_Tanks_Type4_UTR	UTR	A-37	2032	ISD / GCC
57		HTF Type IV FMB	H-Area Tank Farm Type IV Tanks Path to Fourmile Branch	H_Tanks_Type4_FMB	FMB	A-37	2032	ISD / GCC
58		HTF Type III UTR E	H-Area Tank Farm Type III Tanks East Path to Upper Three Runs	H_Tanks_Type3_UTR_East	UTR	A-37	2032	ISD / GCC

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Source #	SRS Area	CA Base Case Abbreviated Name [1]	Source Identification [1]	Aquifer Flow Path Identification [2]	Related POA [3]	Appendix A Inventory Table	Inventory Date	End State[4]
59	H-Area	HTF Type III UTR W	H-Area Tank Farm Type III Tanks West Path to Upper Three Runs	H_Tanks_Type3_UTR_West	UTR	A-37	2032	ISD / GCC
60		HTF Type III FMB	H-Area Tank Farm Type III Tanks Path to Fourmile Branch	H_Tanks_Type3_FMB	FMB	A-37	2032	ISD / GCC
61		HTF Type IIIA UTR E	H-Area Tank Farm Type IIIA Tanks East Path to Upper Three Runs	H_Tanks_Type3AE	UTR	A-37	2032	ISD / GCC
62		HTF Type IIIA UTR W	H-Area Tank Farm Type IIIA Tanks West Path to Upper Three Runs	H_Tanks_Type3AW_UTR	UTR	A-37	2032	ISD / GCC
63		HTF Type IIIA FMB	H-Area Tank Farm Type IIIA Tanks Path to Fourmile Branch	H_Tanks_Type3AW_FMB	FMB	A-37	2032	ISD / GCC
64		HTF Equip UTR	H-Area Tank Farm Ancillary Equipment Path to Upper Three Runs	H_Equipment_UTR	UTR	A-38	2032	GCC
65		HTF Equip FMB	H-Area Tank Farm Ancillary Equipment Path to Fourmile Branch	H_Equipment_FMB	FMB	A-38	2032	GCC
66		H-Area IPSL UTR	H-Area Inactive Process Sewer Lines Path to Upper Three Runs	H_SewerLine_UTR	UTR	A-39	2002	GCC
67		H-Area IPSL FMB	H-Area Inactive Process Sewer Lines Path to Fourmile Branch	H_SewerLine_FMB	FMB	A-39	2002	GCC
68		281-1H, 2H	H-Area Retention Basins 281-1H and 281-2H	H_RetBasin_281_1_2H	UTR	A-40	2002	ISD
69		281-8H	H-Area Retention Basin 281-8H	H_RetBasin_281_8H	FMB[5]	A-40	2002	SB
70		H-Area Seep	H-Area Seepage Basins	H_SeepBasins_904_44G	FMB	A-41	2002	CCC
71		H-Canyon	H-Canyon and HB-Line	H_Canyon	UTR	A-42, A-43	2025	ISD
72		RBOF	Receiving Basin for Offsite Fuels	H_RBOF	UTR	A-44	2007	ISD (244-H) / DTCS (245-H)
73		H-Area Sand	H-Area Sand Filter	H_Sand_Filter	UTR	A-45	2025	ISD
74		232-H	Building 232-H	H_Tritium_232_H	UTR	A-46	2007	ISD
75		HANM	H-Area New Manufacturing	Identification in Hamm et al. 2009	UTR	A-46	2025	ISD

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Source #	SRS Area	CA Base Case Abbreviated Name [1]	Source Identification [1]	Aquifer Flow Path Identification [2]	Related POA [3]	Appendix A Inventory Table	Inventory Date	End State[4]
76	H-Area	HAOM	H-Area Old Manufacturing	Identification in Hamm et al. 2009	UTR	A-46	2025	DTCS
77		234-7H	Materials Test Facility	H_Tritium_234_7H	UTR	A-46	2025	DTCS
78		236-H	Byproduct Purification Facility	H_Tritium_236_H	UTR	A-46	2025	ISD
79		237-H	Reclamation and Empty Reservoir Storage Facility	H_Tritium_237_238 H	UTR	A-46	2025	DTCS
80		TEF	Tritium Extraction Facility	Identification in Hamm et al. 2009	UTR	A-47	2025	ISD
81		264-2H	Remote Handling Building	H_Tritium_TEF_26 4_2H	UTR	A-47	2025	ISD
82		TPBAR	Tritium Producing Burnable Absorber Rods	H_Tritium_TPBAR	UTR	A-47	2025	ISD or removal
83		J-Area SWPF	Salt Waste Processing Facility	J_SWPF	UTR	A-79	2031	DTCS
84	K-Area	KAC SS	K-Area Reactor Building Stainless Steel	K_Reactor_Vessel	STC	A-49	2008	ISD
85		KAC Al	K-Area Reactor Building Aluminum	K_Reactor_Vessel	STC	A-50	2008	ISD
86		KAC Con	K-Area Reactor Building Concrete	K_Reactor_Bld	STC	A-51	2008	ISD
87		KAC Surf	K-Area Reactor Building Surface Contamination	K_Reactor_Bld	STC	A-52	2008	ISD
88		K-Area Seep	K-Area Seepage Basin	K_SeepBasin	STC	A-59	2001	BCG / SB
89		K-Area Dirt	K-Area Reactor Cask Car Railroad Tracks	K_Reactor_Bld	STC	A-62	2007	Removal to KAC
90	L-Area	LAC SS	L-Area Reactor Building Stainless Steel	L_Reactor_Vessel	STC	A-49	2008	ISD
91		LAC Al	L-Area Reactor Building Aluminum	L_Reactor_Vessel	STC	A-50	2008	ISD
92		LAC Con	L-Area Reactor Building Concrete	L_Reactor_Bld	STC	A-51	2008	ISD
93		LAC Surf	L-Area Reactor Building Surface Contamination	L_Reactor_Bld	STC	A-52	2008	ISD
94		L-Area Seep	L-Area Seepage Basin	L_SeepBasin	STC	A-60	2001	SB
95		L-Area Dirt	L-Area Reactor Cask Car Railroad Tracks	L_Reactor_Bld	STC	A-62	2007	Removal to LAC
96	M-Area	M-Area HWMF	M-Area Hazardous Waste Management Facility	M_SettlingBasin	UTR	A-1	2002	GCC

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Source #	SRS Area	CA Base Case Abbreviated Name [1]	Source Identification [1]	Aquifer Flow Path Identification [2]	Related POA [3]	Appendix A Inventory Table	Inventory Date	End State[4]
97	M-Area	M-Area MIPSL	M-Area Inactive Process Sewer Line	M_SewerLines	UTR	A-2	2002	Grouted
98		N-Area Seep	Ford Building Seepage Basin	N_Ford	UTR	A-48	2002	SB
99	N-Area	105-P SS	P-Area Reactor Building Stainless Steel	P_Reactor_Vessel	STC	A-49	2008	ISD
100	P-Area	105-P Al	P-Area Reactor Building Aluminum	P_Reactor_Vessel	STC	A-50	2008	ISD
101		105-P Con	P-Area Reactor Building Concrete	P_Reactor_Bld	STC	A-51	2008	ISD
102		105-P Surf	P-Area Reactor Building Surface Contamination	P_Reactor_Bld	STC	A-52	2008	ISD
103		P-Area Seep	P-Area Seepage Basin	P_SeepBasins	STC	A-58	2001	BCG / CCC
104		P-Area Dirt	P-Area Reactor Cask Car Railroad Tracks + Outfalls P02 and P007	P_Reactor_Bld	STC	A-62, A-64, A-65	2007	Removal to 105-P
105	R-Area	105-R SS	R-Area Reactor Building Stainless Steel	R_Reactor_Vessel	LTR	A-53	2008	ISD
106		105-R Al	R-Area Reactor Building Aluminum	R_Reactor_Vessel	LTR	A-54	2008	ISD
107		105-R Con L	R-Area Reactor Building Concrete Long Path	R_Reactor_Bld_Long	LTR	A-55	2008	ISD
108		105-R Con S	R-Area Reactor Building Concrete Short Path	R_Reactor_Bld_Short	LTR	A-56	2008	ISD
109		105-R Surf L	R-Area Reactor Building Surface Contamination Long Path	R_Reactor_Bld_Long	LTR	A-56	2008	ISD
110		105-R Surf S	R-Area Reactor Building Surface Contamination Short Path	R_Reactor_Bld_Short	LTR	A-56	2008	ISD
111		R-Area Seep	R-Area Seepage Basin	R_SeepBasin	LTR	A-57	1998	CSCC
112		R-Area Dirt L	R-Area Reactor Cask Car Railroad Tracks Long Path	R_Reactor_Bld_Long	LTR	A-63	2007	Removal to 105-R
113		R-Area Dirt S	R-Area Reactor Cask Car Railroad Tracks Short Path	R_Reactor_Bld_Short	LTR	A-63	2007	Removal to 105-R
114	S-Area	221-S	DWPF Building 221-S	S_DWPF	UTR	A-67	2031	ISD
115		511-S LPPP	DWPF Low Point Pump Pit	S_PumpPit	UTR	A-67	2031	DTCS
116		294-S Sand L	DWPF Sand Filter Long Path	S_Sandfilter_Long	UTR	A-67	2031	ISD
117		294-S Sand S	DWPF Sand Filter Short Path	S_Sandfilter_Short	UTR	A-67	2031	ISD
118	T-Area	TNX BG	TNX Burial Ground	T_BurialGround	SAV	A-68	1997	GCC

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Source #	SRS Area	CA Base Case Abbreviated Name [1]	Source Identification [1]	Aquifer Flow Path Identification [2]	Related POA [3]	Appendix A Inventory Table	Inventory Date	End State[4]
119	T-Area	T-Area IPSL	Old TNX Seepage Basin/Inactive Process Sewer Line	T_SeepBasin	SAV	A-69	1997	GCC
120		TNX Gully	TNX Discharge Gully	T_Gully	SAV	A-70	1997	GCC
121		TNX Outfall	TNX Outfall Delta and Swamp	T_Delta	SAV	A-71	1997	GCC
122		677-T	TNX Buildings 677-T and 678-T	T_Buildings	SAV	A-72	2004	GCC
123	Z-Area	Z-Area Vault 1	Saltstone Vault 1	Z_Vault1	UTR	A-73	2030	ISD / GCC
124		Z-Area Vault 4	Saltstone Vault 4	Z_Vault4	UTR	A-74	2030	ISD / GCC
125		Z-Area Vault 2 NL	Saltstone Vault 2 North Long Path	Z_Vault2_North_Long	UTR	A-75	2030	ISD / GCC
126		Z-Area Vault 2 NS	Saltstone Vault 2 North Short Path	Z_Vault2_North_Short	UTR	A-75	2030	ISD / GCC
127		Z-Area Vault 2 S	Saltstone Vault 2 Short Path	Z_Vault2_South	UTR	A-75	2030	ISD / GCC
128	F-Area	MOXFF	Mixed Oxide Fabrication Facility	F_MOX	UTR	A-80	2025	ISD
129	H-Area	T-13 Spill	Spill at Tank 13	H_Type2_Tank13_Spill	UTR	A-78	1997	GCC
130		T-9 Spill	Spill at Tank 9	H_Type1_Tank9_Spill	UTR	A-78	1997	ISD / GCC
131		T-16 Spill	Spill at Tank 16	H_Type2_Tank16_Spill	UTR	A-78	1997	ISD / GCC
132		T-37 Spill UTR	Spill at Tank 37 Path to Upper Three Runs	H_Type3AW_Tank37_Spil_UTR	UTR	A-78	1997	ISD / GCC
133		T-37 Spill FMB	Spill at Tank 37 Path to Fourmile Branch	H_Type3AW_Tank37_Spil_FMB	FMB	A-78	1997	ISD / GCC
134	F-Area	281-3F Spill	Spill at 281-3F	F_RetentBasin	FMB[5]	A-78	1997	SB
135		T-3 Spill	Spill at Tank 3	F_Type1_Tank3_Spill	UTR	A-78	1997	ISD / GCC
136		T-8 Spill	Spill at Tank 8	F_Type1_Tank8_Spill	UTR	A-78	1997	ISD / GCC
137	H-Area	281-3H Spill	Spill at 281-3H	H_Spill_281_3H	FMB[5]	A-78	1997	SB

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Source #	SRS Area	CA Base Case Abbreviated Name [1]	Source Identification [1]	Aquifer Flow Path Identification [2]	Related POA [3]	Appendix A Inventory Table	Inventory Date	End State[4]
138	General Site	FMB GOU	Fourmile Branch Groundwater Operable Unit	NA	FMB	NA	2008	No Action
139		D-Area GOU	D-Area Groundwater Operable Unit	D_420-2	SAV	A-77	2002	No Action
140		K-Area GOU	K-Area Groundwater Operable Unit	K_RetBasin	STC	A-77	2002	No Action
141		L-Area GOU	L-Area Groundwater Operable Unit	L_SeepBasin	STC	A-77	2002	No Action
142		P-Area GOU	P-Area Groundwater Operable Unit	P_SeepBasins	STC	A-77	2002	No Action
143		R-Area GOU	R-Area Groundwater Operable Unit	R_SeepBasin	LTR	A-77	2002	No Action
144		T-Area GOU	TNX Groundwater Operable Unit	T_SeepBasin	SAV	A-77	2002	No Action
145		FMB IOU	Fourmile Branch Integrator Operable Unit	NA	FMB	A-76	2002	No Action
146		LTR IOU	Lower Three Runs Integrator Operable Unit	NA	LTR	A-76	2002	No Action
147		PB IOU	Pen Branch Integrator Operable Unit	NA	STC	A-76	2002	No Action
148		SR-A IOU	Savannah River and Floodplain Swamp Integrator Operable Unit upstream from where Steel Creek flows into the floodplain	NA	SAV	A-76	2002	No Action
149		SR-B IOU	Savannah River and Floodplain Swamp Integrator Operable Unit downstream from where Steel Creek flows into the floodplain	NA	SAV	A-76	2002	No Action
150		SC IOU	Steel Creek Integrator Operable Unit	NA	STC	A-76	2002	No Action
151		UTR IOU	Upper Three Runs Integrator Operable Unit	NA	UTR	A-76	2002	No Action
152		SpdTbr IOU	Steed Pond/Tims Branch Integrator Operable Unit	NA	UTR	A-7	2002	No Action

Notes for Table C-1:
[1] From Appendix C of *Savannah River Site Composite Analysis: Base Case Deterministic Calculations* (Smith et al. 2009a).
[2] From Table 2.2-1 of *Savannah River Site Composite Analysis: Aquifer Flow Path Parameters* (Hamm et al. 2009).
[3] Points of Assessment from Table 2.2-1 of *Savannah River Site Composite Analysis: Aquifer Flow Path Parameters* (Hamm et al. 2009):

UTR = Upper Three Runs FMB = Fourmile Branch LTR = Lower Three Runs
SAV = Savannah River (below site) STC = Steel Creek (includes Pen Branch)

[4] From Table 4 of *Conceptual Model for the Savannah River Site's Composite Analysis* (Phifer et al. 2008):

BCG = Bentonite Cement Grout CCC = Clay Closure Cap CSCC = Concrete Slab Closure Cap
DTCS = Demolish to Concrete Slab GCC = Geosynthetic Closure Cap ISD = In Situ Disposal within Massive Concrete Structure
OSR = Off-Site Removal (i.e., removed from SRS) SB = Soil Backfill

[5] Actual POA parameter passed to GoldSim™ for analysis was 'UTR'. A correction will be implemented in future CA analyses. Note that their dose contributions were very small.

Table C-2. Partitioning and Groupings of Appendix A Source Inventories for use in CA Calculations (Smith et al. 2009a)

Source #	Source	Appendix A Inventory Tables	Partitioning and Groupings of inventories Listed in Appendix A as Used in CA Modeling
20	NR Pad 1 Part 1	A-17	Inventory for NRCDA 643-7E (NR Pad 1) was divided into seven parts representing six distinct metal components and a liquid fraction ("crud") based on information in Yu et al. (2001) and WSRC (2008a). I129 inventory was corrected from 1.48E+05 to 1.48E-05 (Sink 2007).
21	NR Pad 1 Part 2	A-17	
22	NR Pad 1 Part 3	A-17	
23	NR Pad 1 Part 4	A-17	
24	NR Pad 1 Part 5	A-17	
25	NR Pad 1 Part 6	A-17	
26	NR Pad 1 Crud	A-17	
32	LLRWDF FMB	A-23	23.7% of total inventory was split to the LLRWDF which was then split 2.5% to Fourmile Branch 97.5% to Upper Three Runs. H3 was removed from FMB inventory and accounted for through Source #138 FMB GOU.
33	LLRWDF UTR	A-23	
34	MWMF	A-23	
35	ORWBG	A-24	H3 was removed from inventory, 99% of I-129 inventory assumed to be AgI special waste form.
36	ORWBG AgI	A-24	
38	F-Area IPSL UTR	A-26	85.8% of total inventory split to Upper Three Runs 14.2% to Fourmile Branch. H3 was removed from FMB inventory and accounted for through Source #138 FMB GOU.
39	F-Area IPSL FMB	A-26	
40	221-F	A-27, A-28, A-29	Combined inventory from F-Canyon, FB-Line, and canyon vessels.
43	F-Area Seep	A-32	H3 inventory was removed and accounted for through Source #138 FMB GOU.
44	FTF Type I	A-33	Type I = combined inventory from Tanks 1-8. Type IV = combined inventory from Tanks 17-20. Type III = combined inventory from Tanks 33 and 34. Type IIIA UTR = combined inventory from Tanks 25, 26 and 44-46, 94.3% of Tank 27 inventory + 0.5% of Tank 47 inventory. Type IIIA FMB = remainder of Tank 27 and 47 inventory.
45	FTF Type IV	A-33	
46	FTF Type III	A-33	
47	FTF Type IIIA UTR	A-33	
48	FTF Type IIIA FMB	A-33	
54	HTF Type I UTR W	A-37	Type I UTR West = remaining Tank 11 and 12 inventory. Type II = combined inventory from Tanks 13-16. Type IV UTR = combined inventory from Tanks 21 and 23 + 18.5% of the inventory from Tank 24. Type IV FMB = Inventory from Tank 22 + remainder of the inventory from Tank 24. Type III UTR East = 96.5% of the inventory from Tank 29 + 61.1% of the inventory from Tank 30 not going to FMB. Type III UTR West = remaining Tank 29 and 30 inventory. Type III FMB = combined Tanks 31 & 32 inventory + 4.9% of the inventory from Tank 29 & 85.4% of the inventory from Tank 30.
55	HTF Type II	A-37	
56	HTF Type IV UTR	A-37	
57	HTF Type IV FMB	A-37	
58	HTF Type III UTR E	A-37	
59	HTF Type III UTR W	A-37	
60	HTF Type III FMB	A-37	

Table C-2 Partitioning and Groupings of Appendix A Source Inventories for use in CA Calculations (Smith et al. 2009a) - Continued

Source #	Source	Appendix A Inventory Tables	Partitioning and Groupings of inventories Listed in Appendix A as Used in CA Modeling
61	HTF Type IIIA UTR E	A-37	Type IIIA UTR East = combined inventory from Tanks 38-43 and 48-51. Type IIIA UTR West = remaining Tank 37 inventory. Type IIIA FMB = combined inventory from Tanks 35 and 36 + 55.1% of the inventory from Tank 37.
62	HTF Type IIIA UTR W	A-37	
63	HTF Type IIIA FMB	A-37	
64	HTF Equip UTR	A-38	Inventory split 66.8% to Upper Three Runs 33.2% to Fourmile Branch.
65	HTF Equip FMB	A-38	
66	H-Area IPSL UTR	A-39	Inventory split 87.0% to Upper Three Runs 13.0% to Fourmile Branch. H3 was removed from Fourmile Branch inventory and accounted for through Source #138 FMB GOU.
67	H-Area IPSL FMB	A-39	
68	281-1H, 2H	A-40	Combined inventory from 281-H and 281-2H.
70	H-Area Seep	A-41	H3 inventory was removed and accounted for through Source #138 FMB GOU.
71	H-Canyon	A-42, A-43	Combined inventory from H-Canyon and HB-Line.
76	HAOM	A-46	Includes Vault 217-H inventory.
83	J-Area SWPF	A-79	Used total inventory listed in table.
104	P-Area Dirt	A-62, A-64, A-65	Combined inventory from P-Area cask car railroad tracks, P02 outfall and P007 outfall.
107	105-R Con L	A-55	Inventory split 7.1% to long path and 92.9% to short path.
108	105-R Con S	A-55	
109	105-R Surf L	A-56	Inventory split 7.1% to long path and 92.9% to short path.
110	105-R Surf S	A-56	
112	R-Area Dirt L	A-63	Inventory split 7.1% to long path and 92.9% to short path.
113	R-Area Dirt S	A-63	
116	294-S Sand L	A-67	S-Area sand filter inventory split 55.3% to long path and 44.7% to short path.
117	294-S Sand S	A-67	
123	Z-Area Vault 1	A-73	From the aquifer flow analysis (Hamm et al. 2009), the Vault 2 inventory was split 42.0% to long north path, 14.2% to short north path and 43.8% to south path.
124	Z-Area Vault 4	A-74	
125	Z-Area Vault 2 NL	A-75	
126	Z-Area Vault 2 NS	A-75	
127	Z-Area Vault 2 S	A-75	
132	T-37 Spill UTR	A-78	Inventory split 44.4% to Upper Three Runs 55.6% to Fourmile Branch.
133	T-37 Spill FMB	A-78	
138	FMB GOU	NA	C-Area, F-Area (Western GSA) and H-Area GOU inventories in Table A-77 were not used. Rather the measured flux of H3 to Fourmile Branch from these combined plumes was used as described in Section 9.3.1.6.7.2.
139	D-Area GOU	A-77	
140	K-Area GOU	A-77	
141	L-Area GOU	NA	
142	P-Area GOU	A-77	
143	R-Area GOU	A-77	
144	T-Area GOU	A-77	
148	SR-A IOU	A-76	SR-A is the portion of the SR floodplain and swamp upstream of where Steel Creek flows into the floodplain and SR-B is the segment that lies downstream from that point.
149	SR-B IOU	A-76	

Table C-3. Waste Zone Thickness for Sources Requiring Source Release Models

Source(s)	Waste Thickness (ft)	Assumptions and Reference for Thickness
NR Pad 1 (Parts 1 through 6)	15	TRU Pad waste zone height from *E-Area Low-Level Waste Facility DOE 435.1 Performance Assessment* (WSRC 2008)
LLRWDF FMB, LLRWDF UTR, MWMF	20	MWMF trench depth from *Savannah River Site Vadose Zone Data for Composite Analysis Calculations* (Noffsinger et al. 2009)
ORWBG, ORWBG AgI	20	ORWBG trench depth from *Savannah River Site Vadose Zone Data for Composite Analysis Calculations* (Noffsinger et al. 2009)
232-H	0.75	building pad thickness from *Savannah River Site Vadose Zone Data for Composite Analysis Calculations* (Noffsinger et al. 2009)
HANM	2.5	building pad thickness from *Savannah River Site Vadose Zone Data for Composite Analysis Calculations* (Noffsinger et al. 2009)
HAOM	0.33	building pad thickness from *Savannah River Site Vadose Zone Data for Composite Analysis Calculations* (Noffsinger et al. 2009)
TEF	6	building pad thickness from *Savannah River Site Vadose Zone Data for Composite Analysis Calculations* (Noffsinger et al. 2009)
264-2H	6	building pad thickness from *Savannah River Site Vadose Zone Data for Composite Analysis Calculations* (Noffsinger et al. 2009)
105-C SS, KAC SS, LAC SS, 105-P SS, 105-R SS	23	Reactor vessel height from *Constituent Transport Analysis for the R-Area Reactor Building (U)* (Council 2009)
FSB, HSB, H-Area Canyon	1	assumed

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
A/M Area	M-Area Hazardous Waste Mgmt. Facility (HWMF)	---	Closed and capped earthen basin, elev. of M-Area Settling Basin bottom = 337 ft msl	231	104	S – 20 C – 40 S – 44	• Elev. ref. - RCRA Part B Permit, WSRC-IM-91-53, Original Construction Blueprint for M-Area Settling Basin, Map 3302, Sheet 1122, SRP Map Sheet Index 5M5-G-121, from WSRC-IM-91-53, 3/1992, Fig. 4-1 • VZ lith. ref. – RFI/RI, Work Plan, BRA and CMS/FS for MAOU, App. E, WSRC-RP-2006-4060, Rev.1, 7/2007 • Bldg. 321-M lith. data used
	M-Area Inactive Process Sewer Line (MIPSL)	---	MIPSL = 7-12 ft bgs, elev. = 345.25 ft msl at discharge pipe entrance into M-Area Settling Basin (8.75 ft bgs)	235	110	S – 20 C – 40 S – 50	“
	Savannah River Laboratory (SRL) Seepage Basins	904-53G (Basin #1) 904-53G2 (Basin #2) 904-54G Basin #3) 904-55G (Basin #4)	Closed and capped earthen basins, LLW entered Basin 1 then flowed to Basins 2-4, surface elev. = 350 ft msl, Elev. of Basin #1 bottom = 335 ft msl Elev. of Basin #2 bottom = 331.6 ft msl Elev. of Basin #3 bottom = 333 ft msl Elev. of Basin #4 bottom = 329.68 ft msl	238	Basin #1 – 97 Basin #2 – 93 Basin #3 – 95 Basin #4 – 91	C – 3 S – 62 C – 8 S – 18 *for Basin #4 (deepest)	• Elev. refs. – C-CV-A-0042, C-CV-A-0043, ST5-27360 • VZ lith. Ref. - Technical Data Summary, Chemical Characterization of Sediments and Groundwater at SRL Seepage Basins, DPSTD-84-110, Fig. E-1, Driller’s Log for MW ASB-7; • Assuming all SRL Seepage Basin inventory is in Basins #1 and #2

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
A/M Area	Savannah River National Laboratory (SRNL) Bldgs.	SRNL Bldg. 773-A	Elev. of lowest floor slab = 367.67 ft msl, slab thickness = 0.33 to 0.5 ft, Elev. of slab base = 367.17 ft msl	232	135	C – 29 S – 52 C – 8 S – 46	• Elev. drawing refs.- W156563, W156409, W156410, W156412, W157633, W236107, W157390, W166273, ST5-10926 • Basis for VZ lith. is SRL Seepage Basins (see above ref.)
		776-A Complex Waste Storage and Shipping Bldg. (776-1A to 776-6A)	Elev. of lowest floor slab (776-6A) = 350 ft msl, slab thickness = 0.67 ft, Elev. of slab base = 349.33 ft msl	232	117	C – 29 S – 52 C – 8 S – 28	• Elev. drawing refs. – W156978, ST5-27133, ST5-13143, ST5-18492, C-CC-A-0005, W159556, W156979 • Basis for VZ lith. is SRL Seepage Basins (see above ref.)
		778-A Waste Pad/TRU Waste Storage Bldg. Complex (778-1A and 778-2A)	Bldgs. complex consists of structures on surface slabs and slab thickness is unknown, assuming that the ground elev. = elev. of lowest slab base, Ground elevation = 369 ft msl	232	137	C – 29 S – 52 C – 8 S – 48	• Elev. drawing ref. - No drawings found • Basis for VZ lith. is SRL Seepage Basins (see above ref.)
	Tims Branch/ Steed Pond	---	NA	NA	NA	NA	• Assuming contaminants in Tims Branch sediments
D-Area	Bldg. 420-D	---	Bldg. demolished and removed, slab remaining, slab thickness = 0.5 ft, Elev. of ground at slab = 138 ft msl	129	9	S – 9	• Elev. and VZ lith. ref. - RFI/RI WP and RFI/RI Report with BRA for the DAOU, Appendix E, WSRC-RP-2007-4079, Table E.3-1 • Elev. incorrect in drawing W140391

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
D-Area	Bldg. 0420-2D	---	Bldg. demolished and removed, slab remaining, slab thickness = 0.75 ft, Elev. of ground at slab = 137 ft msl	128	9	C – 3 S – 6	• Elev. and VZ lith. ref. - RFI/RI WP and RFI/RI Report with BRA for the DAOU, Appendix E, WSRC-RP-2007-4079, Table E.3-1 • Elev. incorrect in drawing S5-4-1101, S5-4-628
	Bldg. 421-D	---	Bldg. demolished and removed, slab remaining, slab thickness = 0.5 ft, Elev. of ground at slab = 139 ft msl	130	9	C – 3 S – 6	• Elev. and VZ lith. ref. - RFI/RI WP and RFI/RI Report with BRA for the DAOU, Appendix E, WSRC-RP-2007-4079, Table E.3-1 • Elev. incorrect in drawing W140115, W141220
	Bldg. 421-2D	---	Bldg. demolished and removed, slab remaining, slab thickness = 0.75 ft, Elev. of ground at slab = 140 ft msl	131	9	C – 3 S – 6	• Elev. and VZ lith. ref. - RFI/RI WP and RFI/RI Report with BRA for the DAOU, Appendix E, WSRC-RP-2007-4079, Table E.3-1 • Elev. incorrect in drawing W161724
	Bldg. 772-D	---	Bldg. demolished and removed, slab remaining, slab thickness = 0.5 ft, Elev. of ground at slab = 140 ft msl	131	9	C – 3 S – 6	• Elev. and VZ lith. ref. - RFI/RI WP and RFI/RI Report with BRA for the DAOU, Appendix E, WSRC-RP-2007-4079, Table E.3-1, drawing refs. - W15510, W15509
E-Area	E-Area Disposal Facilities	CIG	---	---	35	S – 35	• VZ lith ref. - WSRC-STI-2007-00306, Rev. 0, p.2-46 • Lith. lower VZ

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
E-Area	E-Area Disposal Facilities	Engineered Trenches	---	---	35	S – 35	• VZ lith ref. - WSRC-STI-2007-00306, Rev. 0, p.1-54 • Lith. lower VZ
		ILV	---	---	29	S – 29	• VZ lith ref. - WSRC-STI-2007-00306, Rev. 0, p.4-47 • Lith. lower VZ
		LAWV	---	---	40	C – 11 S – 29	• VZ lith ref. - WSRC-STI-2007-00306, Rev. 0, p.3-22 • Lith. = upper VZ (0-11.5 ft), remainder lower VZ
		Naval Reactor Pad 643-7E	---	---	54	C – 20 S – 34	• VZ thickness ref. - WSRC-STI-2007-00306, Rev. 0, p.5-31 • VZ lith ref. - WSRC-STI-2006-00198, Rev. 0, Sept. 2006
		Naval Reactor Pad 643-26E	---	---	60	S – 60	"
		Slit Trenches (East)	---	---	35	S – 35	• VZ lith ref. – WSRC-STI-2007-00306, Rev. 0, p.1-54 • Lith. lower VZ
		Slit Trenches (Central)	---	---	35	S – 35	"
		Slit Trenches (West)	---	---	35	S – 35	"
		TRU Pad #1 (660-1E)	Surface Elev. at nearby MW BGO 24D = 291 ft msl	240	51	C – 20 S – 31	• Above-grade concrete pad where containerized wastes are stored. Pad covered with soil and plastic. Ref. WSRC-IM-98-30, Vol. VII, p.B-7 • VZ lith ref. – WSRC-STI-2006-00198, Rev. 0, Sept. 2006 • Located adjacent to Naval Reactor Pad 643-7E

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
E-Area	Mixed Waste Mgmt. Facility (MWMF)	---	Avg. Surface Elev. = 285 ft msl, assuming avg. trench depth = 20 ft, Elev. of trench bottoms = 265 ft msl	234	31	S – 31	• VZ lith ref. - WSRC-STI-2006-00198, Rev. 0, Sept. 2006
	Old Radioactive Waste Burial Grounds (ORWBG)	---	Avg. Surface Elev. = 285 ft msl, assuming avg. trench depth = 20 ft, Elev. of trench bottoms = 265 ft msl	232	33	S - 33	“
F-Area	F-Area Materials Storage (FAMS)	235-F	Elev. of lowest floor slab = 302.7 ft msl	221	82	S – 82 (default lith.)	• Elev. drawing refs. - W146263, W146549, W146655, W148123
	F-Area Inactive Process Sewer Line (FIPSL)	---	Surface Elev. at nearby MW FSL 9D = 283.5 ft msl, assuming pipe enters Basin #1 5 ft bgs (1/3 of basin depth), Elev. of pipe = 278.5 ft msl, Elev. of Basin #1 bottom = 268 ft msl	217	51	S – 51 (default lith.)	• Elev. drawing ref. for Basin #1 - W840982 • FSL 9D - SGC-2003-00001, p.74
	F-Canyon and FB Line	F-Canyon Bldg. 221-F	Elev. of top of lowest floor slab = 295.5 ft msl, slab thickness = 5 ft, Elev. of slab base = 290.5 ft msl	221	69	S – 69 (default lith.)	• Elev. drawing refs. - W146799, W149552
		FB Line Bldg. 221-F	“	221	69	S – 69 (default lith.)	• Part of F-Canyon Bldg.
		F-Canyon Process Vessels	“	221	69	S – 69 (default lith.)	“

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
F-Area	Canyon Auxiliary Facilities, Bldg. 211-F	Waste Handling Vault Sludge Tanks Segregated Solvent Facility Acid Recovery Unit Water Handling Facility General Purpose Tanks General Purpose Evaporators	"	221	69	S – 69 (default lith.)	• Assuming Canyon Auxiliary Facilities are treated the same as F-Canyon Bldg.
	F-Retention Basin	281-3F	Surface elev. = 275 ft msl, Basin bottom elev. = 266 ft msl	223	43	S – 12 C – 5 S – 26	• Elev. and VZ lith. ref. - RI/BRA Report, WSRC-RP-96-356, Fig. 3.5-2. Also, GW Sampling Report with Residential RA for the F-Area Retention Basin, WSRC-RP-00905, p. C-40
	F-Seepage Basins	904-41G (Basin #1)	Closed earthen basin, Basin bottom elevation = 268 ft msl	216	52	S – 25 C – 5 S – 22	• Elev. drawing ref. - W840982 • VZ lith. ref. - RI/BRA Report, WSRC-RP-96-356, Fig. 3.5-2. Also, GW Sampling Report with Residential RA for the F-Area Retention Basin, WSRC-RP-96-00905, p. C-40 • Assuming all F-Area Seepage Basin inventory is in Basin #1
		904-42G (Basin #2)	Closed earthen basin, Basin bottom elevation = 268 ft msl	215	53	S – 25 C – 5 S – 23	"

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
F-Area	F-Seepage Basin	904-43G (Basin #3)	Closed earthen basin, Basin bottom elevation = 267.5 ft msl	210	57	S – 25 C – 5 S – 27	"
	F-Area Tank Farm	Tanks 1-8	---	225	9	S – 9	• VZ thickness ref. – FTF PA, SRS-REG-2007-00002, Rev. 0, p.311, VZ excavated below clay and tanks are entrenched in sand
		Tanks 17-20	---	223	1	S – 1	"
		Tanks 33-34	---	225	17	S – 17	"
		Tanks 25-28 and 44-47	---	Tanks 25-28, 223 Tanks 44-47, 222	18	S – 18	"
		FTF Ancillary Equipment	---	---	11	S – 11	• Assuming that ancillary equipment (pump tanks FPT-1, FPT-2 and FPT-3, 242-F Catch Tank, CTS Tank and evaporator pots 242-F and 242-16F) at same level as nearby tank bottoms (average VZ thickness = 11.3 ft and lith. = sand)
	Central Labs	Bldg. 772-F	Elev. of top of lowest floor slab = 288.83 ft msl, slab thickness = 0.33 ft, Elev. of slab base = 288.50 ft msl	225	63	S – 63 (default lith.)	• Elev. drawing refs. for 772-F Fan Room - W156260, W155606
		Bldg. 772-1F	Elev. of top of lowest floor slab = 310.67 ft msl, slab thickness = 0.42 ft, Elev. of slab base = 310.25 ft msl	225	85	S – 85 (default lith.)	• Elev. drawing ref. - W728406

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
F-Area	Central Labs	Bldg. 772-4F	Elev. of top of lowest floor slab = 310.67 ft msl, slab thickness = 0.67 ft, Elev. of slab base = 310.00 ft msl	225	85	S – 85 (default lith.)	• Elev. drawing ref. - W845819
	Old F-Area Seepage Basin	904-49G	Surface elev. = 285 ft msl, Basin bottom elev. = 275 ft msl	212	63	S – 63	• Elev. and VZ lith. ref. - Phase II RFI/RI Plan, WSRC-RP-90-731, p. 3-20, FASB RFI/RI Report WSRC-RP-94-942
H-Area	H-Area Tank Farm	Tanks 9-12	---	East – 274 Middle – 276 West – 271	0	NA	• VZ thickness ref. – SED-GTE-2008-001 (Bagwell and Millings) • Tanks excavated below clay and entrenched in sand
		Tanks 13-16	---	"	1	S – 1	"
		Tanks 21-24	---	"	11	S – 11	"
		Tanks 29-32	---	"	10	S – 10	"
		Tanks 35-37	---	"	12	S – 12	"
		Tanks 38-43	---	"	21	S – 21	"
		Tanks 48-51	---	"	15	S – 15	"
		HTF Ancillary Equipment	---	274	12	S – 12 (default lith.)	• Assuming that ancillary equipment at same level as nearby tank bottoms (average VZ thickness = 12.0 ft)
	H-Area Inactive Process Sewer Line (HIPSL)	---	Surface Elev. at nearby MW HSL 1D = 261.8 ft msl, pipe enters Basin #1, assuming pipe enters at 6.5 ft bgs, Elev. of pipe = 255.3 ft msl, Elev. of Basin #1 bottom = 248 ft msl	223	32	S – 5 C – 5 S – 3 C – 3 S – 9	• Same as H-Area Seepage Basin #1, Elev. drawing refs. - W810310, W810311

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
H-Area	H-Retention Basins	281-1H	Operating concrete basin, top of lowest slab = 304.67 ft – 24.48 ft = 280.19 ft msl, slab thickness = 1.75 ft, Basin Bottom elev. = 278.44 ft msl	272	6	S – 6 (default lith.)	• Elev. drawing ref. - W149387
		281-2H	Operating concrete basin, top of lowest slab (sump) = 273.08 ft – 2.5 ft = 270.58 ft msl, slab thickness = 1ft, Basin bottom elev. = 269.58 ft msl	272	0	NA	• Elev. drawing ref. - W149388 • The water table is above the base of the bldg. slab
		281-8H	Operating EPDM geomembrane lined earthen basin with a concrete inlet structure and sump, top of lowest slab (inlet) = 258.33 ft – 7.25 ft = 251.08 ft msl, slab thickness = 1 ft, Basin bottom elev. = 250.08 ft msl	245	6	S – 6 (default lith.)	• Elev. drawing refs. - W238973, W238897, W238899
	H-Area Seepage Basins	904-44G (Basin #1)	Closed earthen basin Basin bottom elev. = 248 ft msl	223	25	S – 5 C – 5 S – 3 C – 3 S – 9	• Elev. drawing refs. - W810310, W810311 • <u>Assuming all H-Area Seepage Basin inventory is in Basin #1</u>
		904-45G (Basin #2)	Closed earthen basin, Basin bottom elev. = 248 ft msl	225	23	S – 5 C – 5 S – 3 C – 3 S – 7	"
		904-46G (Basin #3)	Closed earthen basin, Basin bottom elev. = 237 ft msl	230	7	S – 5 C – 2	"

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
H-Area	H-Area Seepage Basins	904-56G (Basin #4)	Closed earthen basin, Basin bottom elev. = 243 ft msl	233	10	S – 5 C – 5	“
	HB-Line		Located within H-Canyon (see below)	270	18	S – 18 (default lith.)	• Elev. drawing refs. - W147050, W158890
	H-Canyon		Elev. of top of lowest floor slab = 293.67 ft msl, slab thickness = 5.17 ft, Elev. of slab base = 288.50 ft msl	270	18	S – 18	• Elev. drawing refs. - W147050, W158890
	Receiving Basin for Offsite Fuels (RBOF)	244-H 245-H	Elev. of top of lowest floor slab = 255 ft msl, slab thickness = 3.33 ft, Elev. of slab base = 251.67 ft msl	269	0	NA	• 244-H elev. drawing refs. - W232831, W232807 • 245-H elev. drawing ref. - W233586 • The water table is above the base of the bldg. slab
	Sand Filter	294-1H	Low point elev = 277.23 ft msl, slab thickness =2 ft, Elev. of slab base = 275.2 ft msl	260	15	S – 15 (default lith.)	• Elev. drawing ref. - W449996
	Tritium	Bldg. 232-H	Elev. of top of lowest floor slab = 285.67 ft msl, slab thickness = 0.75 ft, Elev. of slab base = 284.92 ft msl	262	23	S – 23 (default lith.)	• Elev. drawing refs. - W159604
		H-Area New Manufacturing (HANM)	Elev. of top of lowest floor slab = 265.08 ft msl, slab thickness = 2.5 ft, Elev. of slab base = 262.58 ft msl	248	14	S – 14 (default lith.)	• Elev. drawing ref. - W812327
		H-Area Old Manufacturing (HAOM)	Elev. of top of lowest floor slab = 300.76 ft msl, slab thickness = 0.33 ft, Elev. of slab base = 300.43 ft msl	258	42	S – 42 (default lith.)	• Elev. drawing ref. - W746418

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
H-Area	Tritium	Vault 217-H	Located within HAOM (see above)	258	42	S – 42 (default lith.)	• Assuming 217-H is the same as HAOM
		Bldg. 234-7H	Elev. of top of lowest floor slab = 300.76 ft msl, slab thickness = 0.5 ft, Elev. of slab base = 300.26 ft msl	252	48	S – 48 (default lith.)	• Elev. drawing ref. - C-CC-H-8058
		Bldg.236-H	Elev. of top of lowest floor slab = 300.76 ft, slab thickness = 2 ft, Elev. of slab base = 298.76 ft msl	258	41	S – 41 (default lith.)	• Elev. drawing ref. - W729389
		Bldgs.237/238-H	For 237-H, elev. of top of lowest floor slab = 300.67 ft msl, slab thickness = 0.5 ft, Elev. of slab base = 300.17 ft msl, For 238-H, elev. of top of lowest floor slab = 300.67 ft msl, slab thickness =0.33 ft, Elev. of slab base = 300.53 ft msl	259	41	S – 41 (default lith.)	• Elev. drawing ref. - S5-2-4210, W237104
	Tritium Extraction Facility	Tritium Extraction Facility (TEF)	Elev. of top of lowest floor slab = 291.5 ft msl, slab thickness = 6 ft, Elev. of slab base = 285.5 ft msl	260	25	S – 25 (default lith.)	• Elev. drawing ref. - C-CC-H-8173
		Bldg. 264-2H	Elev. of top of lowest floor slab = 265.08 ft msl, slab thickness = 6 ft, Elev. of slab base = 259.08 ft msl	260	0	NA	• Elev. drawing refs. - C-CC-H-8120 • The water table is near the base of the bldg. slab
N-Area	Ford Bldg. Seepage Basin	904-91G	Closed earthen basin, Bottom of basin elevation = 295.0 ft msl	251	44	C – 44	• Elev. drawing and VZ lith. refs. - C-CG-G-2657, WSRC-RP-98-4096, Rev.1

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
Reactors	Reactor Bldgs.	Bldg. 105-P	Elev. at surface = 316 ft msl, elev. of top of lowest floor slab = 276 ft msl, slab thickness =10.5 ft, Elev. of slab base = 265.5 ft msl	278	0	NA	• Elev. drawing ref. - W131043 • Used the same lith. as the P-Reactor Seepage Basin • The water table is above the base of the bldg. slab
		Bldg. 105-R	Elev. at surface = 290 ft msl, elev. of top of lowest floor slab = 250 ft msl, slab thickness =10.5 ft, Elev. of slab base = 239.5 ft msl	285	0	NA	• Elev. drawing refs. - W130481 and W133331 • Used the same lith. as the R-Reactor Seepage Basin • The water table is above the base of the bldg. slab
		Bldg. 105-C	Elev. at surface = 286 ft msl, elev. of top of lowest floor slab = 246 ft msl, slab thickness =12 ft, Elev. of slab base = 234 ft msl	215	19	S – 19	• Elev. drawing refs. - W134588, W134728, W134729, W134591 • Used the same lith. as the C-Reactor Seepage Basin
		K-Area Complex (KAC)	Elev. at surface = 270 ft msl, elev. of top of lowest floor slab = 230 ft msl, slab thickness =12.0 ft, Elev. of slab base = 218 ft msl	207	11	C – 11	• Elev. drawing ref. - W133542 • Used the same lith. as the K-Reactor Seepage Basin
		L-Area Complex (LAC)	Elev. at surface = 250 ft msl, elev. of top of lowest floor slab = 210 ft msl, slab thickness =12 ft, Elev. of slab base = 198 ft msl	219	0	NA	• Elev. drawing ref. - W132953 • The water table is above the base of the bldg. slab
	Reactor Seepage Basins	P-Reactor Seepage Basins: 904-61G (Basin #1) 904-62G (Basin #2 904-63G (Basin #3)	Elev. at surface = 326 ft msl (Basin #1), 322 ft msl (Basin #2), 317 ft msl (Basin #3); Bottom of basin = 315 ft msl (Basin #1), 311 ft msl (Basin #2), 306 ft msl (Basin #3)	277	Basin #1 – 38 Basin #2 – 34 Basin #3 - 29	Basin #1 – C – 38 Basin #2 – C – 34 Basin #3 – C – 29	• Elev. and VZ lith. ref. - Unit-Specific Plug-In TER for the P-Reactor Seepage Basins OU, WSRC-RP-2002-4082, Rev.1, Sept. 2002

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
Reactors	Reactor Seepage Basins	R-Reactor Seepage Basins: 904-57G (Basin #1) 904-58G (Basin #2) 904-59G (Basin #3) 904-60G (Basin #4) 904-103G (Basin #5) 904-104G (Basin #6)	Elev. at surface = 320 ft msl (Basin #1), 315 ft msl (Basin #2), 311 ft msl (Basin #3 and Basin #6), 308 ft msl (Basin #4 and Basin #5); Bottom of basin = 310 ft msl (Basin #1), 305 ft msl (Basin #2), 301 ft msl (Basin #3), 301.4 ft msl (Basin #4), 298 ft msl (Basin #5), 294.6 ft msl (Basin #6)	280	Basin #1 – 30 Basin #2 – 25 Basin #3 – 21 Basin #4 – 21 Basin #5 – 18 Basin #6 – 14	Basin #1 – C – 30 Basin #2 – C – 25 Basin #3 – C – 21 Basin #4 – C – 21 Basin #5 – C – 18 Basin #6 – C – 14	• Elev. and VZ lith. ref. - Preliminary Characterization Report Phase I RFI/RI, WSRC-RP-97-196, June 1997 and R-Area OU Scoping Summary, ERD-EN-2006-0100, May 2008
		C-Reactor Seepage Basins: 904-66G (Basin #1) 904-67G (Basin #2) 904-68G (Basin #3)	Elev. at surface = 287 ft msl (Basin #1), 280 ft msl (Basins #2 and #3); Bottom of basin = 280 ft msl (Basin #1), 269 ft msl (Basin #2), 268 ft msl (Basin #3)	211	Basin #1 – 69 Basin #2 – 58 Basin #3 - 57	Basin #1 – S – 69 Basin #2 – S – 58 Basin #3 – S – 57	• Elev. and VZ lith. ref. - Unit-Specific Plug-In Decision Document for the C-Reactor Seepage Basins OU, WSRC-RP-99-4018, Rev.0, Feb. 1999
		K-Reactor Seepage Basin: 904-65G	Elev. at surface = 260-265 ft msl; Bottom of basin = 253 ft msl	204	49	S – 30 C – 19	• Elev. ref. - PCR/FRR for the K-Area Reactor Seepage Basin (904-65G) OU, WSRC-RP-2002-4030, Rev.1, July 2002 • VZ lith. ref. - Preliminary Characterization Report for the K-Area Reactor SB (U), WSRC-RI-96-870, Dec. 1996

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
Reactors	Reactor Seepage Basins	L-Reactor Seepage Basin: 904-64G	Elev. at surface = 233-236 ft msl; Bottom of basin = 222 ft msl	211	11	C – 11	• Elev. and VZ lith. ref. - PCR/FRR for the L-Area Reactor Seepage Basin (904-64G) (U), WSRC-RP-2003-4118, Rev.0, Sept. 2003
	Reactor Cask Car Railroad Tracks	P-Area	NA	NA	NA	NA	• Assuming that the tracks and soil will be removed and placed in reactor bldgs. - VZ thickness NA
		R-Area	NA	NA	NA	NA	"
		C-Area	NA	NA	NA	NA	"
		K-Area	NA	NA	NA	NA	"
		L-Area	NA	NA	NA	NA	"
	P-Reactor Outfalls P02 and P007	NA	NA	NA	NA	NA	---
S-Area	Defense Waste Processing Facility (DWPF)	221-S	Elev. of top of lowest bldg. floor slab = 270 ft msl, slab thickness = 9.75 ft, Elev. of slab base = 260.25 ft msl, Lowest elev. of the chemical process cell = 279.75 ft msl, slab thickness = 0.75 ft, Elev. of chemical process cell base = 279.00 ft msl	243	17	S – 17	• VZ lith. basis – same as Saltstone • Elev. refs. - W755115, W755113, W755100, W757161, W757162 Section K (Catch Tank Area)
		511-S LPPP Facility	Lowest elev. of the pump pit = 247.20 ft msl, slab thickness = 4 ft, Elev. of pump pit = 243.20 ft msl	246	0	NA	• Elev. refs. - W756448, W756449, W756450, W756451 • The water table is above the base of the bldg. slab
		294-S Sand filter	Elev. of top of lowest bldg. floor slab = 255.67 ft msl, slab thickness = 2 ft, Elev. of slab base = 253.76 ft msl	244	10	S – 10 (default lith.)	• Elev. refs. - W762486, W762484

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
TNX Area	TNX Burying Grounds	---	Surface elev. = 145 ft msl, TBG was four trenches at 6-8 ft deep, Deepest trench elev. = 137 ft msl	102	35	S – 15 C – 3 S – 15 C – 2	• Elev. and VZ lith. refs. - 2006 Comp. TNX Area Annual GW and Eff. Monitoring Strategy Report WSRC-RP-2007-4026
	Old TNX Seepage Basin	IPSL	Surface elev. = 152 ft msl basin depth was 10 ft, Elev. of basin bottom = 142 ft msl	99	43	S – 12 C – 3 S – 3 C – 2 S – 7 C – 10 S – 6	"
	TNX-Discharge Gully	---	Surface elev. = 112 ft msl, gully incised 5-20 ft, Deepest part of Discharge Gully at 92 ft msl	99	7	S – 7	"
	TNX Outfall Delta/Swamp	---	Surface elev. = 100 ft msl	93	7	S – 7	"
	Bldg. 677-T (Maintenance Shop, Lab and Pilot Plant) and 678-T (Chemical Processing and Lab)	---	Surface elev. = 152 ft msl	99	53	S – 12 C – 3 S – 3 C – 2 S – 7 C – 10 S – 16	"
Z-Area	Saltstone Facility	Vault 1	---	NA	48	S – 48	• Elev. and VZ lith. refs. - Draft PA modeling report and WSRC-STI-2006-00198, Fig. 5-3, p.91-92 • Lower VZ lith. (below clay)
		Vault 4	---	NA	38	S – 38	"
		Vault 2 Type Cells (64)	---	NA	42	S – 42	"
General Site	Integrator Operable Units	Fourmile Branch[1]	NA	NA	NA	NA	NA because not accounted for in VZ module
		Lower Three Runs	"	NA	NA	NA	"

Table C-4. Facilities and Vadose Zone Information for the SRS Composite Analysis (from Noffsinger et al. 2009)

SRS Area	Primary Facility	Sub-Unit Facilities	Elev. of OU Base	Elev. of Water Table (ft msl)[1]	VZ Thickness (ft)	VZ Lithology[2, 3, 4] (Sand or Clay) – Thickness (ft)	References[5] and Other Notes
General Site	(IOU)	Pen Branch	"	NA	NA	NA	"
		SR and Floodplain Swamp	"	NA	NA	NA	"
		Steel Creek	"	NA	NA	NA	"
		Upper Three Runs	"	NA	NA	NA	"
	GW Closure Units	C-Area, D-Area, F-Area, H-Area, K-Area, L-Area, P-Area, and R-Area GW OUs	NA	NA	NA	NA	NA because not accounted for in VZ module
SRS Future Facilities (not Operational in 2009)	Salt Waste Processing Facility (SWPF)	221-J	Salt process bldg., top of lowest slab = 269.5 ft msl, slab thickness = 9.5 ft, base of bldg. slab = 260.0 ft msl	239	21	S – 21 (default lith.)	• Elev. refs. - C-CC-J-002 R/1, C-CC-J-0088 R/1 • The bulk of the SWPF contamination will be within Bldg. 221-J, and 221-J has less vadose zone; therefore assuming that all of the SWPF inventory is within 221-J
		221-3J	Salt process bldg., top of lowest slab = 271 ft msl, slab thickness = 1 ft, base of bldg. slab = 270 ft msl	239	31	S – 31 (default lith.)	• Elev. refs. - C-CC-J-0033 R/G, C-CC-J-0060 R/E, M-DS-J-00294 R/0, DCN-0396 R/0 • The bulk of the SWPF contamination will be within Bldg. 221-J, and 221-J has less vadose zone; therefore assuming that all of the SWPF inventory is within 221-J
	Mixed Oxide (MOX) Fuel Fabrication Facility	---	Surface elev. = 272.5 ft msl, Base of lowest excavation for primary facility = 250 ft msl	221	29	S – 29 (default lith.)	• Elev. ref. - DCS01-XGP-DS-PLG-G-00328, Rev.3, Duke COGEMA, Stone & Webster drawing

Notes for Table C-4:

[1] Ref. for water table elev. = An Updated Regional Water Table of the Savannah River Site and Related Coverages, WSRC-TR-2003-00250, Rev. 0, R. A. Hiergesell and W. E. Jones, December 2003.

[2] Lithology per descriptors in Geochemical Data Package for Performance Assessment Calculations Related to the Savannah River Site (U), WSRC-TR-2006-00004, Rev. 0, February 2006 - Sandy Sediment Layer (<25 wt-% silt + clay) and Clay Sediment Layer (25 to 45 wt-% silt + clay).

[3] If sandy sediment-clayey sediment descriptors are not available, then sand is assumed to be the transport medium.

[4] Some physical properties of sandy sediment and clayey sediment are summarized in the following table:

Soil Type	Saturated Effective Diffusion Coefficient (cm^2/s)	Total Porosity (%)	Saturation[d] (fraction)	Dry Bulk Density (g/cm^3)	Particle Density (g/cm^3)
Sandy Sediment (Vadose Zone) (<25% wt-%silt + clay)[a]	8.0E-06	38	0.683[b]	1.65	2.66
Clayey Sediment (Vadose Zone) (25 to 45 wt-% silt + clay)[a]	5.3E-06	37	0.858[c]	1.68	2.67
Saturated Zone Soils[a]	Sandy Sediment: 5.3E-06 Clayey Sediment: 4.0E-06	25	1.0	1.04	1.39

Notes:

[a] The listed soil properties are from Phifer et al. 2006, Table 5-18.

[b] Nichols et al. 2000 reported the measurement of soil suction in background soils within the E-Area Low-Level Waste Facility from Advanced Tensiometer locations AT5 and AT7 from April 1999 through January 2000. The soil suctions for vadose zone soils 41 feet deep or less typically ranged from -125 to -175 cm H_2O. Based upon this the saturation of sandy sediment was estimated for a soil suction of -150 cm H_2O from the Phifer et al. 2006 Table 5-20 characteristic curve data for sand. The sand saturation value at -150 cm H_2O of 0.683 was interpolated from the saturations at -143 and -165 cm H_2O, which were 0.689 and 0.671, respectively.

[c] Nichols et al. 2000 reported the measurement of soil suction in background soils within the E-Area Low-Level Waste Facility from Advanced Tensiometer locations AT5 and AT7 from April 1999 through January 2000. The soil suctions for vadose zone soils 41 feet deep or less typically ranged from -125 to -175 cm H_2O. Based upon this the saturation of clayey sediment was estimated for a soil suction of -150 cm H_2O from the Phifer et al. 2006 Table 5-20 characteristic curve data for clay-sand. The clay-sand saturation value at -150 cm H_2O of 0.858 was interpolated from the saturations at -143 and -165 cm H_2O, which were 0.862 and 0.849, respectively.

[d] These saturation values represent background vadose zone soils that do not underlie a closure cap, concrete slab, building, etc.

References for Soil Properties (above table):

Nichols, R. L., Looney, B. B., Flach, G. P., and Rossabi, J. 2000. Recommendations for Phase II Vadose Zone Characterization and Monitoring at the E-Area Disposal "Slit" Trenches and Mega-Trench (U), WSRC-TR-2000-00059. Westinghouse Savannah River Company, Aiken, SC 29808. February 2000.

Phifer, M. A., Millings, M. R., and Flach, G. P. 2006. Hydraulic Property Data Package for the E-Area and Z-Area Soils, Cementitious Materials, and Waste Zones, WSRC-STI-2006-00198, Rev. 0. Washington Savannah River Company, Aiken, SC 29808. September 2006.

[5] With the exception of 'SRS Future Facilities', the referenced SRS drawings in the table are available at the Document Control Registry (DCR) webpage in ShRINE at: http://drawings.srs.gov/. The referenced reports and documents are available on the ACP webpage under 'Administrative Records' or 'Project Files' at: http://erdweb.srs.gov/default.html. The referenced drawings for SWPF and MOX may be obtained by contacting WSRC Engineering Document Control.

Table C-5. Composite Analysis Infiltration Estimates and Distributions (Phifer and Dixon 2009)

Table C-5. Composite Analysis Infiltration Estimates and Distributions (Phifer and Dixon 2009)						
Infiltration Surface	**Year**	**Type Distribution [1]**	**Average Infiltration [2] (in/yr)**	**1,000-year Mean Standard Deviation [3] (in/yr)**	**Minimum (-3σ) [4] (in/yr)**	**Maximum (+3σ) [5] (in/yr)**
Background Soils	Constant	Truncated Normal	15	0.17	14.48	15.52
Clay Cap	0	Truncated Normal	0.362	0.004	0.349	0.374
	100	Truncated Normal	0.413	0.005	0.399	0.428
	300	Truncated Normal	3.047	0.035	2.942	3.152
	550	Truncated Normal	7.894	0.091	7.622	8.166
	1,000	Truncated Normal	12.037	0.138	11.623	12.451
	1,800	Truncated Normal	13.762	0.158	13.289	14.236
	3,400	Truncated Normal	14.035	0.161	13.552	14.518
	5,600	Truncated Normal	14.079	0.161	13.595	14.563
	10,000	Truncated Normal	14.093	0.162	13.608	14.578
Geosynthetic Cap	0	Truncated Normal	0.00088	0.00001	0.001	0.001
	100	Truncated Normal	0.010	0.0001	0.010	0.011
	180	Truncated Normal	0.173	0.002	0.167	0.179
	290	Truncated Normal	0.371	0.004	0.358	0.384
	300	Truncated Normal	0.503	0.006	0.486	0.521
	340	Truncated Normal	0.996	0.011	0.961	1.030
	380	Truncated Normal	1.457	0.017	1.407	1.507
	560	Truncated Normal	3.230	0.037	3.119	3.341
	1,000	Truncated Normal	7.014	0.080	6.773	7.255
	1,800	Truncated Normal	10.651	0.122	10.284	11.017
	2,623	Truncated Normal	11.472	0.132	11.078	11.867

Table C-5. Composite Analysis Infiltration Estimates and Distributions (Phifer and Dixon 2009)

Infiltration Surface	Year	Type Distribution [1]	Average Infiltration [2] (in/yr)	1,000-year Mean Standard Deviation [3] (in/yr)	Minimum (-3σ) [4] (in/yr)	Maximum (+3σ) [5] (in/yr)
Geo-synthetic Cap	3,200	Truncated Normal	11.530	0.132	11.133	11.926
	5,600	Truncated Normal	11.632	0.133	11.232	12.032
	10,000	Truncated Normal	11.670	0.134	11.269	12.072
Concrete Cap	0	Truncated Normal	0.061	0.0007	0.059	0.063
	100	Truncated Normal	0.066	0.0008	0.064	0.068
	300	Truncated Normal	0.962	0.011	0.929	0.995
	550	Truncated Normal	2.341	0.027	2.260	2.422
	1,000	Truncated Normal	5.692	0.065	5.496	5.888
	1,800	Truncated Normal	10.997	0.126	10.619	11.375
	2,470	Truncated Normal	13.256	0.152	12.800	13.712
	2,805	Truncated Normal	13.278	0.152	12.821	13.735
	3,400	Truncated Normal	13.414	0.154	12.953	13.875
	5,600	Truncated Normal	13.682	0.157	13.211	14.153
	10,000	Truncated Normal	13.752	0.158	13.279	14.225

Notes for Table C-5:

1 All infiltration distributions are assumed to be normal based upon the analysis by Shine 2008 for the Background Soils.

2 Average Infiltration for the Background Soils obtained from Shine 2008 Conclusions and Recommendations; for Clay Cap obtained from Phifer 2003 Table 5.2-1; for Geosynthetic Cap obtained from Phifer et al. 2007 Table 81; and for Concrete Cap obtained from WSRC 2008a Figure 12 for the Concrete Cap.

3 1,000-year Mean Standard Deviation for the Background Soils obtained from Shine 2008 Conclusions and Recommendations; and the 1,000-year Mean Standard Deviation for the Clay Cap, Geosynthetic Cap, and Concrete Cap determined by multiplying their average infiltration by Background Infiltration 1,000-year Mean Standard Deviation (0.17 in/yr) divided by Background Infiltration Average Infiltration (15 in/yr).

4 Minimum Infiltration equals the Average Infiltration minus three times the standard deviation.

5 Maximum Infiltration equals the Average Infiltration plus three times the standard deviation.

Table C-6. Isotopic Species in GoldSim™ CA Transport and Decay Model

CA Species	Non-CA Species	CA Species	Non-CA Species	CA Species	Non-CA Species
Ac227		***Fr223***		**Pu244**	
Ac225		**H3**		***Ra223***	
Ac228		**I129**		***Ra224***	
Ag108		**K40**		***Ra225***	
Ag108m		**Lu174**		**Ra226**	
Al26		**Mo93**		***Ra228***	
Am241		***Nb93m***		***Rn219***	
Am242		**Nb94**		***Rn220***	
Am242m		**Ni59**		***Rn222***	
Am243		**Ni63**			Sb126
At217		**Np237**			Sb126m
At218		***Np238***		**Se79**	
Ba137m		***Np239***			Sm147
Bi210		***Np240m***			Sm151
Bi210m		**Pa231**			Sn126
Bi211		***Pa233***		**Sr90**	
Bi212		***Pa234m***		**Tc99**	
Bi213		***Pb209***		***Th227***	
Bi214		**Pb210**		***Th228***	
C14		***Pb211***		**Th229**	
Ca41		***Pb212***		**Th230**	
Cf249		***Pb214***		***Th231***	
	Cf251	**Pd107**		**Th232**	
Cl36		***Po210***		***Th234***	
	Cm243	***Po211***		***Tl206***	
Cm242		***Po212***		***Tl207***	
Cm244		***Po213***		***Tl208***	
Cm245		***Po214***		***Tl209***	
Cm246		***Po215***			U232
Cm247		***Po216***		**U233**	
Cm248		***Po218***		**U234**	
	Co60	**Pt193**		**U235**	
Cs135		**Pu238**		**U236**	
Cs137		**Pu239**		***U237***	
	Eu152	**Pu240**		**U238**	
	Eu154	**Pu241**		***U240***	
	Eu155	**Pu242**		***Y90***	
	Gd152	***Pu243***		**Zr93**	
Fr221					

Note for Table C-6:

Bold Black Lettering: 49 CA parent radionuclides.
Bold Blue Italic Lettering: 3 CA radionuclide daughters with half-lives >3 yr.
Bold Green Italic Lettering: 50 CA radionuclide daughters with half-lives <3 yr.

Table C-7. Radionuclide-Specific Physical & Chemical Parameters (102 CA Parents & Daughters)

Table C-7. Radionuclide-Specific Physical & Chemical Parameters (102 CA Parents & Daughters)			
GoldSim™ Element ID	**Half-Lives**	**Mol_Wt**	**K_d_SandySoil _DoseSpecies**
Radionuclide	**Half-Life[a] (yr)**	**Molecular Weight[b] (g/mol)**	**Soil K_d[c] (mL/g)**
Ac225	2.74E-02	225	1100
Ac227	2.18E+01	227	1100
Ac228	7.02E-04	228	1100
Ag108	4.51E-06	108	60
Ag108m	4.38E+02	108	60
Al26	7.17E+05	26	1300
Am241	4.32E+02	241	1100
Am242	1.83E-03	242	1100
Am242m	1.41E+02	242	1100
Am243	7.37E+03	243	1100
At217	1.01E-09	217	1.00E-09
At218	4.75E-08	218	1.00E-09
Ba137m	4.85E-06	137	5
Bi210	1.37E-02	210	1100
Bi210m	3.04E+06	210	1100
Bi211	4.07E-06	211	1100
Bi212	1.15E-04	212	1100
Bi213	8.67E-05	213	1100
Bi214	3.78E-05	214	1100
C14	5.70E+03	14	10
Ca41	1.02E+05	41	5
Cf249	3.51E+02	249	1100
Cl36	3.01E+05	36	1.00E-09
Cm242	4.46E-01	242	1100
Cm244	1.81E+01	244	1100
Cm245	8.50E+03	245	1100
Cm246	4.76E+03	246	1100
Cm247	1.56E+07	247	1100
Cm248	3.48E+05	248	1100
Cs135	2.30E+06	135	50
Cs137	3.00E+01	137	50
Fr221	9.32E-06	221	50
Fr223	4.18E-05	223	50
H3	1.23E+01	3	1.00E-09

Table C-7. Radionuclide-Specific Physical & Chemical Parameters (102 CA Parents & Daughters)			
GoldSim™ Element ID	Half-Lives	Mol_Wt	K_d_SandySoil _DoseSpecies
Radionuclide	Half-Life[a] (yr)	Molecular Weight[b] (g/mol)	Soil K_d[c] (mL/g)
I129	1.57E+07	129	1.00E-09
K40	1.25E+09	40	10
Lu174	3.31E+00	174	1100
Mo93	4.00E+03	93	6
Nb93m	1.61E+01	93	1.00E-09
Nb94	2.03E+04	94	1.00E-09
Ni59	7.61E+04	59	7
Ni63	1.00E+02	63	7
Np237	2.14E+06	237	0.6
Np238	5.80E-03	238	0.6
Np239	6.45E-03	239	0.6
Np240m	1.37E-05	240	0.6
Pa231	3.28E+04	231	0.6
Pa233	7.38E-02	233	0.6
Pa234m	2.22E-06	234	0.6
Pb209	3.71E-04	209	2000
Pb210	2.22E+01	210	2000
Pb211	6.86E-05	211	2000
Pb212	1.21E-03	212	2000
Pb214	5.10E-05	214	2000
Pd107	6.50E+06	107	7
Po210	3.79E-01	210	2000
Po211	1.64E-08	211	2000
Po212	9.47E-15	212	2000
Po213	1.16E-13	213	2000
Po214	5.21E-12	214	2000
Po215	5.64E-11	215	2000
Po216	4.59E-09	216	2000
Po218	5.89E-06	218	2000
Pt193	5.00E+01	193	900
Pu238	8.78E+01	238	270
Pu239	2.41E+04	239	270
Pu240	6.56E+03	240	270
Pu241	1.43E+01	241	270
Pu242	3.75E+05	242	270
Pu243	5.65E-04	243	270
Pu244	8.00E+07	244	270
Ra223	3.13E-02	223	5

Table C-7. Radionuclide-Specific Physical & Chemical Parameters (102 CA Parents & Daughters)			
GoldSim™ Element ID	**Half-Lives**	**Mol_Wt**	**K_d_SandySoil _DoseSpecies**
Radionuclide	**Half-Life[a] (yr)**	**Molecular Weight[b] (g/mol)**	**Soil K_d[c] (mL/g)**
Ra224	1.00E-02	224	5
Ra225	4.08E-02	225	5
Ra226	1.60E+03	226	5
Ra228	5.75E+00	228	5
Rn219	1.25E-07	219	1.00E-09
Rn220	1.76E-06	220	1.00E-09
Rn222	1.05E-02	222	1.00E-09
Se79	2.95E+05	79	1000
Sr90	2.89E+01	90	5
Tc99	2.11E+05	99	0.6
Th227	5.13E-02	227	900
Th228	1.91E+00	228	900
Th229	7.34E+03	229	900
Th230	7.54E+04	230	900
Th231	2.91E-03	231	900
Th232	1.41E+10	232	900
Th234	6.60E-02	234	900
Tl206	7.99E-06	206	60
Tl207	9.07E-06	207	60
Tl208	5.80E-06	208	60
Tl209	4.11E-06	209	60
U233	1.59E+05	233	200
U234	2.46E+05	234	200
U235	7.04E+08	235	200
U236	2.34E+07	236	200
U237	1.85E-02	237	200
U238	4.47E+09	238	200
U240	1.61E-03	240	200
Y90	7.30E-03	90	1100
Zr93	1.53E+06	93	900

Notes for Table C-7:

[a] Source: Tuli 2005
[b] Source: KAPL 2006
[c] Source: McDowell-Boyer and Kaplan 2009
g gram
K_d soil/water adsorption coefficient
mL milliliter
mol mole
yr year

Table C-8. Branching Fractions for 102 CA Parents & Daughters (Phifer et al. 2009)

Table C-8. Branching Fractions for 102 CA Parents & Daughters (Phifer et al. 2009)			
GoldSim™ Element ID Branching Fractions			
Radionuclide [a]	Decay Mode [b]	Branching Fractions [c]	Daughter
Ac225	α	1	Fr221
Ac227	β	0.9862	Th227
	α	0.0138	Fr223
Ac228	β	1	Th228
Ag108	β	0.9715	Cd
	ε	0.0285	Pd
Ag108m	ε	0.913	Pd
	IT	0.087	Ag108
Al26	ε	1	Mg
Am241	α	1	Np237
Am242	β	0.827	Cm242
	ε	0.173	Pu242
Am242m	IT	0.9955	Am242
	α	0.0045	Np238
Am243	α	1	Np239
At217	α	0.9999	Bi313
At218	α	1 [d]	Bi214
Ba137m	IT	1	Ba
Bi210	β	1	Po210
Bi210m	α	1	Tl206
Bi211	α	0.9972	Tl207
	β	0.0028	Po211
Bi212	β	0.6406	Po212
	α	0.3594	Tl208
Bi213	β	0.9791	Po213
	α	0.0209	Tl209
Bi214	β	0.9998	Po214
C14	β	1	N
Ca41	ε	1	K
Cf249	α	1	Cm245
Cl36	β	0.981	Ar
	ε	0.019	S
Cm242	α	1	Pu238
Cm244	α	1	Pu240
Cm245	α	1	Pu241
Cm246	α	0.9997	Pu242
Cm247	α	1	Pu243
Cm248	α	0.9161	Pu244
Cs135	β	1	Ba
Cs137	β	1	Ba137m
Fr221	α	1 [d]	At217
Fr223	β	1 [d]	Ra223

Table C-8. Branching Fractions for 102 CA Parents & Daughters (Phifer et al. 2009)			
GoldSim™ Element ID Branching Fractions			
Radionuclide [a]	Decay Mode [b]	Branching Fractions [c]	Daughter
H3	β	1	He
I129	β	1	Xe
K40	β	0.8928	Ca
	ε	0.1072	Ar
Lu174	ε	1	Yb
Mo93	ε	1	Nb93m
Nb93m	IT	1	Nb
Nb94	β	1	Mo
Ni59	ε	1	Co
Ni63	β	1	Cu
Np237	α	1	Pa233
Np238	β	1	Pu238
Np239	β	1	Pu239
Np240m	β	1 [d]	Pu240
Pa231	α	1	Ac227
Pa233	β	1	U233
Pa234m	β	0.9984	U234
Pb209	β	1	Bi
Pb210	β	1	Bi210
Pb211	β	1	Bi211
Pb212	β	1	Bi212
Pb214	β	1	Bi214
Pd107	β	1	Ag
Po210	α	1	Pb
Po211	α	1	Pb
Po212	α	1	Pb
Po213	α	1	Pb209
Po214	α	1	Pb210
Po215	α	1	Pb211
Po216	α	1	Pb212
Po218	α	0.9998	Pb214
	β	0.0002	At218
Pt193	ε	1	Ir
Pu238	α	1	U234
Pu239	α	1	U235
Pu240	α	1	U236
Pu241	β	1	Am241
	α	0.000025	U237
Pu242	α	1	U238
Pu243	β	1	Am243
Pu244	α	0.9988	U240
Ra223	α	1	Rn219
Ra224	α	1	Rn220
Ra225	β	1	Ac225

Table C-8. Branching Fractions for 102 CA Parents & Daughters (Phifer et al. 2009)			
GoldSim™ Element ID Branching Fractions			
Radionuclide [a]	Decay Mode [b]	Branching Fractions [c]	Daughter
Ra226	α	1	Rn222
Ra228	β	1	Ac228
Rn219	α	1	Po215
Rn220	α	1	Po216
Rn222	α	1	Po218
Se79	β	1	Br
Sr90	β	1	Y90
Tc99	β	1	Ru
Th227	α	1	Ra223
Th228	α	1	Ra224
Th229	α	1	Ra225
Th230	α	1	Ra226
Th231	β	1	Pa231
Th232	α	1	Ra228
Th234	β	1	Pa234m
Tl206	β	1	Pb
Tl207	β	1	Pb
Tl208	β	1	Pb
Tl209	β	1	Pb209
U233	α	1	Th229
U234	α	1	Th230
U235	α	1	Th231
U236	α	1	Th232
U237	β	1	Np237
U238	α	1	Th234
U240	β	1	Np240m
Y90	β	1	Zr
Zr93	β	1	Nb93m

Notes for Table C-8:

[a] **Black** text = 49 CA parent radionuclide; **Blue** text = three CA radionuclide daughters with half-lives > 3 yr; **Green** Text = 50 CA radionuclide daughters with half-lives < 3 yr.

[b] Source: Tuli 2005.

[c] Decay Mode:

α = alpha particle
β = beta particle
ε = electron capture
IT = isomeric transition

[d] Rounded to 1 from value greater than 0.99; radionuclide dose from other branch insignificant.

Table C-9. Supplemental Branching Fraction Vector (50 Shorter-Lived Daughters)

Table C-9. Supplemental Branching Fraction Vector (50 Shorter-Lived Daughters)		
GoldSim™ DoseSpecies Array [a]	Supplemental Branching Fractions Vector [b]	Closest Preceding Decay Chain Member in Transport Module [a]
Ac225	1	Th229
Ac228	1	Ra228
Ag108	0.087	Ag108m
Am242	0.9955	Am242m
At217	1	Th229
At218	0.0002	Ra226
Ba137m	1	Cs137
Bi210	1	Pb210
Bi211	1	Ac227
Bi212	1	Ra228
Bi213	0.9999	Th229
Bi214	1	Ra226
Cm242	0.8233	Am242m
Fr221	1	Th229
Fr223	0.0138	Ac227
Np238	0.0045	Am242m
Np239	1	Am243
Np240m	0.9988	Pu244
Pa233	1	Np237
Pa234m	1	U238
Pb209	1	Th229
Pb211	1	Ac227
Pb212	1	Ra228
Pb214	0.9998	Ra226
Po210	1	Pb210
Po211	0.0028	Ac227
Po212	0.6406	Ra228
Po213	0.98	Th229
Po214	0.9998	Ra226
Po215	1	Ac227
Po216	1	Ra228
Po218	1	Ra226
Pu243	1	Cm247
Ra223	1	Ac227
Ra224	1	Ra228
Ra225	1	Th229
Rn219	1	Ac227
Rn220	1	Ra228
Rn222	1	Ra226

Table C-9. Supplemental Branching Fraction Vector (50 Shorter-Lived Daughters)		
GoldSim™ DoseSpecies Array [a]	**Supplemental Branching Fractions Vector [b]**	**Closest Preceding Decay Chain Member in Transport Module [a]**
Th227	0.9862	**Ac227**
Th228	1	**Ra228**
Th231	1	**U235**
Th234	1	**U238**
Tl206	1	**Bi210m**
Tl207	0.9972	**Ac227**
Tl208	0.3594	**Ra228**
Tl209	0.0209	**Th229**
U237	0.000025	**Pu241**
U240	0.9988	**Pu244**
Y90	1	**Sr90**

Notes for Table C-9:

[a] **Black** text = CA parent radionuclides; **Blue** text = CA radionuclide daughters with half-lives > 3 yr; **Green** Text = 50 CA radionuclide daughters with half-lives < 3 yr.

[b] Source: Tuli 2005.

Table C-10. Material Property Estimates and Distributions (Phifer and Dixon 2009)

Material	Property	Units	Distribution Type	Argument 1	Argument 2	Argument 3	Argument 4	Source Note
			Constant data	Value				
			Truncated normal	Mean	Standard Deviation	Minimum (-3σ)	Maximum (+3σ)	
			Triangular	Minimum	Most Likely	Maximum		
Vadose Zone Sandy Soil	Porosity		Truncated normal	0.380	0.008	0.360	0.400	1
	Dry Bulk Density	g/cm^3	Truncated normal	1.650	0.022	1.590	1.710	1
	Particle Density	g/cm^3	Truncated normal	2.660	0.006	2.640	2.680	1
	Background Saturation (15 in/yr infiltration)		Triangular	0.662	0.683	0.705		2
	Tortuosity		Triangular	0.227	0.500	1.000		3
Vadose Zone Clayey Soil	Porosity		Truncated normal	0.370	0.011	0.340	0.400	4
	Dry Bulk Density	g/cm^3	Truncated normal	1.680	0.028	1.600	1.760	4
	Particle Density	g/cm^3	Truncated normal	2.670	0.010	2.640	2.700	4
	Background Saturation (15 in/yr infiltration)		Triangular	0.843	0.858	0.873		2
	Tortuosity		Triangular	0.202	0.331	0.557		3
Vadose Zone Intact Concrete	Porosity		Truncated normal	0.221	0.013	0.172	0.250	5
	Dry Bulk Density	g/cm3	Truncated normal	2.060	0.100	1.760	2.360	5
	Particle Density	g/cm3	Truncated normal	2.610	0.150	2.160	3.060	5
	Background Saturation (15 in/yr infiltration)		Constant data	1.000				6
	Tortuosity		Triangular	0.011	0.050	0.217		7
Saturated Sand	Porosity		Truncated normal	0.250	0.009	0.225	0.276	8
	Dry Bulk Density	g/cm3	Truncated normal	1.040	0.024	0.968	1.112	8
	Particle Density	g/cm3	Truncated normal	1.390	0.006	1.373	1.407	8
	Saturation		Constant data	1.000				9
	Tortuosity		Triangular	0.202	0.331	0.557		10

Table C-10. Material Property Estimates and Distributions (Phifer and Dixon 2009) (continued)

Material	Property	Units	Distribution Type	Argument 1	Argument 2	Argument 3	Argument 4	Source
			Constant data	Value				
			Truncated normal	Mean	Standard Deviation	Minimum (-3σ)	Maximum (+3σ)	
			Triangular	Minimum	Most Likely	Maximum		
Saturated Clay	Porosity		Truncated normal	0.250	0.009	0.225	0.276	8
	Dry Bulk Density	g/cm^3	Truncated normal	1.040	0.024	0.968	1.112	8
	Particle Density	g/cm^3	Truncated normal	1.390	0.006	1.373	1.407	8
	Saturation		Constant data	1.000				9
	Tortuosity		Triangular	0.176	0.250	0.368		10
Saturated Intact Concrete	Porosity		Truncated normal	0.221	0.013	0.172	0.250	11
	Dry Bulk Density	g/cm^3	Truncated normal	2.060	0.100	1.760	2.360	11
	Particle Density	g/cm^3	Truncated normal	2.610	0.150	2.160	3.060	11
	Saturation		Constant data	1.000				9
	Tortuosity		Triangular	0.011	0.050	0.217		11
Water	Diffusivity	cm^2/s	Constant data	0.000016				12
Savannah River Augusta	Flow	cfs	Truncated normal	9198	86.00	8940.00	9456.00	13
Upper Three Runs	Flow	cfs	Truncated normal	237	1.35	232.94	241.06	13
Fourmile Branch	Flow	cfs	Truncated normal	32	0.41	30.78	33.22	13
Pen Branch / Steel Creek	Flow	cfs	Truncated normal	89	1.32	85.04	92.96	13
Lower Three Runs	Flow	cfs	Truncated normal	164	1.91	158.28	169.72	13
Savannah River 301 Bridge	Flow	cfs	Truncated normal	10175	102.63	9867.11	10482.89	13
Natural Infiltration	Flow	in/yr	Truncated normal	15	0.17	14.48	15.52	14
Concrete 1st Stage	# of exchanges		Triangular	1	50	100		15
Concrete 2nd Stage	# of exchanges		Triangular	100	500	1000		15
Concrete 3rd Stage	# of exchanges		Triangular	1000	7000	10000		15

Source Notes for Table C-10:

Source Note	Source Description
1	Vadose zone sandy soil porosity, dry bulk density, and particle density distribution taken as that of sand (<25% mud) from Phifer et al. 2006 Tables 5-15 (standard deviation, minimum, and maximum) and 5-18 (mean).
2	Nichols et al. 2000 reported the measurement of soil suction in background soils within the E-Area Low-Level Waste Facility from advanced tensiometer locations AT5 and AT7 from April 1999 through January 2000. The soil suctions for vadose zone soils 41 feet deep or less typically ranged from -125 to -175 cm H_2O. Based upon this the saturation of sandy soil and clayey soil were estimated at a soil suction of -150 cm H_2O as the most likely from the Phifer et al. 2006 Table 5-20 characteristic curve data for sand and clay-sand, respectively. The saturation at -175 cm H_2O (minimum) and at -125 cm H_2O (maximum) were determined in a like manner from the Phifer et al. 2006 Table 5-20 characteristic curve data for sand and clay-sand. The average of the minimum and maximum deviation from the most likely value (i.e., 0.022 for sand and 0.015 for clay-sand) was used to determine the saturation distribution (minimum and maximum).
3	Vadose zone sandy soil and clayey soil tortuosity distributions were calculated from the effective diffusion coefficient (D_e) distribution for sand and clay-sand, respectively, presented in Phifer et al. 2006 Table 5-17 (minimum and maximum) and Table 5-18 (most likely) per the following equation $\zeta = D_m/D_e$, where D_m = molecular diffusion coefficient in water taken as 1.6e-05 cm/s (Phifer et al. 2006 Section 5.2.5).
4	Vadose zone clayey soil porosity, dry bulk density, and particle density distribution taken as clay-sand (25-50% mud) from Phifer et al. 2006 Tables 5-15 (standard deviation, minimum, and maximum) and 5-18 (mean).
5	Vadose zone intact concrete mean porosity, dry bulk density, and particle density were taken from Phifer et al. 2006 Table 6-47 as that of E-Area CIG concrete mats which represents low quality concrete. The standard deviation, minimum, and maximum values of porosity (Table 6-52), dry bulk density (Table 6-53), and particle density (Table 6-54) were also taken as that of E-Area CIG concrete mats which represents low quality concrete from Phifer et al. 2006.
6	Nichols et al. 2000 reported the measurement of soil suction in background soils within the E-Area Low-Level Waste Facility from advanced tensiometer locations AT5 and AT7 from April 1999 through January 2000. The soil suctions for vadose zone soils 41 feet deep or less typically ranged from -125 to -175 cm H_2O. It is assumed that concrete imbedded in such soils would come to equilibrium with the soil in terms of suction levels. Based upon the Phifer et al. 2006 Table 6-48 characteristic curve data for low quality concrete, the saturation of concrete throughout this soil suction range is 1.0 (i.e., saturated).

Source Notes for Table C-10 (continued):

Source Note	Source Description
7	Vadose zone intact concrete tortuosity distribution was calculated from the effective diffusion coefficient (D_e) distribution for E-Area CIG concrete mats (i.e., low quality concrete) presented in Phifer et al. 2006 Table 6-59 (minimum, most likely, and maximum) per the following equation $\zeta = D_m/D_e$, where D_m = molecular diffusion coefficient in water taken as 1.6e-05 cm/s (Phifer et al. 2006 Section 5.2.5).
8	Saturated sand and saturated clay mean porosity, dry bulk density, and particle density taken as that of saturated zone soils from Phifer et al. 2006 Table 5-18. Phifer et al. 2006 Table 5-18 for saturated zone soils equated the saturated effective diffusion coefficient of saturated sand with clay-sand (25-50% mud) and saturated clay with clay (>50% mud). Therefore the standard deviation for saturated sand and saturated clay porosity, dry bulk density, and particle density was calculated based upon proportionality with the average standard deviation for clay-sand and clay. Clay-sand and clay standard deviations obtained from Phifer et al. 2006 Table 5-15. The minimum was taken as the mean minus three times the calculated standard deviation, and the maximum was taken as the mean plus three times the calculated standard deviation.
9	By definition of saturated.
10	Saturated zone sand and saturated zone clay effective diffusion coefficient has been equated with clay-sand (25-50% mud) and clay (>50% mud) in Phifer et al. 2006 Table 5-18. Saturated zone sand and saturated zone clay tortuosity distributions were calculated from the effective diffusion coefficient (D_e) distribution for clay-sand (25-50% mud) and clay (>50% mud), respectively, presented in Phifer et al. 2006 Table 5-17 (minimum and maximum) and Table 5-18 (most likely) per the following equation $\zeta = D_m/D_e$, where D_m = molecular diffusion coefficient in water taken as 1.6e-05 cm/s (Phifer et al. 2006 Section 5.2.5).
11	Taken as the same as for vadose zone intact concrete.
12	Phifer et al. 2006 Section 5.2.5.
13	Average from Jones 2009 Table 3-1; standard deviation from Shine 2009 Table 4.0.2; the minimum was taken as the mean minus three times the calculated standard deviation, and the maximum was taken as the mean plus three times the calculated standard deviation.
14	Average and standard deviation from Shine 2008 executive summary; the minimum was taken as the mean minus three times the calculated standard deviation, and the maximum was taken as the mean plus three times the calculated standard deviation.
15	Kaplan 2006 Section 4.2.1 for concrete 1st stage; Section 4.2.2 for concrete 2nd stage; Section 4.2.3 for concrete 3rd stage.

Table C-11. Saturation Estimates and Distributions (Phifer and Dixon 2009)

Material	Infiltration (in/yr)	Minimum Saturation [1, 2] (fraction)	Most Likely Saturation [3] (fraction)	Maximum Saturation [4, 5] (fraction)
Sandy Soil	20	0.674	0.696	0.717
	15	0.662	0.683	0.705
	10	0.645	0.666	0.688
	8	0.636	0.658	0.679
	6	0.624	0.646	0.668
	4	0.609	0.631	0.652
	2	0.585	0.606	0.628
	1	0.563	0.585	0.607
	0.5	0.546	0.568	0.590
	0.1	0.524	0.545	0.567
	0.01	0.515	0.537	0.559
	0.001	0.514	0.536	0.558
Clayey Soil	20	0.855	0.870	0.886
	15	0.843	0.858	0.873
	10	0.826	0.841	0.857
	8	0.818	0.833	0.848
	6	0.807	0.823	0.838
	4	0.794	0.809	0.824
	2	0.774	0.789	0.804
	1	0.757	0.772	0.787
	0.5	0.744	0.759	0.774
	0.1	0.724	0.740	0.755
	0.01	0.715	0.731	0.746
	0.001	0.714	0.729	0.745

Notes for Table C-11:

[1] Sandy Soil minimum saturation equals the Most Likely saturation minus 0.022 (see Table C-10 Source Note 2).

[2] Clayey Soil minimum saturation equals the Most Likely saturation minus 0.015 (see Table C-10 Source Note 2).

[3] Most Likely Saturation determined through deterministic PORFLOW modeling described in Section 9.2.

[4] Sandy Soil maximum saturation equals the Most Likely saturation plus 0.022 (see Table C-10 Source Note 2).

[5] Clayey Soil maximum saturation equals the Most Likely saturation plus 0.015 (see Table C-10 Source Note 2).

Table C-12. Best Value K_d's Recommended for Use in Composite Analysis (McDowell-Boyer and Kaplan 2009)[5]

Table C-12. Best Value K_d's Recommended for Use in Composite Analysis (McDowell-Boyer and Kaplan 2009)[5]											
	Soil K_d (mL/g) [1]		Oxidizing Cement K_d (mL/g) [1]			Reducing Cement K_d (mL/g) [1]			K_d Reference		
Element	**Sandy**	**Clayey**	**Young**	**Middle**	**Old**	**Young**	**Middle**	**Old**	**Sand/Clay**	**Concrete**	**Notes [2]**
Ac	1100	8500	6000	6000	600	5000	5000	1000	Kaplan 2007a	Kaplan and Coates 2007	NA
Ag	60	150	1	1	0.1	1	1	0.1	Kaplan 2007b	Kaplan 2007b, Kaplan and Coates 2007	NA
Al	1300	1300	6000	6000	600	5000	5000	1000	Kaplan 2007b	Kaplan and Coates 2007	NA
Am	1100	8500	6000	6000	600	5000	5000	1000	Kaplan 2007a	Kaplan and Coates 2007	NA
At	1.00E-09	0.6	8	15	4	2	10	4	Kaplan 2007a	Kaplan and Coates 2007	NA
Ba	5	17	100	100	70	0.5	3	20	Kaplan 2007a	Kaplan and Coates 2007	NA
Bi	1100	8500	6000	6000	600	5000	5000	1000	Kaplan 2007a [3]	Kaplan and Coates 2007 [3]	trivalent, analog Am
C	10	400	3000	3000	3000	3000	3000	3000	Roberts and Kaplan 2008	Roberts and Kaplan 2008, Bradbury and Sarott 1995	Reducing cement assigned K_d consistent with Bradbury and Sarott 1995
Ca	5	17	3	30	15	0.5	3	20	Kaplan 2007a [3]	Kaplan and Coates 2007 [3]	alkali earth metal, analog Sr

Table C-12. Best Value K_d's Recommended for Use in Composite Analysis (McDowell-Boyer and Kaplan 2009)[5]

	Soil K_d (mL/g) [1]		Oxidizing Cement K_d (mL/g) [1]			Reducing Cement K_d (mL/g) [1]			K_d Reference		
Element	**Sandy**	**Clayey**	**Young**	**Middle**	**Old**	**Young**	**Middle**	**Old**	**Sand/Clay**	**Concrete**	**Notes [2]**
Cd	4	10	4000	4000	1000	5000	5000	1000	Kaplan 2007b	Kaplan and Coates 2007	NA
Cf	1100	8500	6000	6000	600	5000	5000	1000	Kaplan 2007a	Kaplan and Coates 2007	NA
Cl	1.0E-09	1.0E-09	20	20	2	20	20	2	Kaplan 2007a	Kaplan and Coates 2007	NA
Cm	1100	8500	6000	6000	600	5000	5000	1000	Kaplan 2007a	Kaplan and Coates 2007	NA
Cs	50	250	2	20	10	1.0E-09	2	10	Kaplan 2007a	Kaplan and Coates 2007	NA
Fr	50	250	2	20	10	1.0E-09	2	10	Kaplan 2007a	Kaplan and Coates 2007	NA
H	1.0E-09	1.0E-09	1.0E-09	1.0E-09	1.0E-09	1.0E-09	1.0E-09	1.0E-09	Kaplan 2007a	Kaplan and Coates 2007	NA
I	1.0E-09	0.6	8	15	4	5	9	1.0E-09	Kaplan 2007a	Kaplan and Coates 2007, Kaplan et al. 2008a	Reducing cement assigned K_d from Kaplan et al. 2008a
K	10	60	2	20	10	1.0E-09	2	10	Kaplan 2007b	Kaplan and Coates 2007	NA
Lu	1100	8500	6000	6000	600	5000	5000	1000	Kaplan 2007a [3]	Kaplan and Coates 2007 [3]	trivalent transition metal, analog Am

Table C-12. Best Value K_d's Recommended for Use in Composite Analysis (McDowell-Boyer and Kaplan 2009)[5]											
	Soil K_d (mL/g) [1]		Oxidizing Cement K_d (mL/g) [1]			Reducing Cement K_d (mL/g) [1]			K_d Reference		
Element	**Sandy**	**Clayey**	**Young**	**Middle**	**Old**	**Young**	**Middle**	**Old**	**Sand/Clay**	**Concrete**	**Notes [2]**
Mo	6	120	0.1	0.1	0.1	0.1	0.1	0.1	Kaplan 2007b	Kaplan 2007b, Kaplan and Coates 2007	NA
Nb	1.0E-09	1.0E-09	1000	1000	500	1000	1000	500	Kaplan 2007a	Kaplan 2007a, Kaplan and Coates 2007	NA
Ni	7	30	4000	4000	1000	5000	5000	1000	Kaplan 2007a	Kaplan and Coates 2007	NA
Np	0.6	35	1600	1600	250	4000	4000	3000	Kaplan 2007a	Kaplan and Coates 2007, Kaplan et al. 2008a	Reducing cement assigned K_d from Kaplan et al. 2008a
Pa	0.6	35	1600	1600	250	5000	5000	500	Kaplan 2007a	Kaplan and Coates 2007	NA
Pb	2000	5000	500	500	250	500	500	250	Kaplan 2007a	Kaplan 2007a, Kaplan and Coates 2007	NA
Pd	7	30	4000	4000	1000	5000	5000	1000	Kaplan 2007b	Kaplan and Coates 2007	NA
Po	2000	5000	500	500	250	500	500	250	Kaplan 2007a	Kaplan 2007a, Kaplan and Coates 2007	NA

Table C-12. Best Value K_d's Recommended for Use in Composite Analysis (McDowell-Boyer and Kaplan 2009)[5]

	Soil K_d (mL/g) [1]		Oxidizing Cement K_d (mL/g) [1]			Reducing Cement K_d (mL/g) [1]			K_d Reference		
Element	**Sandy**	**Clayey**	**Young**	**Middle**	**Old**	**Young**	**Middle**	**Old**	**Sand/Clay**	**Concrete**	**Notes [2]**
Pt	900	2000	5000	5000	500	5000	5000	500	Kaplan 2007a [3]	Kaplan 2007a [3]	tetravalent, analog Zr
Pu	270	5900	10000	10000	10000	10000	10000	10000	Kaplan 2007a	Kaplan and Coates 2007, Kaplan et al. 2008a	Reducing cement assigned K_d from Kaplan et al. 2008a See note [4] for source of old oxidizing cement K_d values
Ra	5	17	100	100	70	0.5	3	20	Kaplan 2007a	Kaplan and Coates 2007	NA
Rn	1.0E-09	1.0E-09	1.0E-09	1.0E-09	1.0E-09	1.0E-09	1.0E-09	1.0E-09	Kaplan 2007a	Kaplan and Coates 2007	NA
Se	1000	1000	300	300	150	300	300	150	Kaplan 2007a	Kaplan 2007a, Kaplan and Coates 2007	NA
Sr	5	17	3	30	15	0.5	3	20	Kaplan 2007a	Kaplan and Coates 2007	NA
Tc	0.6	1.8	0.8	0.8	0.5	5000	5000	5000	Kaplan et al. 2008b and Kaplan 2009	Kaplan and Coates 2007	NA
Th	900	2000	5000	5000	500	5000	5000	500	Kaplan 2007a	Kaplan 2007a, Kaplan and Coates 2007	NA

Table C-12. Best Value K_d's Recommended for Use in Composite Analysis (McDowell-Boyer and Kaplan 2009)[5]

	Soil K_d (mL/g) [1]		Oxidizing Cement K_d (mL/g) [1]			Reducing Cement K_d (mL/g) [1]			K_d Reference		
Element	**Sandy**	**Clayey**	**Young**	**Middle**	**Old**	**Young**	**Middle**	**Old**	**Sand/Clay**	**Concrete**	**Notes [2]**
Tl	60	150	1	1	0.1	1	1	0.1	Kaplan 2007b [3]	Kaplan and Coates 2007 [3]	monovalent, analog Ag
U	200	300	250	250	70	2500	2500	2500	Kaplan 2007a	Kaplan and Coates 2007	NA
Y	1100	8500	6000	6000	600	5000	5000	1000	Kaplan 2007a [3]	Kaplan and Coates 2007 [3]	trivalent transition metal, analog Am
Zr	900	2000	5000	5000	500	5000	5000	500	Kaplan 2007a	Kaplan 2007a, Kaplan and Coates 2007	NA

Notes for Table C-12:

(1) Zero values represented as 1E-09 for computational purposes (to avoid divide by zero errors.

(2) Notes regarding analogs represent professional judgment.

(3) Value for analog taken from cited reference.

(4) Kaplan and Coates 2007 (Table 4) recommended best estimate Pu K_ds for stage 1 and 2 oxidizing concrete of 10,000 mL/g, while recommending a value of 1,000 for stage 3 oxidizing concrete. Kaplan and Coates 2007 (Table 4) also recommended best estimate Pu K_ds for stage 1 and 2 reducing concrete of 4,000 mL/g and that of 500 mL/g for stage 3 reducing concrete. However Kaplan and Coates 2007 (Table 3) documented an average measured Pu K_d for stage 3 oxidizing concrete of 92,200 mL/g. Based upon additional K_d measurements of reducing concrete, Kaplan et al. 2008a (Tables 4 and 5) recommended a best estimate Pu K_ds for stage 1, 2, and 3 reducing concrete of 10,000 mL/g. As noted by Kaplan 2007a, reducing concrete does not have a greater Pu K_d than oxidizing concrete. Since the measured Pu K_d for stage 3 oxidizing concrete averaged 92,200 mL/g (Kaplan and Coates 2007), the recommended Pu K_d for stage 3 reducing concrete is 10,000 mL/g (Kaplan et al. 2008a), and reducing concrete should not have a greater Pu K_d than oxidizing concrete (Kaplan 2007a), the Pu K_d for stage 3 oxidizing concrete will be taken as 10,000 mL/g.

(5) K_d distributions were calculated from the above "best value" K_ds. Following McDowell-Boyer and Kaplan (2009), it was assumed that K_d values have log-normal distributions and that for sandy soil and cement the standard deviation is 75% of the mean value ("best value" K_ds above) and for clayey soil the standard deviation is 50% of the mean value (also "best value" K_ds above).

Table C-13 Cellulose Degradation Product Correction Factors for Composite Analysis (McDowell-Boyer and Kaplan 2009)

Table C-13. Cellulose Degradation Product Correction Factors for Composite Analysis (McDowell-Boyer and Kaplan 2009)				
Element	**Correction Factor**	**Reference**	**Analog**	**Comments (1)**
Ac	0.049	Kaplan 2007a (2)	Ce & Eu	NA
Ag	1.41	Kaplan 2007a (2)	Ni	Soft monovalent
Al	0.049	Kaplan 2007a (2)	Ce & Eu	Hard trivalent
Am	0.049	Kaplan 2007a (2)	Ce & Eu	NA
At	0.5	Kaplan 2007a (2)	None	NA
Ba	1.89	Kaplan 2007a (2)	Sr	NA
Bi	0.049	Kaplan 2007a (2)	Ce & Eu	trivalent cation
C	0.5	Kaplan 2007a (2)	None	NA
Ca	1.89	Kaplan 2007a (2)	Sr	divalent cation
Cd	1.89	Kaplan 2007a (2)	Sr	divalent cation
Cf	0.049	Kaplan 2007a (2)	Ce & Eu	NA
Cl	1	Kaplan and Serkiz 2006 (3)	Iodide	NA
Cm	0.049	Kaplan 2007a (2)	Ce & Eu	NA
Cs	1.66	Kaplan 2007a (2)	Cs	NA
Fr	1.66	Kaplan 2007a (2)	Cs	NA
H	1	Kaplan 2007a (2)	N/A	NA
I	1	Kaplan and Serkiz 2006 (3)	None	NA
K	1.66	Kaplan 2007a (2)	Cs	monovalent cation
Lu	0.049	Kaplan 2007a (2)	Ce & Eu	trivalent lanthanide
Mo	1	Kaplan and Serkiz 2006 (3)	Selenate	oxyanion
Nb	1	Kaplan and Serkiz 2006 (3)	Selenate	oxyanion
Ni	1.41	Kaplan 2007a (2)	Ni	NA
Np	1.66	Kaplan 2007a (2)	Cs	NA
Pa	1.66	Kaplan 2007a (2)	Cs	NA
Pb	1.41	Kaplan 2007a (2)	Ni	NA
Pd	1.41	Kaplan 2007a (2)	Ni	No analog for soft divalent, Ni is closest
Po	1.41	Kaplan 2007a (2)	Ni	NA
Pt	1.41	Kaplan 2007a (2)	Ni	No analog for soft divalent, Ni is closest
Pu	0.51	Kaplan 2007a (2)	Th	NA
Ra	1.89	Kaplan 2007a (2)	Sr	NA
Rn	1	Kaplan 2007a (2)	N/A	NA
Se (sand)	0.2	Kaplan and Serkiz 2006 (3)	Selenate	NA
Se (clay)	1	Kaplan and Serkiz 2006 (3)	Selenate	NA
Sr	1.89	Kaplan 2007a (2)	Sr	NA

Table C-13. Cellulose Degradation Product Correction Factors for Composite Analysis (McDowell-Boyer and Kaplan 2009)

Element	Correction Factor	Reference	Analog	Comments [1]
Tc	1	Kaplan and Serkiz 2006 [3]	None	NA
Th	0.51	Kaplan 2007a [2]	Th	NA
Tl	1.66	Kaplan 2007a [2]	Cs	monovalent cation except under strongly reducing conditions
U	1.89	Kaplan 2007a [2]	Sr	NA
Y	0.049	Kaplan 2007a [2]	Ce & Eu	trivalent cation
Zr	0.08	Kaplan 2007a [2]	Zr	NA

Notes for Table C-13:

[1] Comments provide basis for selection of analog for elements not listed in Kaplan (2007a) or Kaplan and Serkiz (2006) based on professional judgment.

[2] Correction factor for element or analog given in Table 15 of Kaplan (2007a).

[3] Correction factor for element or analog given in Table 7 of Kaplan and Serkiz (2006).

Table C-14. Element-specific Transfer Factors (52 CA Parents & Daughters with Half-lives >3 Years) (Phifer et al. 2009)

Table C-14. Element-specific Transfer Factors (52 CA Parents & Daughters with Half-lives >3 Years) (Phifer et al. 2009)				
GoldSim™ Element ID	**Plant Soil Ratios**	**Beef TFs by Dose Species**	**Milk TFs by Dose Species**	**Fish Water Ratios**
Radionuclide	**Value[a] (kg/kg fresh wt)**	**Value[a] (d/kg)**	**Value[a] (d/L)**	**Value[a] (Bq/kg fish tissue per Bq/L)**
Ac227	6.83E-05	4.00E-04	2.00E-05	2.50E+01
Ag108m	1.18E-02	3.00E-03	1.58E-03	5.00E+00
Al26	1.27E-04	1.50E-03	2.06E-04	5.00E+02
Am241	6.83E-05	4.00E-05	1.50E-06	3.00E+01
Am242m	6.83E-05	4.00E-05	1.50E-06	3.00E+01
Am243	6.83E-05	4.00E-05	1.50E-06	3.00E+01
Bi210m	9.75E-02	4.00E-04	5.00E-04	1.50E+01
C14	1.37E-01	3.10E-02	1.20E-02	3.00E+00
Ca41	6.83E-02	2.00E-03	3.00E-03	4.00E+01 [b]
Cf249	6.83E-05	4.00E-05	1.50E-06	2.50E+01
Cl36	1.37E+01	2.00E-02	1.70E-02	5.00E+01
Cm244	8.39E-05	4.00E-05	2.00E-05	3.00E+01
Cm245	8.39E-05	4.00E-05	2.00E-05	3.00E+01
Cm246	8.39E-05	4.00E-05	2.00E-05	3.00E+01
Cm247	8.39E-05	4.00E-05	2.00E-05	3.00E+01
Cm248	8.39E-05	4.00E-05	2.00E-05	3.00E+01
Cs135	9.00E-01	5.00E-02	7.90E-03	3.00E+03
Cs137	9.00E-01	5.00E-02	7.90E-03	3.00E+03
H3	4.80E+00	1.20E-02	1.50E-02	1.00E+00
I129	7.80E-03	4.00E-02	9.00E-03	4.00E+01
K40	1.07E-01	2.00E-02	7.20E-03	1.00E+03
Lu174	7.80E-04	4.50E-03	2.06E-05	2.50E+01
Mo93	1.56E-01	1.00E-03	1.70E-03	1.00E+01
Nb93m	4.88E-03	2.90E-04	3.20E-05	3.00E+02
Nb94	4.88E-03	2.90E-04	3.20E-05	3.00E+02
Ni59	1.17E-02	5.00E-03	1.60E-02	1.00E+02
Ni63	1.17E-02	5.00E-03	1.60E-02	1.00E+02
Np237	2.54E-03	1.00E-03	5.00E-06	2.10E+01
Pa231	4.18E-04	4.47E-04	5.00E-06	1.00E+01
Pb210	1.17E-03	4.00E-04	2.60E-04	3.00E+02
Pd107	7.80E-03	4.00E-03	1.00E-02	1.00E+01
Pt193	4.88E-03	4.00E-03	5.15E-03	3.50E+01
Pu238	2.15E-04	1.00E-05	1.10E-06	3.00E+01
Pu239	2.15E-04	1.00E-05	1.10E-06	3.00E+01
Pu240	2.15E-04	1.00E-05	1.10E-06	3.00E+01
Pu241	2.15E-04	1.00E-05	1.10E-06	3.00E+01

Table C-14. Element-specific Transfer Factors (52 CA Parents & Daughters with Half-lives >3 Years) (Phifer et al. 2009)

GoldSim™ Element ID	Plant Soil Ratios	Beef TFs by Dose Species	Milk TFs by Dose Species	Fish Water Ratios
Radionuclide	**Value[a] (kg/kg fresh wt)**	**Value[a] (d/kg)**	**Value[a] (d/L)**	**Value[a] (Bq/kg fish tissue per Bq/L)**
Pu242	2.15E-04	1.00E-05	1.10E-06	3.00E+01
Pu244	2.15E-04	1.00E-05	1.10E-06	3.00E+01
Ra226	4.64E-03	9.00E-04	1.30E-03	5.00E+01
Ra228	4.64E-03	9.00E-04	1.30E-03	5.00E+01
Se79	5.14E-02	1.50E-02	4.00E-03	1.70E+02
Sr90	9.75E-02	8.00E-03	2.80E-03	6.00E+01
Tc99	4.68E-02	6.32E-03	1.87E-03	2.00E+01
Th229	6.44E-05	4.00E-05	5.00E-06	1.00E+02
Th230	6.44E-05	4.00E-05	5.00E-06	1.00E+02
Th232	6.44E-05	4.00E-05	5.00E-06	1.00E+02
U233	2.34E-03	3.00E-04	4.00E-04	1.00E+01
U234	2.34E-03	3.00E-04	4.00E-04	1.00E+01
U235	2.34E-03	3.00E-04	4.00E-04	1.00E+01
U236	2.34E-03	3.00E-04	4.00E-04	1.00E+01
U238	2.34E-03	3.00E-04	4.00E-04	1.00E+01
Zr93	1.95E-04	1.84E-04	5.50E-07	3.00E+02

Notes for Table C-14:

[a] Source: Lee and Coffield 2008.

[b] The bioaccumulation factor for Ca was changed from 400 L/kg to 40 L/Kg based on a review of the data that revealed an error.

Bq Becquerel
d day
kg kilogram
L liter
wt weight

Table C-15. Element-Specific Transfer Factors (50 Additional CA Daughters with Half-lives <3 Years) (Phifer et al. 2009)

Table C-15. Element-Specific Transfer Factors (50 Additional CA Daughters with Half-lives <3 Years) (Phifer et al. 2009)				
GoldSim™ Element ID	**Plant Soil Ratios**	**Beef TFs by Dose Species**	**Milk TFs by Dose Species**	**Fish Water Ratios**
Radionuclide	**Value[a] (kg/kg fresh wt)**	**Value[a] (d/kg)**	**Value[a] (d/L)**	**Value[a] (Bq/kg fish tissue per Bq/L)**
Ac225	6.83E-05	4.00E-04	2.00E-05	2.50E+01
Ac228	6.83E-05	4.00E-04	2.00E-05	2.50E+01
Ag108	1.18E-02	3.00E-03	1.58E-03	5.00E+00
Am242	6.83E-05	4.00E-05	1.50E-06	3.00E+01
At217	2.93E-02	1.00E-02	1.03E-02	1.50E+01
At218	2.93E-02	1.00E-02	1.03E-02	1.50E+01
Ba137m	2.93E-03	2.00E-04	4.80E-04	4.00E+00
Bi210	9.75E-02	4.00E-04	5.00E-04	1.50E+01
Bi211	9.75E-02	4.00E-04	5.00E-04	1.50E+01
Bi212	9.75E-02	4.00E-04	5.00E-04	1.50E+01
Bi213	9.75E-02	4.00E-04	5.00E-04	1.50E+01
Bi214	9.75E-02	4.00E-04	5.00E-04	1.50E+01
Cm242	8.39E-05	4.00E-05	2.00E-05	3.00E+01
Fr221	5.85E-03	2.50E-03	2.06E-02	3.00E+01
Fr223	5.85E-03	2.50E-03	2.06E-02	3.00E+01
Np238	2.54E-03	1.00E-03	5.00E-06	2.10E+01
Np239	2.54E-03	1.00E-03	5.00E-06	2.10E+01
Np240m	2.54E-03	1.00E-03	5.00E-06	2.10E+01
Pa233	4.18E-04	4.47E-04	5.00E-06	1.00E+01
Pa234m	4.18E-04	4.47E-04	5.00E-06	1.00E+01
Pb209	1.17E-03	4.00E-04	2.60E-04	3.00E+02
Pb211	1.17E-03	4.00E-04	2.60E-04	3.00E+02
Pb212	1.17E-03	4.00E-04	2.60E-04	3.00E+02
Pb214	1.17E-03	4.00E-04	2.60E-04	3.00E+02
Po210	1.37E-03	5.00E-03	3.40E-04	5.00E+01
Po211	1.37E-03	5.00E-03	3.40E-04	5.00E+01
Po212	1.37E-03	5.00E-03	3.40E-04	5.00E+01
Po213	1.37E-03	5.00E-03	3.40E-04	5.00E+01
Po214	1.37E-03	5.00E-03	3.40E-04	5.00E+01
Po215	1.37E-03	5.00E-03	3.40E-04	5.00E+01
Po216	1.37E-03	5.00E-03	3.40E-04	5.00E+01
Po218	1.37E-03	5.00E-03	3.40E-04	5.00E+01
Pu243	2.15E-04	1.00E-05	1.10E-06	3.00E+01
Ra223	4.64E-03	9.00E-04	1.30E-03	5.00E+01
Ra224	4.64E-03	9.00E-04	1.30E-03	5.00E+01
Ra225	4.64E-03	9.00E-04	1.30E-03	5.00E+01
Rn219	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C-15. Element-Specific Transfer Factors (50 Additional CA Daughters with Half-lives <3 Years) (Phifer et al. 2009)

GoldSim™ Element ID	Plant Soil Ratios	Beef TFs by Dose Species	Milk TFs by Dose Species	Fish Water Ratios
Radionuclide	**Value[a] (kg/kg fresh wt)**	**Value[a] (d/kg)**	**Value[a] (d/L)**	**Value[a] (Bq/kg fish tissue per Bq/L)**
Rn220	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rn222	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Th227	6.44E-05	4.00E-05	5.00E-06	1.00E+02
Th228	6.44E-05	4.00E-05	5.00E-06	1.00E+02
Th231	6.44E-05	4.00E-05	5.00E-06	1.00E+02
Th234	6.44E-05	4.00E-05	5.00E-06	1.00E+02
Tl206	7.80E-05	4.00E-02	2.00E-03	1.00E+04
Tl207	7.80E-05	4.00E-02	2.00E-03	1.00E+04
Tl208	7.80E-05	4.00E-02	2.00E-03	1.00E+04
Tl209	7.80E-05	4.00E-02	2.00E-03	1.00E+04
U237	2.34E-03	3.00E-04	4.00E-04	1.00E+01
U240	2.34E-03	3.00E-04	4.00E-04	1.00E+01
Y90	1.95E-03	1.00E-03	2.00E-05	3.00E+01

Notes for Table C-15:
[a] Source: Lee and Coffield 2008.
Bq Becquerel
d day
kg kilogram
L liter
wt weight

Table C-16. Dose Conversion Factors (52 CA Parents & Daughters with Half-lives >3 Years) (Phifer et al. 2009)

Table C-16. Dose Conversion Factors (52 CA Parents & Daughters with Half-lives >3 Years) (Phifer et al. 2009)					
GoldSim™ Element ID	**\Dose_Module\DoseConversionFactors**				
Radionuclide	**Ingestion DCF[a] (Sv/Bq)**	**Inhalation DCF[a] (Sv/Bq)**	**Shore Shine DCF[b] (Sv-m^2/Bq-s)**	**Submersion DCF[b] (Sv-m^3/Bq-s)**	**External DCF[b] (15 cm soil) (Sv-m^3/Bq-s)**
Ac227	1.10E-06	5.50E-04	1.57E-19	1.30E-20	2.62E-21
Ag108m	2.30E-09	3.70E-08	1.60E-15	1.69E-16	4.61E-17
Al26	3.50E-09	2.00E-08	2.49E-15	2.94E-16	7.73E-17
Am241	2.00E-07	9.60E-05	2.75E-17	1.88E-18	2.34E-19
Am242m	1.90E-07	9.20E-05	3.02E-18	7.28E-20	9.00E-21
Am243	2.00E-07	9.60E-05	5.35E-17	4.94E-18	7.60E-19
Bi210m	1.50E-08	3.40E-06	2.50E-16	2.68E-17	6.93E-18
C14	5.80E-10	5.80E-09	1.61E-20	4.39E-22	7.20E-23
Ca41	1.90E-10	1.80E-10	0.00E+00	0.00E+00	0.00E+00
Cf249	3.50E-07	7.00E-05	3.28E-16	3.45E-17	9.19E-18
Cl36	9.30E-10	7.30E-09	6.73E-19	4.48E-20	1.22E-20
Cm244	1.20E-07	5.70E-05	8.78E-19	1.15E-20	6.74E-22
Cm245	2.10E-07	9.90E-05	8.70E-17	8.84E-18	1.80E-18
Cm246	2.10E-07	9.80E-05	7.85E-19	1.05E-20	6.22E-22
Cm247	1.90E-07	9.00E-05	3.10E-16	3.27E-17	8.80E-18
Cm248	7.70E-07	3.60E-04	6.00E-19	7.96E-21	4.70E-22
Cs135	2.00E-09	8.60E-09	3.33E-20	1.10E-21	2.05E-22
Cs137	1.30E-08	3.90E-08	2.85E-19	1.49E-20	3.94E-21
H3	1.80E-11[c]	4.50E-11[d]	0.00E+00	0.00E+00	0.00E+00
I129	1.10E-07	9.60E-08	2.58E-17	8.91E-19	6.93E-20
K40	6.20E-09	2.10E-09	1.46E-16	1.74E-17	4.57E-18
Lu174	2.70E-10	4.20E-09	1.20E-16	1.20E-17	2.65E-18
Mo93	3.10E-09	2.30E-09	5.34E-18	5.92E-20	3.16E-21

Table C-16. Dose Conversion Factors (52 CA Parents & Daughters with Half-lives >3 Years) (Phifer et al. 2009)

GoldSim™ Element ID	\Dose_Module\DoseConversionFactors				
Radionuclide	**Ingestion DCF[a] (Sv/Bq)**	**Inhalation DCF[a] (Sv/Bq)**	**Shore Shine DCF[b] (Sv-m^2/Bq-s)**	**Submersion DCF[b] (Sv-m^3/Bq-s)**	**External DCF[b] (15 cm soil) (Sv-m^3/Bq-s)**
Nb93m	1.10E-10	1.70E-10	2.12E-15	2.45E-16	6.48E-17
Nb94	1.70E-09	4.90E-08	1.53E-15	1.67E-16	4.53E-17
Ni59	6.30E-11	8.30E-10	0.00E+00	0.00E+00	0.00E+00
Ni63	1.50E-10	2.00E-09	0.00E+00	0.00E+00	0.00E+00
Np237	1.10E-07	5.00E-05	2.87E-17	2.32E-18	4.16E-19
Pa231	7.10E-07	1.40E-04	4.07E-17	3.78E-18	9.62E-19
Pb210	6.90E-07	5.60E-06	2.48E-18	1.31E-19	1.31E-20
Pd107	3.70E-11	5.90E-10	0.00E+00	0.00E+00	0.00E+00
Pt193	3.10E-11	2.10E-11	1.19E-19	9.23E-22	3.03E-23
Pu238	2.30E-07	1.10E-04	8.38E-19	1.14E-20	8.07E-22
Pu239	2.50E-07	1.20E-04	3.67E-19	9.60E-21	1.52E-21
Pu240	2.50E-07	1.20E-04	8.03E-19	1.11E-20	7.84E-22
Pu241	4.80E-09	2.30E-06	1.93E-21	1.62E-22	3.15E-23
Pu242	2.40E-07	1.10E-04	6.67E-19	9.35E-21	6.85E-22
Pu244	2.40E-07	1.10E-04	5.58E-19	6.96E-21	4.04E-22
Ra226	2.80E-07	9.50E-06	6.44E-18	6.95E-19	1.65E-19
Ra228	6.90E-07	1.60E-05	0.00E+00	0.00E+00	0.00E+00
Se79	2.90E-09	6.80E-09	2.07E-20	5.93E-22	9.96E-23
Sr90	2.80E-08	1.60E-07	2.84E-19	1.46E-20	3.72E-21
Tc99	6.40E-10	1.30E-08	7.80E-20	3.14E-21	6.70E-22
Th229	4.90E-07	2.40E-04	8.54E-17	8.56E-18	1.70E-18
Th230	2.10E-07	1.00E-04	7.50E-19	3.94E-20	6.39E-21
Th232	2.30E-07	1.10E-04	5.51E-19	1.99E-20	2.78E-21
U233	5.10E-08	9.60E-06	7.16E-19	3.64E-20	7.24E-21
U234	4.90E-08	9.40E-06	7.48E-19	1.75E-20	2.14E-21
U235	4.70E-08	8.50E-06	1.48E-16	1.59E-17	3.75E-18
U236	4.70E-08	8.70E-06	6.50E-19	1.16E-20	1.14E-21
U238	4.50E-08	8.00E-06	5.51E-19	7.95E-21	5.52E-22
Zr93	1.10E-09	2.50E-08	0.00E+00	0.00E+00	0.00E+00

Notes for Table C-16:

[a]	Source: ICRP 1995	DCF	Dose conversion factor	s	second
[b]	Source: EPA 1993	Bq	Becquerel	Sv	Sievert
[c]	For HTO form	cm	centimeter		
[d]	For clearance rate M	m	meter		

Table C-17. Dose Conversion Factors (50 Additional CA Daughters with Half-lives <3 Years) (Phifer et al. 2009)

Table C-17. Dose Conversion Factors (50 Additional CA Daughters with Half-lives <3 Years) (Phifer et al. 2009)					
GoldSim™ Element ID	**\Dose_Module\DoseConversionFactors**				
Radionuclide	**Ingestion DCF[a] (Sv/Bq)**	**Inhalation DCF[a] (Sv/Bq)**	**Shore Shine DCF[b] (Sv-m^2/Bq-s)**	**Submersion DCF[b] (Sv-m^3/Bq-s)**	**External DCF[b] (15 cm soil) (Sv-m^3/Bq-s)**
Ac225	2.40E-08	8.50E-06	1.58E-17	1.61E-18	3.34E-19
Ac228	4.30E-10	2.50E-08	9.28E-16	1.04E-16	2.76E-17
Ag108	0.00E+00[c]	0.00E+00[c]	1.99E-17	2.00E-18	5.51E-19
Am242	3.00E-10	2.00E-08	1.57E-17	1.38E-18	2.67E-19
At217	0.00E+00[c]	0.00E+00[c]	3.03E-19	3.22E-20	8.61E-21
At218	0.00E+00[c]	0.00E+00[c]	4.18E-18	2.75E-19	3.13E-20
Ba137m	0.00E+00[c]	0.00E+00[c]	5.86E-16	6.26E-17	1.71E-17
Bi210	1.30E-09	9.30E-08	1.05E-18	6.33E-20	1.86E-20
Bi211	0.00E+00[c]	0.00E+00[c]	4.58E-17	4.85E-18	1.28E-18
Bi212	2.60E-10	3.10E-08	1.79E-16	2.00E-17	5.36E-18
Bi213	2.00E-10	3.00E-08	1.32E-16	1.39E-17	3.75E-18
Bi214	1.10E-10	1.40E-08	1.41E-15	1.66E-16	4.36E-17
Cm242	1.20E-08	5.90E-06	9.56E-19	1.33E-20	9.07E-22
Fr221	0.00E+00[c]	0.00E+00[c]	2.98E-17	3.22E-18	7.90E-19
Fr223	2.40E-09	8.90E-10	5.65E-17	5.11E-18	1.01E-18
Np238	9.10E-10	3.50E-09	5.29E-16	5.89E-17	1.58E-17
Np239	8.00E-10	1.00E-09	1.63E-16	1.70E-17	3.90E-18
Np240m	0.00E+00[c]	0.00E+00[c]	3.27E-16	3.51E-17	9.50E-18
Pa233	8.70E-10	3.90E-09	1.95E-16	2.05E-17	5.16E-18
Pa234m	0.00E+00[c]	0.00E+00[c]	1.53E-17	1.52E-18	4.20E-19
Pb209	5.70E-11	6.10E-11	3.01E-19	1.57E-20	4.08E-21
Pb211	1.80E-10	1.20E-08	5.08E-17	5.41E-18	1.46E-18
Pb212	6.00E-09	1.90E-07	1.43E-16	1.52E-17	3.62E-18
Pb214	1.40E-10	1.50E-08	2.44E-16	2.59E-17	6.70E-18
Po210	1.20E-06	4.30E-06	8.29E-21	9.03E-22	2.45E-22
Po211	0.00E+00[c]	0.00E+00[c]	7.61E-18	8.27E-19	2.24E-19
Po212	0.00E+00[c]	0.00E+00[c]	0.00E+00	0.00E+00	0.00E+00
Po213	0.00E+00[c]	0.00E+00[c]	0.00E+00	0.00E+00	0.00E+00
Po214	0.00E+00[c]	0.00E+00[c]	8.13E-20	8.85E-21	2.40E-21
Po215	0.00E+00[c]	0.00E+00[c]	1.74E-19	1.84E-20	4.98E-21
Po216	0.00E+00[c]	0.00E+00[c]	1.65E-20	1.80E-21	4.87E-22
Po218	0.00E+00[c]	0.00E+00[c]	8.88E-21	9.71E-22	2.63E-22
Pu243	8.50E-11	8.60E-11	2.41E-17	2.31E-18	4.20E-19
Ra223	1.00E-07	8.70E-06	1.28E-16	1.35E-17	3.10E-18
Ra224	6.50E-08	3.40E-06	9.57E-18	1.03E-18	2.62E-19
Ra225	9.90E-08	7.70E-06	1.33E-17	6.49E-19	5.90E-20
Rn219	0.00E+00[c]	0.00E+00[c]	5.49E-17	5.85E-18	1.54E-18
Rn220	0.00E+00[c]	0.00E+00[c]	3.81E-19	4.03E-20	1.10E-20

Table C-17. Dose Conversion Factors (50 Additional CA Daughters with Half-lives <3 Years) (Phifer et al. 2009)

GoldSim™ Element ID	\Dose_Module\DoseConversionFactors				
Radionuclide	**Ingestion DCF[a] (Sv/Bq)**	**Inhalation DCF[a] (Sv/Bq)**	**Shore Shine DCF[b] (Sv-m^2/Bq-s)**	**Submersion DCF[b] (Sv-m^3/Bq-s)**	**External DCF[b] (15 cm soil) (Sv-m^3/Bq-s)**
Rn222	0.00E+00[c]	0.00E+00[c]	3.95E-19	4.16E-20	1.14E-20
Th227	8.80E-09	1.00E-05	1.04E-16	1.07E-17	2.65E-18
Th228	7.20E-08	4.00E-05	2.35E-18	2.05E-19	4.17E-20
Th231	3.40E-10	3.30E-10	1.85E-17	1.18E-18	1.94E-19
Th234	3.40E-09	7.70E-09	8.32E-18	7.64E-19	1.29E-19
Tl206	0.00E+00[c]	0.00E+00[c]	1.99E-18	1.30E-19	3.93E-20
Tl207	0.00E+00[c]	0.00E+00[c]	3.76E-18	3.38E-19	9.48E-20
Tl208	0.00E+00[c]	0.00E+00[c]	2.98E-15	3.84E-16	9.68E-17
Tl209	0.00E+00[c]	0.00E+00[c]	1.90E-15	2.22E-16	5.79E-17
U237	7.60E-10	1.90E-09	1.33E-16	1.33E-17	2.78E-18
U240	1.10E-09	5.80E-10	4.23E-18	9.11E-20	7.62E-21
Y90	2.70E-09	1.50E-09	5.32E-18	3.63E-19	1.20E-19

Notes: for Table C-17
[a] Source: ICRP 1995.
[b] Source: EPA 1993.
[c] Source: NCRP 1996.
DCF Dose conversion factor
cm centimeter
Bq Becquerel
m meter
s second
Sv Sievert

Table C-18. Dose Module Exposure Parameters and Consumption Rates (Phifer et al. 2009)

GoldSim™ Element ID	Equation symbol (from CA Section 9.3.2.2)	Units	Deterministic Value	Distribution			
				Most Likely Value	Min	Max	Distribution Type
Irrigation_rate	Irr	$L/d/m^2$	3.6	3.6	2.08	5.5	Triangular
retention_leaf	r_i	--	0.25	0.25	0.2	0.25	Triangular
Holdup_vegetable	t_hold_veg	days	6	8.7 [d]	0.0417 [d] (1 hr)	Large [b]	Normal (SD = 1)
Irrigation_time_garden[a]	t_{irr}	d	70	70	--	--	Normal (SD = 7)
Weathering_loss	λ_e	1/d	0.0495	0.0495	0.03	0.0495	Triangular
Vegetable_Yield	Y	kg/m^2	0.7	0.7	0.2	4	Triangular
Irrigation_time_pasture	t_{irr}	d	30 [d]	70 [d]	--	--	Normal (SD = 7)
Fodder_intake_beef	IR_{fodder}	kg/d	36	36	--	--	Normal [c] (SD = 8)
Fodder_intake_milk	IR_{fodder}	kg/d	52	52	--	--	Normal [c] (SD = 11)
Frac_Pasture_beef	$F_{pasture}$	--	0.75	0.75	0.5	1	Triangular [c]
Frac_Pasture_milk	$F_{pasture}$	--	0.56	0.56	0.5	1	Triangular [c]
Water_intake_beef	IR_water	L/d	28	28	28	50	Triangular [c]
Water_intake_milk	IR_water	L/d	50	50	50	60	Triangular [c]
Frac_Water_beef	F_{water}	--	1	1	--	--	Discrete
Frac_Water_milk	F_{water}	--	1	1	--	--	Discrete
holdup_beef	t_hold-beef	d	6	6	6	20	Triangular [d]
holdup_milk	t_hold_milk	d	3	3	1	4	Triangular [d]
Soil_Ing_Rate	IR_soil	kg/yr	0.0365	0.0365	0.0008	0.05	Triangular [c]
Inhalation_Rate	InhR	m^3/yr	5,548	5,548	1,267	11,600	Triangular [d]
Dust_Loading_Garden	dust_loading	kg/m^3	1.00E-07	1.00E-07	1.00E-08	3.00E-07	Triangular
Water_Ing_Rate	IR_water	L/yr	337	337	184	730	Triangular [c]
Garden_Time_frac	T_garden	--	0.01	0.01	0.01	0.08	Triangular [c]
Veg_Ing_Rate _nonleafy	IR_veg	kg/yr	163	163	90	276	Triangular [d]
Veg_Ing_Rate_leafy	IR_veg	kg/yr	21	21	18	43	Triangular [d]
Meat_Ing_Rate	IR_beef	kg/yr	43	43	26	81	Triangular [d]
Milk_Ing_Rate	IR_milk	L/yr	120	120	73.7	230	Triangular [d]
Wash_Factor _Veg_leafy	F_w	--	0.5	0.5	--	--	Discrete value
Wash_Factor_Veg _nonleafy	F_w	--	1 [e]	1 [e]	--	--	Discrete value
Frac_Local_Veg_ leafy	F_{veg}	--	0.173	0.173	0	0.5	Triangular
Frac_Local_Veg _nonleafy	F_{veg}	--	0.173	0.173	0	0.5	Triangular
Frac_Local_Meat	F_{beef}	--	0.306	0.306	0	0.5	Triangular
Frac_Local_Milk	F_{milk}	--	0.207	0.207	0	0.5	Triangular
Fish_Ing_Rate [f]	IR_fish	kg/yr	9	9	2.2	19	Triangular [c]
t_holdup_fish	t_hold_fish	d	0 [g]	0 [g]	--	--	Discrete
Shoreline_Time	T_shore	hr/yr	23	23	11	35	Triangular [c]
Swimming_Time	T_swimming	hr/yr	8.9	8.9	8.9	13	Triangular [c]

Table C-18. Dose Module Exposure Parameters and Consumption Rates (Phifer et al. 2009)-continued

GoldSim™ Element ID	Equation symbol (from CA Section 9.3.2.2)	Units	Deterministic Value	Distribution			
				Most Likely Value	Min	Max	Distribution Type
Boating_Time	T boating	hr/yr	21	21	9.1	31.5	Triangular [c]
Shore_External_Factor [f]	Fshape	--	0.3	0.3	--	--	Discrete
Swimming_Geometry [f]	Fgeo, swim	--	1	1	--	--	Discrete
Boating_Geometry [f]	Fgeo, boat	--	0.5	0.5	--	--	Discrete
Swimming_Absorption [f]	Fabsorption	mL/hr	35	35	--	--	Discrete
Porosity_SandySoil [h]	η	--	0.38	0.38	0.36	0.40	Normal (SD = 0.008)
Saturation_SandySoil [h]	S	--	0.683	0.683	0.661	0.705	Triangular
DryBulkDensity_Sandy Soil [h, i]	ρ or sed_dens	kg/m2	1,650	1,650	1590	1710	Normal (SD = 22)

Notes for Table C-18:
Source: Taylor et al. 2008, App. A, Tables 4-1 & 5-1 unless otherwise noted.

[a] Also referred to as Irrigation_time_veg.
[b] Large = 1.0E+40 days as selected by Neptune and Company, Inc.
[c] Distribution shape taken from WSRC 2008.
[d] Assumed by Neptune and Company, Inc.
[e] Conservatively assumed to be one (i.e., no effect of washing).
[f] Source: Jannik and Dixon 2006.
[g] Conservatively assumed to be zero (i.e., immediately consumed).
[h] Source: Phifer and Dixon 2009.
[i] DryBulkDensity_SandySoil used for all soil and sediment dry bulk densities.

d day
hr hour
kg kilogram
L liter
m meter
SD standard deviation
yr year

Table C-19. Water Flow and Concrete Aging Distribution (Smith et al. 2009b)

Parameter	Units	Distribution	Mean or Most Likely	Standard Deviation	Minimum (-3σ)	Maximum (+3σ)
Upper Three Runs Flow	cfs	Normal	237	1.35	232.94	241.06
Fourmile Branch Flow	cfs	Normal	32	0.41	30.78	33.22
Pen Branch / Steel Creek Flow	cfs	Normal	89	1.32	85.04	92.96
Lower Three Runs Flow	cfs	Normal	164	1.91	158.28	169.72
Savannah River Augusta Flow	cfs	Normal	9198	86.00	8940.00	9456.00
Savannah River U.S. 301 Bridge Flow	cfs	Normal	10175	102.63	9867.11	10482.89
Natural Infiltration Flow	in/yr	Normal	15	0.17	14.48	15.52
Clay Cap Infiltration	NA	Normal	1	0.01133	0.966	1.034
Geosynthetic Cap Infiltration	NA	Normal	1	0.01133	0.966	1.034
Concrete Cap Infiltration	NA	Normal	1	0.01133	0.966	1.034
Aquifer Sand Velocity Ratio [1]	NA	Log-normal	1	0.869	0.1	10
Aquifer Clay Velocity Ratio [1]	NA	Log-normal	1	1.31	0.1	10
Concrete 1st Stage exchanges	NA	Triangular	50	NA	1	100
Concrete 2nd Stage exchanges	NA	Triangular	500	NA	100	1000
Concrete 3rd Stage exchanges	NA	Triangular	7000	NA	1000	10000

[1] Velocity Ratio represents travel times normalized to the nominal (base case) value.

C.2 REFERENCES

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D. APPENDIX D: AQUIFER FLOW DATA AND FIGURES

D.1 AQUIFER FLOW PATH ONE-DIMENSIONAL DATA

Table D-1. List of Input Parameters for the 1D Aquifer GoldSim™ Model for those CA Units within the AM Flow Model Domain (Hamm et al. 2009, Appendix A)

CA Unit Name	POA	Model Length (ft)	Segment 1 Length (ft)	Segment 3 Length (ft)	Sand Pore Velocity (ft/yr)	Clay Pore Velocity (ft/yr)	Number of Clay Cells (-)
A_773	5	48441.72	37069.81	250.96	35.63	7.02	1
A_776_Complex	1	48535.50	37136.36	258.23	44.78	7.09	1
A_778_Complex	5	50147.21	38389.55	240.79	35.28	7.13	1
A_SeepBasins	1	44434.66	33931.50	323.71	33.24	5.73	1
M_SettlingBasin	1	20711.11	14985.51	1229.95	300.86	5.70	1
M_SewerLines	1	19877.16	14407.88	1146.93	276.20	4.89	1

Table D-2. List of Input Parameters for the 1D Aquifer GoldSim™ Model for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)

CA Unit Name	POA	Model Length (ft)	Segment 1 Length (ft)	Segment 3 Length (ft)	Sand Pore Velocity (ft/yr)	Clay Pore Velocity (ft/yr)	Number of Clay Cells (-)
C_Reactor_Vessel	3	7579.49	3360.02	2001.86	46.42	80.42	40
C_Reactor_Bld	3	7589.95	3450.87	1861.51	46.44	94.32	40
C_SeepBasins	3	6580.31	2979.18	1634.87	55.79	84.90	40
D_420	5	1684.47	842.23	336.89	235.24	0	0
D_420-2	5	1548.54	774.27	309.71	238.52	0	0
D_421	5	1890.80	945.40	378.16	224.40	0	0
D_421-2	5	2214.05	1107.02	442.81	217.52	0	0
D_772	5	1649.36	824.68	329.87	216.40	0	0
K_Reactor_Vessel	4	3811.91	1233.35	1727.55	90.79	86.48	40
K_Reactor_Bld	4	4182.44	1518.68	1615.87	85.94	84.57	40
K_SeepBasin	4	2966.98	1481.34	596.84	125.36	184.52	35
K_RetBasin	4	2443.99	1222.00	488.80	119.89	0.00	0
L_Reactor_Vessel	4	2512.20	1256.10	502.44	91.47	0.00	0
L_Reactor_Bld	4	2857.24	1430.02	569.20	89.61	24.39	5
L_SeepBasin	4	1656.10	828.05	331.22	107.81	0.00	0
N_Ford	1	30532.81	18339.76	4490.36	104.10	59.40	2
P_Reactor_Vessel	4	1938.02	969.01	387.60	77.30	0.00	0
P_Reactor_Bld	4	2105.82	1052.91	421.16	76.60	0.00	0
P_SeepBasins	4	2236.23	1118.11	447.25	104.10	0.00	0
T_Buildings	5	1395.75	697.88	279.15	295.31	0.00	0
T_BurialGround	5	1771.59	885.80	354.32	268.91	0.00	0
T_Delta	5	660.63	510.22	60.16	316.35	0.00	0
T_Gully	5	766.40	383.20	153.28	322.27	0.00	0
T_SeepBasin	5	1176.93	588.46	235.39	298.75	0.00	0

Table D-3. List of Input Parameters for the 1D Aquifer GoldSim™ Model for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Table D-3. List of Input Parameters for the 1D Aquifer GoldSim™ Model for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

CA Unit Name	POA	Model Length (ft)	Segment 1 Length (ft)	Segment 3 Length (ft)	Sand Pore Velocity (ft/yr)	Clay Pore Velocity (ft/yr)	Number of Clay Cells (-)
E_CIG	1	2444.38	1258.32	468.81	201.81	53.41	10
E_ET	1	2774.00	1376.76	571.20	166.21	31.41	9
E_ILV	1	1948.76	1007.62	366.80	232.01	18.06	3
E_LAWV	1	2541.79	1264.92	517.92	150.49	90.78	18
E_MWMF_FMB	3	6819.94	4788.36	595.08	58.94	5.72	1
E_MWMF_UTR	1	6609.33	3745.76	1065.60	43.00	13.79	1
E_NR_Pad1	1	1867.10	933.55	373.42	328.21	0.00	0
E_NR_Pad2	1	7028.06	5162.98	316.19	39.24	4.20	1
E_OBG	3	4682.59	2289.85	1087.52	132.07	14.79	1
E_ST_Center	1	2490.80	1237.08	511.47	198.92	49.83	9
E_ST_East	1	3259.62	1624.95	659.70	144.81	73.28	13
E_ST_West	1	2312.55	1652.40	164.43	292.03	17.34	1
E_ST_Plot3	1	1291.07	910.52	107.39	645.00	97.70	7
E_ST_Plot6a	1	1566.93	1278.48	115.38	497.56	0.00	0
E_ST_Plot6b	1	1525.87	1086.30	175.83	705.27	0.00	0
E_ST_Plot7	1	1458.86	1013.97	140.70	692.06	28.86	1
E_ST_Plot7a	1	1184.00	592.00	236.80	718.61	0.00	0
E_ST_Plot8	1	1953.60	976.80	390.72	744.84	0.00	0
E_ST_Plot8a	1	2260.77	1130.39	452.15	705.13	0.00	0
E_ST_Plot8b	1	1463.75	939.78	209.59	790.52	0.00	0
E_ST_Plot9	1	1795.39	900.58	354.46	617.81	89.39	7
E_ST_Plot9a	1	1722.73	947.38	292.18	609.94	71.74	3
E_ST_Plot9b	1	1535.44	853.66	272.02	545.22	90.92	10
E_TRU_Pad	1	7620.57	5771.83	117.20	38.10	4.24	1
F_AuxFacilities	1	4971.77	3373.52	586.19	48.78	13.32	1
F_Canyon	1	5365.75	3682.33	578.72	59.22	14.80	1
F_Central_Labs1	1	4311.47	2086.04	911.22	92.06	18.18	1
F_Central_Labs2	1	4553.98	2731.96	674.60	73.85	15.73	1
F_Equipment	1	6459.53	4594.41	486.80	22.19	6.55	1
F_FAMS	1	3142.73	1484.31	723.30	226.32	14.71	1
F_MOX	1	2063.84	1031.92	412.77	285.83	0.00	0
F_Old_SeepBasin	1	1860.35	1384.56	60.43	255.67	27.35	1
F_RetentBasin	1	3769.78	1891.85	742.82	126.94	13.47	2
F_SeepBasin	3	3114.75	1572.34	599.02	138.57	19.57	5
F_SewerLine_FMB	3	4261.19	2707.99	578.33	95.36	13.28	2
F_SewerLine_UTR	1	4777.47	2470.97	898.05	126.79	13.49	1
F_Tanks_Type1	1	5793.46	3978.28	621.69	13.06	9.72	1
F_Tanks_Type3	1	4838.95	2986.10	688.27	99.98	7.81	1
F_Tanks_Type3A_FMB	3	5815.17	4048.32	552.35	86.69	6.74	1
F_Tanks_Type3A_UTR	1	7056.20	5113.10	409.18	27.05	5.32	1
F_Tanks_Type4	1	5338.97	3793.31	407.66	9.77	6.74	1
F_Type1_Tank3_Spill	1	5599.87	3941.73	475.62	14.47	7.10	1
F_Type1_Tank8_Spill	1	6257.40	3957.29	875.48	24.13	16.40	1
H_Canyon	1	6087.92	3162.22	1123.24	96.52	21.95	1
H_Equipment_UTR	1	11673.76	8373.22	788.58	30.58	15.10	1
H_Equipment_FMB	3	4942.92	3050.05	703.35	93.75	41.02	1
H_RBOF	1	4555.03	2299.11	876.46	79.09	47.06	8
H_RetBasin_281_1_2H	1	9000.39	6243.41	883.95	35.76	25.74	1
H_RetBasin_281_8H	1	845.72	484.64	157.54	45.11	10.94	15
H_Sand_Filter	1	3200.99	1758.04	546.35	123.88	29.25	3
H_SeepBasins_904_44G	1	672.88	336.44	134.58	202.19	0.00	0
H_SewerLine_FMB	3	4138.31	2156.40	752.76	112.77	25.45	5
H_SewerLine_UTR	1	6071.44	3257.41	1055.04	101.05	25.22	2

Table D-3. List of Input Parameters for the 1D Aquifer GoldSim™ Model for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

CA Unit Name	POA	Model Length (ft)	Segment 1 Length (ft)	Segment 3 Length (ft)	Sand Pore Velocity (ft/yr)	Clay Pore Velocity (ft/yr)	Number of Clay Cells (-)
H_Spill_281_3H	1	1213.31	606.20	243.39	48.89	29.26	30
H_Tanks_Type1_East	1	5494.07	2648.63	1176.80	49.08	9.32	1
H_Tanks_Type1_West	1	9447.68	6664.22	784.19	21.42	11.76	1
H_Tanks_Type2	1	10829.78	7552.89	1011.02	35.76	19.25	1
H_Tanks_Type3_UTR_East	1	10661.83	7525.46	878.74	25.93	9.38	1
H_Tanks_Type3_UTR_West	1	17458.91	12788.23	834.21	33.40	8.64	1
H_Tanks_Type3_FMB	3	5508.24	2597.93	1273.62	66.44	13.06	1
H_Tanks_Type3AE	1	10265.14	7019.94	1139.23	38.73	12.36	1
H_Tanks_Type3AW_UTR	1	14007.80	10412.60	471.42	35.47	11.03	1
H_Tanks_Type3AW_FMB	3	5549.89	3897.35	483.33	75.25	9.86	1
H_Tanks_Type4_UTR	1	11206.38	7961.30	856.69	26.12	20.99	1
H_Tanks_Type4_FMB	3	5812.70	3700.72	779.73	63.19	25.25	1
H_Type1_Tank9_Spill	1	5400.76	2423.32	1378.04	49.19	9.25	1
H_Type2_Tank13_Spill	1	10695.43	7555.40	873.41	38.14	21.63	1
H_Type2_Tank16_Spill	1	10847.47	7661.52	887.49	37.10	15.64	1
H_Type3AW_Tank37_Spil_UTR	1	13998.47	10404.72	472.33	35.53	11.03	1
H_Type3AW_Tank37_Spil_FMB	3	6044.74	4386.66	342.09	62.04	8.65	1
H_Tritium_232_H	1	4012.72	2147.83	705.07	126.16	27.80	3
HANM	1	2114.22	1346.03	283.62	180.98	15.36	2
H_Tritium_234_7H	1	3213.01	1602.59	648.87	142.58	39.51	13
HAOM	1	3834.21	2169.86	622.82	128.09	30.54	3
H_Tritium_236_H	1	3665.96	2197.71	545.22	133.47	34.94	2
H_Tritium_237_238H	1	3676.11	1962.37	654.05	116.86	46.78	3
TEF	1	1754.43	877.22	350.89	158.40	0.00	0
H_Tritium_TEF_264_2H	1	2047.13	1574.69	0.03	152.90	0.37	1
H_Tritium_TPBAR	1	2047.13	1574.69	0.03	152.90	0.37	1
J_SWPF	1	1380.31	652.97	315.97	198.76	13.25	2
S_DWPF	1	4215.75	2009.98	939.49	102.08	12.02	1
S_PumpPit	1	2754.60	1329.91	586.85	155.33	25.22	1
S_Sandfilter_Long	1	6902.11	4981.44	426.25	26.44	12.67	1
S_Sandfilter_Short	1	4771.90	3336.70	434.20	98.80	9.55	1
Z_Vault1	1	3188.39	2164.46	374.59	109.30	17.56	1
Z_Vault4	1	2525.39	1180.61	601.00	174.88	17.03	1
Z_Vault2_North_Long	1	3643.93	2310.00	502.34	109.92	16.61	1
Z_Vault2_North_Short	1	3066.67	2302.06	73.99	108.32	5.50	1
Z_Vault2_South	1	2757.98	1748.33	380.25	181.26	23.83	1

Table D-4. List of Input Parameters for the 1D Aquifer GoldSim™ Model for those CA Units within the R Flow Model Domain (Hamm et al. 2009, Appendix D)

CA Unit Name	POA	Model Length (ft)	Segment 1 Length (ft)	Segment 3 Length (ft)	Sand Pore Velocity (ft/yr)	Clay Pore Velocity (ft/yr)	Number of Clay Cells (-)
R_Reactor_Vessel	2	3385.28	1758.37	624.65	87.10	110.67	7
R_Reactor_Bld_Long	2	5784.83	1327.65	3541.10	70.69	19.70	40
R_Reactor_Bld_Short	2	3800.89	1892.88	772.29	94.83	122.36	20
R_SeepBasin	2	3401.79	1700.89	680.36	128.19	0.00	0

Table D-5. List of Inventory Fractions for Segment 1 Cells for those CA Units within the AM Flow Model Domain (Hamm et al. 2009, Appendix A)

Table D-5. List of Inventory Fractions for Segment 1 Cells for those CA Units within the AM Flow Model Domain (Hamm et al. 2009, Appendix A)						
Segment 1 Cell Number	A_773	A_776_Complex	A_778_Complex	A_SeepBasins	M_SettlingBasin	M_SewerLines
1	0.4916141	1	1	1	0.378023	0.083817
2	0.5083859	0	0	0	0.543789	0.067486
3	0	0	0	0	0.078188	0.088632
4	0	0	0	0	0	0.089262
5	0	0	0	0	0	0.054984
6	0	0	0	0	0	0.080223
7	0	0	0	0	0	0.082692
8	0	0	0	0	0	0.113654
9	0	0	0	0	0	0.058757
10	0	0	0	0	0	0.027252
11	0	0	0	0	0	0.020608
12	0	0	0	0	0	0.031894
13	0	0	0	0	0	0.056256
14	0	0	0	0	0	0.022898
15	0	0	0	0	0	0.024756
16	0	0	0	0	0	0.028002
17	0	0	0	0	0	0.027037
18	0	0	0	0	0	0.025939
19	0	0	0	0	0	0.015853
20	0	0	0	0	0	0
21	0	0	0	0	0	0
22	0	0	0	0	0	0
23	0	0	0	0	0	0
24	0	0	0	0	0	0
25	0	0	0	0	0	0
26	0	0	0	0	0	0
27	0	0	0	0	0	0
28	0	0	0	0	0	0
29	0	0	0	0	0	0
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	0	0	0	0	0	0
33	0	0	0	0	0	0
34	0	0	0	0	0	0
35	0	0	0	0	0	0
36	0	0	0	0	0	0
37	0	0	0	0	0	0
38	0	0	0	0	0	0
39	0	0	0	0	0	0
40	0	0	0	0	0	0
41	0	0	0	0	0	0
42	0	0	0	0	0	0
43	0	0	0	0	0	0
44	0	0	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0

Table D-5. List of Inventory Fractions for Segment 1 Cells for those CA Units within the AM Flow Model Domain (Hamm et al. 2009, Appendix A)

Segment 1 Cell Number	A_773	A_776_Complex	A_778_Complex	A_SeepBasins	M_SettlingBasin	M_SewerLines
56	0	0	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-6. List of Inventory Fractions for Segment 1 Cells for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)

Table D-6. List of Inventory Fractions for Segment 1 Cells for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)						
Segment 1 Cell Number	C_Reactor_Vessel	C_Reactor_Bld	C_SeepBasins	D_420	D_420-2	D_421
1	0.955754	0.019007	0.274048	0.052829	0.358315	0.092576
2	0.044246	0.060493	0.244079	0.069819	0.370056	0.203974
3	0	0.076349	0.094204	0.070037	0.271629	0.205051
4	0	0.083132	0.09419	0.07805	0	0.205079
5	0	0.090825	0.093903	0.080152	0	0.18374
6	0	0.082431	0.093889	0.079957	0	0.084992
7	0	0.082456	0.093603	0.080231	0	0.024588
8	0	0.063721	0.012084	0.080037	0	0
9	0	0.024792	0	0.080076	0	0
10	0	0.037326	0	0.080351	0	0
11	0	0.073413	0	0.080156	0	0
12	0	0.085638	0	0.062134	0	0
13	0	0.065127	0	0.044937	0	0
14	0	0.050161	0	0.044958	0	0
15	0	0.047024	0	0.016276	0	0
16	0	0.041281	0	0	0	0
17	0	0.016824	0	0	0	0
18	0	0	0	0	0	0
19	0	0	0	0	0	0
20	0	0	0	0	0	0
21	0	0	0	0	0	0
22	0	0	0	0	0	0
23	0	0	0	0	0	0
24	0	0	0	0	0	0
25	0	0	0	0	0	0
26	0	0	0	0	0	0
27	0	0	0	0	0	0
28	0	0	0	0	0	0
29	0	0	0	0	0	0
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	0	0	0	0	0	0
33	0	0	0	0	0	0
34	0	0	0	0	0	0
35	0	0	0	0	0	0
36	0	0	0	0	0	0
37	0	0	0	0	0	0
38	0	0	0	0	0	0
39	0	0	0	0	0	0
40	0	0	0	0	0	0
41	0	0	0	0	0	0
42	0	0	0	0	0	0
43	0	0	0	0	0	0
44	0	0	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0

Table D-6. List of Inventory Fractions for Segment 1 Cells for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)

Segment 1 Cell Number	C_Reactor_Vessel	C_Reactor_Bld	C_SeepBasins	D_420	D_420-2	D_421
55	0	0	0	0	0	0
56	0	0	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-6. List of Inventory Fractions for Segment 1 Cells for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)

Segment 1 Cell Number	D_421-2	D_772	K_Reactor_Vessel	K_Reactor_Bld	K_SeepBasin	K_RetBasin
1	0.082645	0.016336	0.3408448	0.024036	0.034217	0.000428
2	0.100359	0.027561	0.4631084	0.025506	0.102775	0.001296
3	0.100067	0.027668	0.1960468	0.025439	0.145299	0.002163
4	0.100188	0.0275	0	0.026279	0.145914	0.003031
5	0.099896	0.027607	0	0.029376	0.145703	0.003898
6	0.100017	0.027439	0	0.035111	0.146018	0.004766
7	0.099932	0.027408	0	0.03625	0.145042	0.005633
8	0.09964	0.027515	0	0.039089	0.10094	0.006501
9	0.099761	0.027347	0	0.04343	0.034087	0.007368
10	0.099469	0.027453	0	0.043078	4.5E-06	0.008236
11	0.018025	0.027286	0	0.043384	0	0.009102
12	0	0.027256	0	0.043359	0	0.009939
13	0	0.028337	0	0.036279	0	0.010762
14	0	0.052992	0	0.026861	0	0.011572
15	0	0.080249	0	0.024853	0	0.012295
16	0	0.081598	0	0.02213	0	0.013004
17	0	0.081597	0	0.026552	0	0.013694
18	0	0.082004	0	0.034416	0	0.014334
19	0	0.081509	0	0.034436	0	0.014972
20	0	0.063634	0	0.034552	0	0.015602
21	0	0.033533	0	0.035888	0	0.016177
22	0	0.027041	0	0.035656	0	0.016741
23	0	0.027229	0	0.029762	0	0.017305
24	0	0.027147	0	0.026734	0	0.017629
25	0	0.014757	0	0.018196	0	0.018348
26	0	0	0	0.016017	0	0.018842
27	0	0	0	0.016151	0	0.019336
28	0	0	0	0.016165	0	0.019818
29	0	0	0	0.016055	0	0.020241
30	0	0	0	0.01808	0	0.020658
31	0	0	0	0.018347	0	0.021076
32	0	0	0	0.01628	0	0.021491
33	0	0	0	0.01617	0	0.021874
34	0	0	0	0.016176	0	0.022245
35	0	0	0	0.016055	0	0.022614
36	0	0	0	0.01618	0	0.02298
37	0	0	0	0.015204	0	0.023306
38	0	0	0	0.002469	0	0.023612
39	0	0	0	0	0	0.023917
40	0	0	0	0	0	0.024222
41	0	0	0	0	0	0.024488
42	0	0	0	0	0	0.024721
43	0	0	0	0	0	0.024956
44	0	0	0	0	0	0.025189
45	0	0	0	0	0	0.025021
46	0	0	0	0	0	0.024341
47	0	0	0	0	0	0.023657
48	0	0	0	0	0	0.022647
49	0	0	0	0	0	0.022173
50	0	0	0	0	0	0.021408
51	0	0	0	0	0	0.020638
52	0	0	0	0	0	0.019792
53	0	0	0	0	0	0.018922
54	0	0	0	0	0	0.018052
55	0	0	0	0	0	0.017123
56	0	0	0	0	0	0.016123
57	0	0	0	0	0	0.015098
58	0	0	0	0	0	0.01391

Table D-6. List of Inventory Fractions for Segment 1 Cells for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)

Segment 1 Cell Number	D_421-2	D_772	K_Reactor_Vessel	K_Reactor_Bld	K_SeepBasin	K_RetBasin
59	0	0	0	0	0	0.012588
60	0	0	0	0	0	0.010961
61	0	0	0	0	0	0.009105
62	0	0	0	0	0	0.006433
63	0	0	0	0	0	0.001628
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-6. List of Inventory Fractions for Segment 1 Cells for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)

Segment 1 Cell Number	L_Reactor _Vessel	L_Reactor _Bld	L_SeepBa sin	N_Ford	P_Reactor _Vessel	P_Reactor _Bld
1	0.350123	0.001592	0.005853	1	0.247326	0.001324
2	0.469591	0.004802	0.017612	0	0.368138	0.003993
3	0.180286	0.009449	0.025607	0	0.321798	0.006023
4	0	0.0136	0.02574	0	0.062738	0.006427
5	0	0.01697	0.025879	0	0	0.008598
6	0	0.020236	0.025899	0	0	0.011302
7	0	0.023283	0.026004	0	0	0.013948
8	0	0.027294	0.032998	0	0	0.016594
9	0	0.031019	0.044801	0	0	0.019081
10	0	0.033984	0.056476	0	0	0.023181
11	0	0.034736	0.067578	0	0	0.028076
12	0	0.028923	0.070087	0	0	0.034807
13	0	0.024506	0.070536	0	0	0.042182
14	0	0.023134	0.070659	0	0	0.048894
15	0	0.02319	0.070454	0	0	0.049693
16	0	0.023933	0.070904	0	0	0.048808
17	0	0.026908	0.071027	0	0	0.047862
18	0	0.02513	0.066126	0	0	0.046291
19	0	0.021721	0.054256	0	0	0.043612
20	0	0.020358	0.042855	0	0	0.041739
21	0	0.025461	0.031175	0	0	0.041361
22	0	0.028367	0.019494	0	0	0.0417
23	0	0.029486	0.007805	0	0	0.041023
24	0	0.028437	0.000176	0	0	0.039308
25	0	0.028765	0	0	0	0.03809
26	0	0.026322	0	0	0	0.035763
27	0	0.022379	0	0	0	0.033074
28	0	0.018881	0	0	0	0.030386
29	0	0.018955	0	0	0	0.027493
30	0	0.024578	0	0	0	0.02503
31	0	0.031039	0	0	0	0.022341
32	0	0.038025	0	0	0	0.019653
33	0	0.042614	0	0	0	0.016843
34	0	0.041111	0	0	0	0.014297
35	0	0.035291	0	0	0	0.011609
36	0	0.029034	0	0	0	0.00892
37	0	0.024762	0	0	0	0.006194
38	0	0.020184	0	0	0	0.003564
39	0	0.016824	0	0	0	0.000916
40	0	0.013408	0	0	0	0
41	0	0.010227	0	0	0	0
42	0	0.006914	0	0	0	0
43	0	0.003584	0	0	0	0
44	0	0.000588	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0
56	0	0	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0

Table D-6. List of Inventory Fractions for Segment 1 Cells for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)

Segment 1 Cell Number	L_Reactor _Vessel	L_Reactor _Bld	L_SeepBa sin	N_Ford	P_Reactor _Vessel	P_Reactor _Bld
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-6. List of Inventory Fractions for Segment 1 Cells for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)

Segment 1 Cell Number	P_SeepBasins	T_Buildings	T_BurialGround	T_Delta	T_Gully	T_SeepBasin
1	0.008995	0.004953	0.0041	0.000574	0.001502	0.011185
2	0.026401	0.007098	0.012258	0.001184	0.001802	0.033483
3	0.030939	0.009286	0.020539	0.001541	0.001885	0.056003
4	0.030914	0.01886	0.028615	0.001896	0.001896	0.077666
5	0.030604	0.025451	0.036978	0.0023	0.00198	0.08489
6	0.030578	0.035819	0.044972	0.004488	0.00199	0.084536
7	0.030269	0.046527	0.052405	0.00554	0.002076	0.084745
8	0.030242	0.047909	0.05404	0.006121	0.002084	0.084392
9	0.029934	0.047089	0.053769	0.006956	0.002172	0.078838
10	0.029905	0.04341	0.054039	0.007602	0.002178	0.066317
11	0.029599	0.043421	0.053768	0.007924	0.002239	0.065795
12	0.029569	0.04259	0.053768	0.008245	0.002189	0.066038
13	0.0294	0.035391	0.054038	0.008566	0.002223	0.065842
14	0.029097	0.032473	0.053767	0.008886	0.002174	0.064438
15	0.033839	0.027348	0.054037	0.009206	0.002208	0.046546
16	0.056587	0.030326	0.053766	0.009524	0.002158	0.024859
17	0.081698	0.031809	0.053766	0.009828	0.002192	0.004428
18	0.105835	0.030204	0.054036	0.010111	0.002143	0
19	0.113013	0.015796	0.053195	0.011675	0.002176	0
20	0.091151	0.000727	0.046729	0.012876	0.002127	0
21	0.066039	0	0.038142	0.013887	0.00216	0
22	0.040322	0	0.029939	0.014013	0.002112	0
23	0.014978	0.000313	0.021437	0.014095	0.002144	0
24	9.26E-05	0.008262	0.013105	0.014177	0.002096	0
25	0	0.019671	0.004776	0.014274	0.002142	0
26	0	0.03106	0.000019	0.0144	0.00213	0
27	0	0.037153	0	0.014591	0.0022	0
28	0	0.037453	0	0.014801	0.002187	0
29	0	0.037452	0	0.015013	0.002258	0
30	0	0.037451	0	0.015192	0.002244	0
31	0	0.03717	0	0.015395	0.002316	0
32	0	0.037448	0	0.015605	0.002301	0
33	0	0.037447	0	0.015816	0.00242	0
34	0	0.037446	0	0.016016	0.002525	0
35	0	0.032712	0	0.016182	0.002729	0
36	0	0.021578	0	0.016346	0.00283	0
37	0	0.010081	0	0.016523	0.003072	0
38	0	0.000816	0	0.016699	0.00389	0
39	0	0	0	0.016921	0.005463	0
40	0	0	0	0.017181	0.006811	0
41	0	0	0	0.017402	0.008191	0
42	0	0	0	0.017813	0.009472	0
43	0	0	0	0.01867	0.011303	0
44	0	0	0	0.019198	0.01247	0
45	0	0	0	0.019229	0.014072	0
46	0	0	0	0.018817	0.014679	0
47	0	0	0	0.017897	0.01569	0
48	0	0	0	0.017127	0.015978	0
49	0	0	0	0.016331	0.016747	0
50	0	0	0	0.015487	0.016883	0
51	0	0	0	0.014878	0.017669	0
52	0	0	0	0.014701	0.017788	0
53	0	0	0	0.014491	0.018592	0
54	0	0	0	0.014011	0.01865	0
55	0	0	0	0.013692	0.019303	0
56	0	0	0	0.01363	0.019155	0
57	0	0	0	0.013708	0.019699	0
58	0	0	0	0.014058	0.0195	0

Table D-6. List of Inventory Fractions for Segment 1 Cells for those CA Units within the CKLP Flow Model Domain (Hamm et al. 2009, Appendix B)

Segment 1 Cell Number	P_SeepBasins	T_Buildings	T_BurialGround	T_Delta	T_Gully	T_SeepBasin
59	0	0	0	0.014427	0.02005	0
60	0	0	0	0.014686	0.019845	0
61	0	0	0	0.014277	0.020402	0
62	0	0	0	0.013845	0.02019	0
63	0	0	0	0.013416	0.020754	0
64	0	0	0	0.013011	0.020535	0
65	0	0	0	0.012568	0.021105	0
66	0	0	0	0.011954	0.020875	0
67	0	0	0	0.010984	0.021237	0
68	0	0	0	0.009155	0.020668	0
69	0	0	0	0.008529	0.020894	0
70	0	0	0	0.008448	0.020331	0
71	0	0	0	0.008272	0.020551	0
72	0	0	0	0.007953	0.019995	0
73	0	0	0	0.007775	0.020208	0
74	0	0	0	0.007692	0.019656	0
75	0	0	0	0.007631	0.019745	0
76	0	0	0	0.007572	0.018987	0
77	0	0	0	0.0075	0.018958	0
78	0	0	0	0.007381	0.018213	0
79	0	0	0	0.007232	0.018169	0
80	0	0	0	0.007084	0.017439	0
81	0	0	0	0.006662	0.01738	0
82	0	0	0	0.005588	0.016665	0
83	0	0	0	0.005035	0.016576	0
84	0	0	0	0.004354	0.015699	0
85	0	0	0	0.003472	0.015349	0
86	0	0	0	0.002897	0.014222	0
87	0	0	0	0.00243	0.013601	0
88	0	0	0	0.00209	0.012498	0
89	0	0	0	0.00184	0.011917	0
90	0	0	0	0.001619	0.010886	0
91	0	0	0	0.001459	0.010145	0
92	0	0	0	0.001301	0.008783	0
93	0	0	0	0.001144	0.0071	0
94	0	0	0	0.00103	0.004432	0
95	0	0	0	0.000975	0.002581	0
96	0	0	0	0.000904	9.82E-05	0
97	0	0	0	0.000831	0	0
98	0	0	0	0.000757	0	0
99	0	0	0	0.000645	0	0
100	0	0	0	0.000271	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	E_CIG	E_ET	E_ILV	E_LAWV	E_MWMF_FMB	E_MWMF_UTR
1	0.002883	0.000915	0.017396	0.002349	0.03125	0.001585
2	0.008392	0.002767	0.052113	0.006998	0.0625	0.003962
3	0.01114	0.004618	0.086831	0.011788	0.0625	0.005547
4	0.016807	0.00647	0.121548	0.016341	0	0.007132
5	0.019375	0.008322	0.15459	0.020661	0.03125	0.008716
6	0.019376	0.010173	0.162523	0.021785	0	0.010301
7	0.019375	0.012025	0.150858	0.021565	0	0.011094
8	0.019376	0.013702	0.116266	0.021785	0	0.011094
9	0.01916	0.013998	0.080991	0.021564	0	0.010301
10	0.019376	0.013998	0.045712	0.021565	0	0.011094
11	0.019375	0.013998	0.011172	0.021785	0	0.011886
12	0.019376	0.013998	0	0.021565	0	0.011094
13	0.019375	0.013998	0	0.021565	0	0.011886
14	0.019376	0.013998	0	0.021784	0	0.011094
15	0.019376	0.014563	0	0.021565	0	0.011094
16	0.019376	0.016371	0	0.021785	0	0.012678
17	0.019376	0.018222	0	0.021565	0	0.014263
18	0.01916	0.020073	0	0.021565	0	0.015848
19	0.019376	0.021926	0	0.021785	0	0.017433
20	0.019376	0.023777	0	0.021565	0	0.020602
21	0.019376	0.025628	0	0.021565	0	0.022187
22	0.019376	0.027444	0	0.021785	0	0.024564
23	0.019376	0.02804	0	0.021565	0	0.025357
24	0.019376	0.028041	0	0.021786	0	0.026941
25	0.019376	0.028041	0	0.021565	0	0.026149
26	0.019376	0.02804	0	0.021565	0	0.027734
27	0.019161	0.028041	0	0.021785	0	0.026941
28	0.019376	0.028041	0	0.021566	0	0.028526
29	0.019376	0.02804	0	0.021786	0	0.027734
30	0.019376	0.028041	0	0.021565	0	0.027734
31	0.019376	0.028041	0	0.021565	0.03125	0.027734
32	0.019376	0.02804	0	0.021785	0	0.028526
33	0.019376	0.028042	0	0.021566	0.0625	0.026941
34	0.019376	0.028041	0	0.021566	0.09375	0.027734
35	0.019161	0.02804	0	0.021785	0.0625	0.026941
36	0.019376	0.027939	0	0.021565	0.03125	0.026941
37	0.019376	0.0265	0	0.021785	0.09375	0.027734
38	0.019376	0.024649	0	0.021567	0.09375	0.027734
39	0.019376	0.022797	0	0.021565	0	0.027734
40	0.019376	0.020945	0	0.021785	0.09375	0.026941
41	0.019376	0.019094	0	0.021565	0.09375	0.025357
42	0.019376	0.017243	0	0.021565	0	0.024564
43	0.019376	0.015391	0	0.021788	0.09375	0.022979
44	0.019161	0.014091	0	0.021565	0.0625	0.022187
45	0.019377	0.014043	0	0.021785	0	0.022187
46	0.019376	0.014043	0	0.021565	0	0.020602
47	0.019376	0.014043	0	0.02005	0	0.019017
48	0.019376	0.014043	0	0.015622	0	0.018225
49	0.019377	0.014043	0	0.01079	0	0.015056
50	0.019376	0.014029	0	0.006143	0	0.014263
51	0.019376	0.012897	0	0.001568	0	0.011094
52	0.019042	0.011045	0	0	0	0.008716
53	0.01436	0.009193	0	0	0	0.007924
54	0.010014	0.007342	0	0	0	0.003962

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	E_CIG	E_ET	E_ILV	E_LAWV	E_MWMF _FMB	E_MWMF _UTR
55	0.006643	0.005491	0	0	0	0.00317
56	0.001131	0.003639	0	0	0	0.001585
57	0	0.001788	0	0	0	0.001585
58	0	0.000201	0	0	0	0
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	F_AuxFacilities	F_Canyon	F_Central_Labs1	F_Central_Labs2	F_Equipment	F_FAMS
1	0.062517	0.006802	0.016875	0.022977	0.174422	0.017394
2	0.077644	0.008294	0.055286	0.052637	0.173952	0.052262
3	0.009038	0.015279	0.079477	0.05338	0.005425	0.089926
4	0	0.009671	0.064874	0.053084	0	0.134877
5	0	0.01151	0.061474	0.022645	0	0.138921
6	0.027881	0.034184	0.061436	1E-07	0.281254	0.138646
7	0.062124	0.053958	0.061177	0.000224	0.116359	0.139015
8	0.062072	0.053619	0.061361	0.027594	0.087638	0.121965
9	0.06231	0.04509	0.061324	0.068198	0	0.08761
10	0.030132	0.045387	0.061287	0.078608	0.052434	0.053255
11	0.022162	0.045243	0.061249	0.105866	0.108515	0.02182
12	0.058288	0.045562	0.061212	0.148086	0	0.004308
13	0.058424	0.045418	0.061174	0.159736	0	0
14	0.058087	0.04928	0.060916	0.13201	0	0
15	0.043996	0.046823	0.061099	0.069709	0	0
16	0.010244	0.052709	0.058494	0.005247	0	0
17	0.018381	0.05	0.037756	0	0	0
18	0.038487	0.045621	0.013491	0	0	0
19	0.036354	0.045573	0.000038	0	0	0
20	0.0582	0.053684	0	0	0	0
21	0.058384	0.060312	0	0	0	0
22	0.05477	0.045044	0	0	0	0
23	0.068148	0.045324	0	0	0	0
24	0.022358	0.045141	0	0	0	0
25	0	0.033233	0	0	0	0
26	0	0.007241	0	0	0	0
27	0	0	0	0	0	0
28	0	0	0	0	0	0
29	0	0	0	0	0	0
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	0	0	0	0	0	0
33	0	0	0	0	0	0
34	0	0	0	0	0	0
35	0	0	0	0	0	0
36	0	0	0	0	0	0
37	0	0	0	0	0	0
38	0	0	0	0	0	0
39	0	0	0	0	0	0
40	0	0	0	0	0	0
41	0	0	0	0	0	0
42	0	0	0	0	0	0
43	0	0	0	0	0	0
44	0	0	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0
56	0	0	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)						
Segment 1 Cell Number	F_AuxFacilities	F_Canyon	F_Central_Labs1	F_Central_Labs2	F_Equipment	F_FAMS
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	F_MOX	F_Old_SeepBasin	F_Retent Basin	F_SeepBasin	F_SewerLine_FMB	F_SewerLine_UTR
1	0.001339	0.003726	0.015722	0.017404	0.015152	0.006297
2	0.003996	0.011144	0.04724	0.045004	0.030303	0.010076
3	0.006774	0.018562	0.078758	0.061114	0.045455	0.008816
4	0.01144	0.024096	0.109899	0.064836	0.060606	0.008816
5	0.015592	0.030729	0.123793	0.064604	0.05303	0.010076
6	0.018418	0.038278	0.123816	0.064343	0.068182	0.008816
7	0.020933	0.046058	0.123839	0.064082	0.05303	0.007557
8	0.023489	0.053409	0.123565	0.06382	0.060606	0.008816
9	0.024186	0.060957	0.110908	0.063559	0.05303	0.008816
10	0.025283	0.068505	0.079271	0.063297	0.05303	0.008816
11	0.027929	0.076427	0.047411	0.063035	0.060606	0.008816
12	0.030852	0.082904	0.015778	0.062774	0.045455	0.007557
13	0.032661	0.079492	0	0.062512	0.060606	0.008816
14	0.0328	0.07233	0	0.062251	0.05303	0.007557
15	0.032787	0.065108	0	0.061989	0.05303	0.006297
16	0.032787	0.059751	0	0.060903	0.05303	0.005038
17	0.032787	0.052461	0	0.048196	0.05303	0.005038
18	0.033061	0.044681	0	0.006277	0.045455	0.003778
19	0.032787	0.037163	0	0	0.05303	0.005038
20	0.032787	0.029646	0	0	0.030303	0.005038
21	0.032787	0.022215	0	0	0	0.005038
22	0.032787	0.014576	0	0	0	0.005038
23	0.03306	0.007059	0	0	0	0.006297
24	0.032787	0.000724	0	0	0	0.005038
25	0.032787	0	0	0	0	0.005038
26	0.032787	0	0	0	0	0.003778
27	0.032787	0	0	0	0	0.006297
28	0.032787	0	0	0	0	0.007557
29	0.03306	0	0	0	0	0.015113
30	0.032083	0	0	0	0	0.04534
31	0.029524	0	0	0	0	0.080605
32	0.026868	0	0	0	0	0.098237
33	0.024501	0	0	0	0	0.103275
34	0.024112	0	0	0	0	0.088161
35	0.022707	0	0	0	0	0.073048
36	0.019851	0	0	0	0	0.055416
37	0.017193	0	0	0	0	0.040302
38	0.014536	0	0	0	0	0.041562
39	0.010935	0	0	0	0	0.042821
40	0.005991	0	0	0	0	0.039043
41	0.002912	0	0	0	0	0.036524
42	0.000472	0	0	0	0	0.028967
43	0	0	0	0	0	0.015113
44	0	0	0	0	0	0.002519
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0
56	0	0	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	F_MOX	F_Old_SepBasin	F_Retent Basin	F_SeepBasin	F_SewerLine_FMB	F_SewerLine_UTR
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	F_Tanks_ Type1	F_Tanks_ Type3	F_Tanks_ Type3A_FMB	F_Tanks_ Type3A_UTR	F_Tanks_ Type4	F_Type1_ Tank3_Spill
1	0.155927	0.156499	0.170347	0.138656	0.205431	0.53759
2	0.291237	0.219715	0.290221	0.257353	0.259582	0.46241
3	0.077172	0.123787	0.063092	0.089286	0.036112	0
4	0.259113	0.10975	0	0.011555	0.217987	0
5	0.209896	0.218728	0.022082	0.179622	0.254544	0
6	0.006654	0.168708	0.252366	0.255252	0.026345	0
7	0	0.002813	0.201893	0.068277	0	0
8	0	0	0	0	0	0
9	0	0	0	0	0	0
10	0	0	0	0	0	0
11	0	0	0	0	0	0
12	0	0	0	0	0	0
13	0	0	0	0	0	0
14	0	0	0	0	0	0
15	0	0	0	0	0	0
16	0	0	0	0	0	0
17	0	0	0	0	0	0
18	0	0	0	0	0	0
19	0	0	0	0	0	0
20	0	0	0	0	0	0
21	0	0	0	0	0	0
22	0	0	0	0	0	0
23	0	0	0	0	0	0
24	0	0	0	0	0	0
25	0	0	0	0	0	0
26	0	0	0	0	0	0
27	0	0	0	0	0	0
28	0	0	0	0	0	0
29	0	0	0	0	0	0
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	0	0	0	0	0	0
33	0	0	0	0	0	0
34	0	0	0	0	0	0
35	0	0	0	0	0	0
36	0	0	0	0	0	0
37	0	0	0	0	0	0
38	0	0	0	0	0	0
39	0	0	0	0	0	0
40	0	0	0	0	0	0
41	0	0	0	0	0	0
42	0	0	0	0	0	0
43	0	0	0	0	0	0
44	0	0	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0
56	0	0	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	F_Tanks_ Type1	F_Tanks_ Type3	F_Tanks_ Type3A_FMB	F_Tanks_ Type3A_UTR	F_Tanks_ Type4	F_Type1_ Tank3_Spill
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	F_Type1_Tank8_Spill	H_Canyon	H_Equipment_UTR	H_Equipment_FMB	H_RBOF	H_RetBasin_281_1_2H
1	0.536856	0.010344	0.143223	0.121739	0.034973	0.343089
2	0.463144	0.02676	0	0.136232	0.092503	0.539639
3	0	0.045189	0.005115	0.173913	0.099296	0.117272
4	0	0.042783	0.107417	0.186957	0.141913	0
5	0	0.047987	0	0.189855	0.150523	0
6	0	0.052103	0	0.102899	0.162544	0
7	0	0.045451	0	0	0.158915	0
8	0	0.039475	0.332481	0	0.114614	0
9	0	0.039287	0.002558	0.002899	0.04444	0
10	0	0.0391	0	0.023188	0.00028	0
11	0	0.038913	0.179028	0	0	0
12	0	0.045336	0	0	0	0
13	0	0.040821	0	0	0	0
14	0	0.043152	0	0.008696	0	0
15	0	0.043014	0	0.02029	0	0
16	0	0.040795	0	0.033333	0	0
17	0	0.040684	0.230179	0	0	0
18	0	0.040573	0	0	0	0
19	0	0.042532	0	0	0	0
20	0	0.042902	0	0	0	0
21	0	0.041168	0	0	0	0
22	0	0.042611	0	0	0	0
23	0	0.040548	0	0	0	0
24	0	0.037856	0	0	0	0
25	0	0.025819	0	0	0	0
26	0	0.004801	0	0	0	0
27	0	0	0	0	0	0
28	0	0	0	0	0	0
29	0	0	0	0	0	0
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	0	0	0	0	0	0
33	0	0	0	0	0	0
34	0	0	0	0	0	0
35	0	0	0	0	0	0
36	0	0	0	0	0	0
37	0	0	0	0	0	0
38	0	0	0	0	0	0
39	0	0	0	0	0	0
40	0	0	0	0	0	0
41	0	0	0	0	0	0
42	0	0	0	0	0	0
43	0	0	0	0	0	0
44	0	0	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0
56	0	0	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	F_Type1_Tank8_Spill	H_Canyon	H_Equipment_UTR	H_Equipment_FMB	H_RBOF	H_RetBasin_281_1_2H
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_RetBasin_281_8H	H_Sand_Filter	H_SeepBasins_904_44G	H_SewerLine_FMB	H_SewerLine_UTR	H_Spill_281_3H
1	0.002025	0.00908	0.002869	0.05	0.014925	0.001772
2	0.002897	0.02718	0.005123	0.05	0.00995	0.005341
3	0.003301	0.04546	0.006342	0.008333	0.012438	0.00891
4	0.003703	0.06344	0.007656	0.058333	0.037313	0.01248
5	0.004194	0.080657	0.008795	0	0.03607	0.016049
6	0.004518	0.084158	0.010014	0.058333	0.026119	0.019462
7	0.004921	0.084234	0.011369	0.041667	0.012438	0.023161
8	0.005324	0.084588	0.012466	0.016667	0.006219	0.026731
9	0.005846	0.084384	0.013685	0.066667	0.003731	0.0303
10	0.006139	0.084739	0.014904	0	0.002488	0.033869
11	0.006541	0.084535	0.016316	0.058333	0.004975	0.037152
12	0.006946	0.084197	0.017357	0.016667	0.004975	0.039487
13	0.007498	0.07211	0.018576	0.041667	0.006219	0.039507
14	0.00776	0.054037	0.020029	0.058333	0.002488	0.039515
15	0.008162	0.036321	0.021028	0	0.012438	0.039523
16	0.008566	0.018366	0.022246	0.066667	0.011194	0.039243
17	0.008969	0.002515	0.022818	0	0.007463	0.039539
18	0.009563	0	0.02312	0.033333	0.006219	0.039547
19	0.009784	0	0.022896	0.041667	0.004975	0.039555
20	0.010186	0	0.022935	0	0.007463	0.039564
21	0.010589	0	0.023239	0.033333	0.004975	0.039572
22	0.011104	0	0.023014	0.033333	0.006219	0.039291
23	0.011117	0	0.023053	0.008333	0.004975	0.039588
24	0.011344	0	0.023092	0.025	0.007463	0.039596
25	0.011571	0	0.023397	0.033333	0.008707	0.039604
26	0.012037	0	0.023171	0.008333	0.00995	0.03842
27	0.01203	0	0.02321	0.025	0.016169	0.034741
28	0.012259	0	0.023516	0.033333	0.012438	0.031518
29	0.012485	0	0.023288	0.008333	0.011194	0.028027
30	0.012714	0	0.023327	0.025	0.011194	0.024537
31	0.013118	0	0.023367	0.033333	0.019901	0.021046
32	0.012933	0	0.023675	0.016667	0.019901	0.01744
33	0.013004	0	0.023445	0.016667	0.011194	0.01409
34	0.013075	0	0.023484	0.025	0.007463	0.0106
35	0.01341	0	0.023794	0.008333	0.012438	0.007109
36	0.013218	0	0.023563	0	0.011194	0.003618
37	0.01329	0	0.023602	0	0.018657	0.000503
38	0.01336	0	0.023641	0	0.017413	0
39	0.013701	0	0.023952	0	0.033582	0
40	0.013503	0	0.023587	0	0.031095	0
41	0.013575	0	0.022648	0	0.042289	0
42	0.013646	0	0.0218	0	0.032338	0
43	0.013992	0	0.020456	0	0.039801	0
44	0.01379	0	0.019366	0	0.042289	0
45	0.013859	0	0.018276	0	0.027363	0
46	0.013932	0	0.017377	0	0.041045	0
47	0.014001	0	0.016083	0	0.026119	0
48	0.014356	0	0.014993	0	0.037313	0
49	0.014145	0	0.014057	0	0.042289	0
50	0.014216	0	0.012801	0	0.032338	0
51	0.014287	0	0.011711	0	0.042289	0
52	0.014646	0	0.010621	0	0.032338	0
53	0.014431	0	0.009634	0	0.04602	0
54	0.014501	0	0.008425	0	0.00995	0
55	0.01457	0	0.00661	0	0	0
56	0.014808	0	0.004268	0	0	0
57	0.014407	0	0.00183	0	0	0
58	0.014298	0	8.66E-05	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_RetBasin_281_8H	H_Sand_Filter	H_SeepBasins_904_44G	H_SewerLine_FMB	H_SewerLine_UTR	H_Spill_281_3H
59	0.014188	0	0	0	0	0
60	0.014079	0	0	0	0	0
61	0.014248	0	0	0	0	0
62	0.013858	0	0	0	0	0
63	0.013748	0	0	0	0	0
64	0.013639	0	0	0	0	0
65	0.013799	0	0	0	0	0
66	0.013418	0	0	0	0	0
67	0.013308	0	0	0	0	0
68	0.013062	0	0	0	0	0
69	0.012966	0	0	0	0	0
70	0.012361	0	0	0	0	0
71	0.012014	0	0	0	0	0
72	0.011666	0	0	0	0	0
73	0.011542	0	0	0	0	0
74	0.010965	0	0	0	0	0
75	0.010618	0	0	0	0	0
76	0.01027	0	0	0	0	0
77	0.009923	0	0	0	0	0
78	0.009763	0	0	0	0	0
79	0.009221	0	0	0	0	0
80	0.008874	0	0	0	0	0
81	0.008527	0	0	0	0	0
82	0.008339	0	0	0	0	0
83	0.007825	0	0	0	0	0
84	0.007478	0	0	0	0	0
85	0.00713	0	0	0	0	0
86	0.006915	0	0	0	0	0
87	0.006429	0	0	0	0	0
88	0.006081	0	0	0	0	0
89	0.005734	0	0	0	0	0
90	0.005387	0	0	0	0	0
91	0.005137	0	0	0	0	0
92	0.004685	0	0	0	0	0
93	0.004338	0	0	0	0	0
94	0.003991	0	0	0	0	0
95	0.003713	0	0	0	0	0
96	0.003279	0	0	0	0	0
97	0.002589	0	0	0	0	0
98	0.001734	0	0	0	0	0
99	0.000889	0	0	0	0	0
100	0.000111	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Tanks_ Type1_East	H_Tanks_ Type1_West	H_Tanks_ Type2	H_Tanks_ Type3_UTR_East	H_Tanks_ Type3_UTR_West	H_Tanks_ Type3_FMB
1	0.137856	0.470588	0.421213	0.845188	1	0.01532
2	0.210066	0.26087	0.238038	0.154812	0	0.002786
3	0.19256	0.268542	0.340749	0	0	0.054318
4	0.161926	0	0	0	0	0.107242
5	0.181619	0	0	0	0	0.110028
6	0.115974	0	0	0	0	0.116992
7	0	0	0	0	0	0.158774
8	0	0	0	0	0	0.211699
9	0	0	0	0	0	0.14624
10	0	0	0	0	0	0.076602
11	0	0	0	0	0	0
12	0	0	0	0	0	0
13	0	0	0	0	0	0
14	0	0	0	0	0	0
15	0	0	0	0	0	0
16	0	0	0	0	0	0
17	0	0	0	0	0	0
18	0	0	0	0	0	0
19	0	0	0	0	0	0
20	0	0	0	0	0	0
21	0	0	0	0	0	0
22	0	0	0	0	0	0
23	0	0	0	0	0	0
24	0	0	0	0	0	0
25	0	0	0	0	0	0
26	0	0	0	0	0	0
27	0	0	0	0	0	0
28	0	0	0	0	0	0
29	0	0	0	0	0	0
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	0	0	0	0	0	0
33	0	0	0	0	0	0
34	0	0	0	0	0	0
35	0	0	0	0	0	0
36	0	0	0	0	0	0
37	0	0	0	0	0	0
38	0	0	0	0	0	0
39	0	0	0	0	0	0
40	0	0	0	0	0	0
41	0	0	0	0	0	0
42	0	0	0	0	0	0
43	0	0	0	0	0	0
44	0	0	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0
56	0	0	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Tanks_ Type1_East	H_Tanks_ Type1_West	H_Tanks_ Type2	H_Tanks_ Type3_UTR_East	H_Tanks_ Type3_UTR_West	H_Tanks_ Type3_FMB
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Tanks_ Type3AE	H_Tanks_ Type3AW _UTR	H_Tanks_ Type3AW _FMB	H_Tanks_ Type4_UT R	H_Tanks_ Type4_FM B	H_Type1_ Tank9_Sp ill
1	0.09925	1	0.154211	0.564299	0.222738	0.239877
2	0.109788	0	0.192171	0.435701	0.329466	0.360683
3	0.122109	0	0.048636	0	0.301624	0.322504
4	0.068853	0	0.088968	0	0.146172	0.076937
5	0.031241	0	0.141163	0	0	0
6	0.118153	0	0.209964	0	0	0
7	0.217461	0	0.162515	0	0	0
8	0.159837	0	0.002373	0	0	0
9	0.073309	0	0	0	0	0
10	0	0	0	0	0	0
11	0	0	0	0	0	0
12	0	0	0	0	0	0
13	0	0	0	0	0	0
14	0	0	0	0	0	0
15	0	0	0	0	0	0
16	0	0	0	0	0	0
17	0	0	0	0	0	0
18	0	0	0	0	0	0
19	0	0	0	0	0	0
20	0	0	0	0	0	0
21	0	0	0	0	0	0
22	0	0	0	0	0	0
23	0	0	0	0	0	0
24	0	0	0	0	0	0
25	0	0	0	0	0	0
26	0	0	0	0	0	0
27	0	0	0	0	0	0
28	0	0	0	0	0	0
29	0	0	0	0	0	0
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	0	0	0	0	0	0
33	0	0	0	0	0	0
34	0	0	0	0	0	0
35	0	0	0	0	0	0
36	0	0	0	0	0	0
37	0	0	0	0	0	0
38	0	0	0	0	0	0
39	0	0	0	0	0	0
40	0	0	0	0	0	0
41	0	0	0	0	0	0
42	0	0	0	0	0	0
43	0	0	0	0	0	0
44	0	0	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0
56	0	0	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Tanks_ Type3AE	H_Tanks_ Type3AW _UTR	H_Tanks_ Type3AW _FMB	H_Tanks_ Type4_UT R	H_Tanks_ Type4_FM B	H_Type1_ Tank9_Sp ill
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Type2_Tank13_Spill	H_Type2_Tank16_Spill	H_Type3AW_Tank37_Spil_UTR	H_Type3AW_Tank37_Spil_FMB	H_Tritium_232_H	HANM
1	0.847717	0.85906	1	0.457195	0.013352	0.028361
2	0.152283	0.140941	0	0.502732	0.039768	0.057845
3	0	0	0	0.040073	0.061807	0.057833
4	0	0	0	0	0.090586	0.057811
5	0	0	0	0	0.128894	0.057789
6	0	0	0	0	0.145548	0.057767
7	0	0	0	0	0.143064	0.057745
8	0	0	0	0	0.10794	0.057724
9	0	0	0	0	0.066649	0.057702
10	0	0	0	0	0.058209	0.05768
11	0	0	0	0	0.043986	0.0581
12	0	0	0	0	0.031418	0.059052
13	0	0	0	0	0.028233	0.053843
14	0	0	0	0	0.028537	0.057395
15	0	0	0	0	0.012011	0.060766
16	0	0	0	0	0	0.058008
17	0	0	0	0	0	0.057791
18	0	0	0	0	0	0.043973
19	0	0	0	0	0	0.002815
20	0	0	0	0	0	0
21	0	0	0	0	0	0
22	0	0	0	0	0	0
23	0	0	0	0	0	0
24	0	0	0	0	0	0
25	0	0	0	0	0	0
26	0	0	0	0	0	0
27	0	0	0	0	0	0
28	0	0	0	0	0	0
29	0	0	0	0	0	0
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	0	0	0	0	0	0
33	0	0	0	0	0	0
34	0	0	0	0	0	0
35	0	0	0	0	0	0
36	0	0	0	0	0	0
37	0	0	0	0	0	0
38	0	0	0	0	0	0
39	0	0	0	0	0	0
40	0	0	0	0	0	0
41	0	0	0	0	0	0
42	0	0	0	0	0	0
43	0	0	0	0	0	0
44	0	0	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0
56	0	0	0	0	0	0
57	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Type2_Tank13_Spill	H_Type2_Tank16_Spill	H_Type3AW_Tank37_Spil_UTR	H_Type3AW_Tank37_Spil_FMB	H_Tritium_232_H	HANM
58	0	0	0	0	0	0
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Tritium _234_7H	HAOM	H_Tritium _236_H	H_Tritium _237_238 H	TEF	H_Tritium _TEF_264 _2H
1	0.110459	0.013656	0.244552	0.133002	0.057035	0.213015
2	0.219276	0.027145	0.321782	0.136799	0.071119	0.213661
3	0.219102	0.037196	0.402546	0.136929	0.071129	0.213188
4	0.218685	0.033364	0.03112	0.137302	0.071344	0.212251
5	0.197873	0.042402	0	0.13719	0.071149	0.135592
6	0.034605	0.058548	0	0.080737	0.071159	0.012292
7	0	0.061837	0	0.06894	0.071373	0
8	0	0.0625	0	0.09451	0.071179	0
9	0	0.064372	0	0.074591	0.071189	0
10	0	0.059283	0	0	0.071403	0
11	0	0.058918	0	0	0.071209	0
12	0	0.058591	0	0	0.071219	0
13	0	0.061284	0	0	0.071229	0
14	0	0.060741	0	0	0.071444	0
15	0	0.059055	0	0	0.01682	0
16	0	0.064227	0	0	0	0
17	0	0.061484	0	0	0	0
18	0	0.05689	0	0	0	0
19	0	0.03792	0	0	0	0
20	0	0.020589	0	0	0	0
21	0	0	0	0	0	0
22	0	0	0	0	0	0
23	0	0	0	0	0	0
24	0	0	0	0	0	0
25	0	0	0	0	0	0
26	0	0	0	0	0	0
27	0	0	0	0	0	0
28	0	0	0	0	0	0
29	0	0	0	0	0	0
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	0	0	0	0	0	0
33	0	0	0	0	0	0
34	0	0	0	0	0	0
35	0	0	0	0	0	0
36	0	0	0	0	0	0
37	0	0	0	0	0	0
38	0	0	0	0	0	0
39	0	0	0	0	0	0
40	0	0	0	0	0	0
41	0	0	0	0	0	0
42	0	0	0	0	0	0
43	0	0	0	0	0	0
44	0	0	0	0	0	0
45	0	0	0	0	0	0
46	0	0	0	0	0	0
47	0	0	0	0	0	0
48	0	0	0	0	0	0
49	0	0	0	0	0	0
50	0	0	0	0	0	0
51	0	0	0	0	0	0
52	0	0	0	0	0	0
53	0	0	0	0	0	0
54	0	0	0	0	0	0
55	0	0	0	0	0	0
56	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Tritium _234_7H	HAOM	H_Tritium _236_H	H_Tritium _237_238 H	TEF	H_Tritium _TEF_264 _2H
57	0	0	0	0	0	0
58	0	0	0	0	0	0
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Tritium _TPBAR	J_SWPF	S_DWPF	S_PumpPit	S_Sandfilter_Long	S_Sandfilter_Short
1	0.213015	0.002746	0.009314	0.025007	0.148743	0.100456
2	0.213661	0.008298	0.027448	0.075115	0.392974	0.240238
3	0.213188	0.013844	0.032935	0.124941	0.347665	0.269251
4	0.212251	0.016172	0.034094	0.174822	0.110618	0.25022
5	0.135592	0.016353	0.056622	0.168345	0	0.127654
6	0.012292	0.016352	0.076245	0.151831	0	0.012181
7	0	0.016351	0.094361	0.139811	0	0
8	0	0.016168	0.094061	0.092911	0	0
9	0	0.016349	0.091912	0.043551	0	0
10	0	0.016348	0.091807	0.003666	0	0
11	0	0.016347	0.091392	0	0	0
12	0	0.016165	0.091596	0	0	0
13	0	0.016346	0.082623	0	0	0
14	0	0.016345	0.05871	0	0	0
15	0	0.016344	0.03901	0	0	0
16	0	0.016161	0.020507	0	0	0
17	0	0.016342	0.007367	0	0	0
18	0	0.016341	0	0	0	0
19	0	0.01634	0	0	0	0
20	0	0.016158	0	0	0	0
21	0	0.016339	0	0	0	0
22	0	0.016338	0	0	0	0
23	0	0.016337	0	0	0	0
24	0	0.016155	0	0	0	0
25	0	0.016335	0	0	0	0
26	0	0.016334	0	0	0	0
27	0	0.016334	0	0	0	0
28	0	0.016151	0	0	0	0
29	0	0.016475	0	0	0	0
30	0	0.020545	0	0	0	0
31	0	0.023384	0	0	0	0
32	0	0.023121	0	0	0	0
33	0	0.023378	0	0	0	0
34	0	0.023375	0	0	0	0
35	0	0.023113	0	0	0	0
36	0	0.023369	0	0	0	0
37	0	0.023366	0	0	0	0
38	0	0.023363	0	0	0	0
39	0	0.023101	0	0	0	0
40	0	0.023357	0	0	0	0
41	0	0.023354	0	0	0	0
42	0	0.023351	0	0	0	0
43	0	0.023088	0	0	0	0
44	0	0.023345	0	0	0	0
45	0	0.023342	0	0	0	0
46	0	0.023339	0	0	0	0
47	0	0.022068	0	0	0	0
48	0	0.021385	0	0	0	0
49	0	0.021392	0	0	0	0
50	0	0.021399	0	0	0	0
51	0	0.021169	0	0	0	0
52	0	0.019726	0	0	0	0
53	0	0.014794	0	0	0	0
54	0	0.00978	0	0	0	0
55	0	0.00517	0	0	0	0
56	0	0.001162	0	0	0	0
57	0	0	0	0	0	0
58	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	H_Tritium _TPBAR	J_SWPF	S_DWPF	S_PumpPit	S_Sandfilter_Long	S_Sandfilter_Short
59	0	0	0	0	0	0
60	0	0	0	0	0	0
61	0	0	0	0	0	0
62	0	0	0	0	0	0
63	0	0	0	0	0	0
64	0	0	0	0	0	0
65	0	0	0	0	0	0
66	0	0	0	0	0	0
67	0	0	0	0	0	0
68	0	0	0	0	0	0
69	0	0	0	0	0	0
70	0	0	0	0	0	0
71	0	0	0	0	0	0
72	0	0	0	0	0	0
73	0	0	0	0	0	0
74	0	0	0	0	0	0
75	0	0	0	0	0	0
76	0	0	0	0	0	0
77	0	0	0	0	0	0
78	0	0	0	0	0	0
79	0	0	0	0	0	0
80	0	0	0	0	0	0
81	0	0	0	0	0	0
82	0	0	0	0	0	0
83	0	0	0	0	0	0
84	0	0	0	0	0	0
85	0	0	0	0	0	0
86	0	0	0	0	0	0
87	0	0	0	0	0	0
88	0	0	0	0	0	0
89	0	0	0	0	0	0
90	0	0	0	0	0	0
91	0	0	0	0	0	0
92	0	0	0	0	0	0
93	0	0	0	0	0	0
94	0	0	0	0	0	0
95	0	0	0	0	0	0
96	0	0	0	0	0	0
97	0	0	0	0	0	0
98	0	0	0	0	0	0
99	0	0	0	0	0	0
100	0	0	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	Z_Vault1	Z_Vault4	Z_Vault2_North_Long	Z_Vault2_North_Short	Z_Vault2_South	E_ST_Plot 3
1	0.024052	0.001577	0.003979	0.015686	0.00233	0.000309
2	0.072243	0.004753	0.006631	0.011765	0.004003	0.000913
3	0.120433	0.00793	0.007958	0.019608	0.004806	0.001518
4	0.168624	0.011106	0.01061	0.031373	0.005438	0.002123
5	0.207473	0.014283	0.009284	0.027451	0.008024	0.002728
6	0.174768	0.017459	0.01061	0.023529	0.009234	0.003332
7	0.125901	0.020635	0.009284	0.011765	0.009548	0.003937
8	0.077492	0.023812	0.011936	0.003922	0.008659	0.004542
9	0.028811	0.026988	0.014589	0.003922	0.006215	0.005147
10	0.000205	0.030165	0.013263	0.015686	0.005078	0.005751
11	0	0.033341	0.009284	0.035294	0.005438	0.006356
12	0	0.036517	0.011936	0.047059	0.009453	0.006961
13	0	0.039694	0.006631	0.043137	0.01203	0.007566
14	0	0.04287	0.006631	0.035294	0.014094	0.00817
15	0	0.046047	0.007958	0.023529	0.017231	0.008775
16	0	0.049124	0.006631	0.007843	0.023067	0.00938
17	0	0.049654	0.002653	0.003922	0.026909	0.009985
18	0	0.05	0.007958	0.003922	0.027721	0.01059
19	0	0.049988	0.01061	0	0.025435	0.011194
20	0	0.049976	0.013263	0	0.019655	0.011799
21	0	0.048545	0.01061	0	0.015581	0.012404
22	0	0.04536	0.013263	0	0.014449	0.013008
23	0	0.042169	0.013263	0.007843	0.013896	0.013613
24	0	0.038979	0.01061	0.031373	0.011315	0.014218
25	0	0.035789	0.01061	0.043137	0.010125	0.014823
26	0	0.032598	0.011936	0.047059	0.012527	0.015427
27	0	0.029408	0.013263	0.062745	0.015696	0.016032
28	0	0.026218	0.009284	0.054902	0.018279	0.016976
29	0	0.023027	0.014589	0.035294	0.018655	0.017253
30	0	0.019837	0.018568	0.023529	0.016064	0.017533
31	0	0.016647	0.014589	0.007843	0.012571	0.01754
32	0	0.013456	0.014589	0	0.009895	0.017548
33	0	0.010266	0.017241	0	0.00828	0.017555
34	0	0.007037	0.018568	0.007843	0.005043	0.017562
35	0	0.003908	0.013263	0.015686	0.001825	0.01757
36	0	0.000839	0.014589	0.019608	0	0.017577
37	0	0	0.018568	0.031373	0	0.017585
38	0	0	0.017241	0.054902	0.000181	0.017592
39	0	0	0.015915	0.054902	0.003166	0.017599
40	0	0	0.022546	0.047059	0.007156	0.017607
41	0	0	0.022546	0.035294	0.009178	0.017614
42	0	0	0.017241	0.035294	0.010856	0.017621
43	0	0	0.022546	0.015686	0.013641	0.017611
44	0	0	0.019894	0.003922	0.014594	0.017373
45	0	0	0.018568	0	0.01451	0.017065
46	0	0	0.014589	0	0.014488	0.016757
47	0	0	0.022546	0	0.012543	0.016449
48	0	0	0.025199	0	0.010127	0.016141
49	0	0	0.022546	0	0.009363	0.015834
50	0	0	0.023873	0	0.012009	0.015526
51	0	0	0.025199	0	0.013259	0.015218
52	0	0	0.023873	0	0.014095	0.01491
53	0	0	0.02122	0	0.01398	0.014603
54	0	0	0.019894	0	0.011923	0.014295
55	0	0	0.019894	0	0.011752	0.013987
56	0	0	0.013263	0	0.01595	0.01395
57	0	0	0.009284	0	0.018987	0.013365
58	0	0	0.017241	0	0.017104	0.013058

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	Z_Vault1	Z_Vault4	Z_Vault2_North_Long	Z_Vault2_North_Short	Z_Vault2_South	E_ST_Plot 3
59	0	0	0.013263	0	0.017672	0.01275
60	0	0	0.014589	0	0.023405	0.012442
61	0	0	0.014589	0	0.027801	0.012135
62	0	0	0.015915	0	0.02778	0.011827
63	0	0	0.011936	0	0.024943	0.011519
64	0	0	0.01061	0	0.019323	0.011211
65	0	0	0.013263	0	0.015087	0.010904
66	0	0	0.013263	0	0.016636	0.010596
67	0	0	0.01061	0	0.016298	0.010288
68	0	0	0.01061	0	0.01211	0.00998
69	0	0	0.009284	0	0.010738	0.009672
70	0	0	0.007958	0	0.015282	0.009365
71	0	0	0.007958	0	0.018565	0.009057
72	0	0	0.006631	0	0.018799	0.008749
73	0	0	0.005305	0	0.017505	0.008442
74	0	0	0	0	0.013313	0.008134
75	0	0	0	0	0.010062	0.007826
76	0	0	0	0	0.008884	0.007518
77	0	0	0	0	0.006678	0.007211
78	0	0	0	0	0.001691	0.006903
79	0	0	0	0	0	0.006595
80	0	0	0	0	0	0.006287
81	0	0	0	0	0	0.00598
82	0	0	0	0	0	0.005672
83	0	0	0	0	0	0.005468
84	0	0	0	0	0	0.00505
85	0	0	0	0	0	0.004742
86	0	0	0	0	0	0.004435
87	0	0	0	0	0	0.004127
88	0	0	0	0	0	0.003819
89	0	0	0	0	0	0.003512
90	0	0	0	0	0	0.003204
91	0	0	0	0	0	0.002896
92	0	0	0	0	0	0.002588
93	0	0	0	0	0	0.00228
94	0	0	0	0	0	0.001973
95	0	0	0	0	0	0.001665
96	0	0	0	0	0	0.001357
97	0	0	0	0	0	0.00105
98	0	0	0	0	0	0.000742
99	0	0	0	0	0	0.000434
100	0	0	0	0	0	0.000127

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	E_ST_Plot 6a	E_ST_Plot 6b	E_ST_Plot 7	E_ST_Plot 7a	E_ST_Plot 8	E_ST_Plot 8a
1	0.000285	0.000369	0.001433	0.000538	0.0012	0.008223
2	0.000866	0.001121	0.004279	0.001599	0.003554	0.00951
3	0.001448	0.001874	0.007125	0.002703	0.006049	0.00974
4	0.00203	0.002627	0.009971	0.003736	0.008308	0.0103
5	0.002611	0.003379	0.012817	0.004868	0.010833	0.010699
6	0.003192	0.004132	0.015663	0.005873	0.011541	0.010909
7	0.003774	0.004884	0.018509	0.007033	0.01212	0.011489
8	0.004355	0.005637	0.021354	0.00801	0.012223	0.011887
9	0.004937	0.00639	0.0242	0.009198	0.012816	0.012078
10	0.005518	0.007142	0.027046	0.010147	0.012906	0.012675
11	0.0061	0.007895	0.029892	0.011364	0.013498	0.012648
12	0.006681	0.008648	0.032738	0.012284	0.013437	0.012916
13	0.007263	0.009352	0.035584	0.013529	0.013882	0.012971
14	0.007844	0.009757	0.03843	0.014421	0.013783	0.012808
15	0.008426	0.010111	0.040747	0.015694	0.014235	0.01308
16	0.009007	0.010465	0.040927	0.016558	0.014129	0.013135
17	0.009589	0.010819	0.04088	0.017859	0.014588	0.012969
18	0.01017	0.011173	0.040833	0.018695	0.014474	0.013243
19	0.010752	0.011526	0.040786	0.020024	0.01494	0.013298
20	0.011333	0.01188	0.041032	0.020832	0.01482	0.01313
21	0.011915	0.012234	0.040692	0.022189	0.015293	0.013407
22	0.012496	0.012588	0.040645	0.022969	0.015166	0.013237
23	0.013078	0.012942	0.040599	0.024354	0.015615	0.013516
24	0.013659	0.013182	0.040552	0.025064	0.015223	0.01357
25	0.014241	0.013304	0.040505	0.025608	0.01539	0.013398
26	0.014788	0.013427	0.040428	0.025177	0.014953	0.013679
27	0.014872	0.013549	0.039172	0.025409	0.015115	0.013734
28	0.014808	0.013671	0.036275	0.024981	0.014683	0.013559
29	0.014744	0.013794	0.033157	0.024883	0.014839	0.013843
30	0.014681	0.013916	0.029989	0.025059	0.014414	0.013898
31	0.014617	0.014039	0.026186	0.024364	0.014564	0.013719
32	0.014553	0.014161	0.022192	0.024287	0.014144	0.014006
33	0.01449	0.014283	0.018199	0.023579	0.014289	0.013826
34	0.014426	0.014406	0.014205	0.02349	0.013874	0.014115
35	0.014362	0.014528	0.009795	0.022792	0.014014	0.01417
36	0.014299	0.014651	0.003165	0.022694	0.013604	0.013987
37	0.014235	0.014773	0	0.022006	0.01347	0.014279
38	0.014171	0.014895	0	0.021898	0.013602	0.014333
39	0.014108	0.015018	0	0.02122	0.013201	0.014148
40	0.014044	0.01514	0	0.021101	0.013327	0.014442
41	0.01398	0.015262	0	0.020435	0.012931	0.014255
42	0.013917	0.015385	0	0.020305	0.013052	0.014551
43	0.013853	0.015507	0	0.019648	0.012661	0.014606
44	0.013789	0.015629	0	0.019506	0.012776	0.014416
45	0.013726	0.015752	0	0.018648	0.012391	0.014714
46	0.013662	0.015874	0	0.01816	0.012501	0.014769
47	0.013598	0.015874	0	0.0172	0.012121	0.014576
48	0.013535	0.015737	0	0.016693	0.012226	0.014878
49	0.013471	0.015601	0	0.015752	0.011851	0.014933
50	0.013407	0.015464	0	0.015225	0.01195	0.014737
51	0.013344	0.015327	0	0.014304	0.011581	0.015041
52	0.01328	0.01519	0	0.013758	0.011675	0.014844
53	0.013217	0.015054	0	0.012855	0.011311	0.01515
54	0.013153	0.014917	0	0.012291	0.0114	0.015205
55	0.013089	0.01478	0	0.011407	0.011042	0.015005
56	0.013025	0.014643	0	0.010687	0.011125	0.015314
57	0.012962	0.014506	0	0.010094	0.010772	0.015369
58	0.012898	0.01437	0	0.009239	0.01085	0.015166

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	E_ST_Plot 6a	E_ST_Plot 6b	E_ST_Plot 7	E_ST_Plot 7a	E_ST_Plot 8	E_ST_Plot 8a
59	0.012834	0.014233	0	0.008627	0.010502	0.015477
60	0.012771	0.014085	0	0.007732	0.010617	0.015532
61	0.012707	0.013785	0	0.006577	0.01037	0.015323
62	0.012644	0.013436	0	0.005157	0.010539	0.01555
63	0.01258	0.013087	0	0.003872	0.010295	0.015227
64	0.012516	0.012738	0	0.002487	0.010462	0.015419
65	0.012453	0.012389	0	0.001167	0.010219	0.015354
66	0.012389	0.01204	0	8.69E-05	0.010385	0.015033
67	0.012325	0.011691	0	0	0.010143	0.014842
68	0.012261	0.011342	0	0	0.010307	0.013504
69	0.012198	0.010993	0	0	0.010067	0.011938
70	0.012134	0.010644	0	0	0.010197	0.010778
71	0.012071	0.010294	0	0	0.009906	0.009403
72	0.012007	0.009945	0	0	0.009815	0.007906
73	0.011943	0.009596	0	0	0.009917	0.006677
74	0.01188	0.009247	0	0	0.009631	0.005225
75	0.011816	0.008898	0	0	0.009729	0.00395
76	0.011752	0.008549	0	0	0.009447	0.002596
77	0.011689	0.0082	0	0	0.009542	0.001935
78	0.011396	0.007851	0	0	0.009214	0.001699
79	0.010352	0.007502	0	0	0.008733	0.001429
80	0.009249	0.007153	0	0	0.007776	0.001142
81	0.008147	0.006804	0	0	0.007131	0.000894
82	0.007044	0.006454	0	0	0.006206	0.000615
83	0.006017	0.006105	0	0	0.005529	0.000358
84	0.005763	0.005756	0	0	0.004635	9.17E-05
85	0.005711	0.005407	0	0	0.003927	0
86	0.005659	0.005058	0	0	0.003065	0
87	0.005607	0.004709	0	0	0.002381	0
88	0.005554	0.00436	0	0	0.002053	0
89	0.005502	0.004011	0	0	0.001912	0
90	0.00545	0.003662	0	0	0.001697	0
91	0.005398	0.003313	0	0	0.001549	0
92	0.005346	0.002964	0	0	0.00134	0
93	0.005294	0.002615	0	0	0.001185	0
94	0.005004	0.002266	0	0	0.000983	0
95	0.004243	0.001917	0	0	0.000821	0
96	0.003471	0.001567	0	0	0.000627	0
97	0.002697	0.001218	0	0	0.000457	0
98	0.001925	0.000869	0	0	0.00027	0
99	0.001152	0.00052	0	0	9.37E-05	0
100	0.000379	0.000171	0	0	0	0

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	E_ST_Plot 8b	E_ST_Plot 9	E_ST_Plot 9a	E_ST_Plot 9b
1	0.000766	7.84E-05	0.006658	0.000133
2	0.002328	0.000239	0.015505	0.000403
3	0.00389	0.000389	0.015902	0.000674
4	0.005452	0.000556	0.015974	0.000944
5	0.007015	0.0007	0.016376	0.001215
6	0.008578	0.000873	0.016443	0.001454
7	0.01014	0.001033	0.016676	0.001751
8	0.011702	0.001168	0.017085	0.002022
9	0.013264	0.00135	0.017146	0.002292
10	0.014525	0.001479	0.017559	0.002563
11	0.014896	0.001667	0.017615	0.002775
12	0.015209	0.00179	0.017848	0.003099
13	0.015522	0.001984	0.018269	0.003369
14	0.015835	0.002144	0.018317	0.00364
15	0.016149	0.002258	0.018743	0.003911
16	0.016462	0.002461	0.018786	0.004181
17	0.016775	0.002569	0.019217	0.004362
18	0.017088	0.002778	0.019255	0.004718
19	0.017401	0.00288	0.019489	0.004988
20	0.017714	0.003095	0.019926	0.005259
21	0.018028	0.003255	0.019958	0.005529
22	0.018341	0.003348	0.0204	0.005684
23	0.018654	0.003572	0.020427	0.006065
24	0.018967	0.003659	0.02066	0.006336
25	0.01928	0.003889	0.02111	0.006606
26	0.019594	0.004049	0.021129	0.006877
27	0.019907	0.004127	0.021584	0.007148
28	0.02022	0.004366	0.021598	0.00727
29	0.020533	0.004438	0.022058	0.007684
30	0.020846	0.004683	0.022067	0.007954
31	0.021133	0.004749	0.022301	0.008225
32	0.02087	0.005118	0.022767	0.008495
33	0.020389	0.005995	0.02277	0.008592
34	0.019909	0.006814	0.023241	0.009031
35	0.019429	0.007899	0.023239	0.009302
36	0.018949	0.008681	0.023472	0.009836
37	0.018468	0.009802	0.023951	0.011087
38	0.017988	0.010547	0.023941	0.012122
39	0.017508	0.011706	0.024425	0.013372
40	0.017027	0.012667	0.02441	0.013551
41	0.016547	0.014296	0.024643	0.01367
42	0.016067	0.016145	0.024972	0.013788
43	0.015587	0.016021	0.023605	0.013907
44	0.015106	0.016526	0.022478	0.01375
45	0.014626	0.016395	0.020893	0.014142
46	0.014146	0.016905	0.019739	0.014261
47	0.013665	0.017097	0.018182	0.014379
48	0.013185	0.016955	0.016834	0.014498
49	0.012705	0.017478	0.015637	0.014329
50	0.012225	0.017327	0.014122	0.014733
51	0.011744	0.017857	0.009976	0.014852
52	0.011264	0.018049	0.000624	0.01497
53	0.010784	0.017889	0	0.015089
54	0.010303	0.01843	0	0.015208
55	0.009823	0.018252	0	0.015025
56	0.009343	0.018657	0	0.015443
57	0.008863	0.018301	0	0.015561

Table D-7. List of Inventory Fractions for Segment 1 Cells for those CA Units within the GSA Flow Model Domain (Hamm et al. 2009, Appendix C)

Segment 1 Cell Number	E_ST_Plot 8b	E_ST_Plot 9	E_ST_Plot 9a	E_ST_Plot 9b
58	0.008382	0.018663	0	0.01568
59	0.007902	0.018664	0	0.015799
60	0.007422	0.018307	0	0.015604
61	0.006942	0.018668	0	0.016034
62	0.006478	0.018312	0	0.016153
63	0.006242	0.018673	0	0.016271
64	0.006075	0.018315	0	0.01639
65	0.005909	0.018676	0	0.016422
66	0.005742	0.018679	0	0.015901
67	0.005576	0.018322	0	0.016004
68	0.005409	0.018683	0	0.015786
69	0.005243	0.018326	0	0.015568
70	0.005076	0.018662	0	0.01535
71	0.00491	0.018326	0	0.014838
72	0.004743	0.017558	0	0.014919
73	0.004577	0.017478	0	0.014701
74	0.00441	0.016726	0	0.014483
75	0.004244	0.016631	0	0.014265
76	0.004077	0.015895	0	0.013774
77	0.003911	0.015783	0	0.013834
78	0.003745	0.015355	0	0.013616
79	0.003578	0.014644	0	0.013398
80	0.003412	0.014507	0	0.01318
81	0.003245	0.013813	0	0.012962
82	0.003079	0.01366	0	0.012497
83	0.002912	0.012981	0	0.012531
84	0.002745	0.012812	0	0.012313
85	0.002579	0.012384	0	0.012095
86	0.002412	0.01173	0	0.011877
87	0.002246	0.011537	0	0.011433
88	0.002079	0.010899	0	0.011446
89	0.001913	0.010407	0	0.011228
90	0.001747	0.009012	0	0.01101
91	0.00158	0.007953	0	0.010792
92	0.001414	0.006706	0	0.010574
93	0.001247	0.005365	0	0.010009
94	0.001081	0.004235	0	0.009383
95	0.000914	0.002941	0	0.008019
96	0.000748	0.001764	0	0.005947
97	0.000581	0.00052	0	0.003869
98	0.000415	0	0	0.001777
99	0.000248	0	0	0.000152
100	8.15E-05	0	0	0

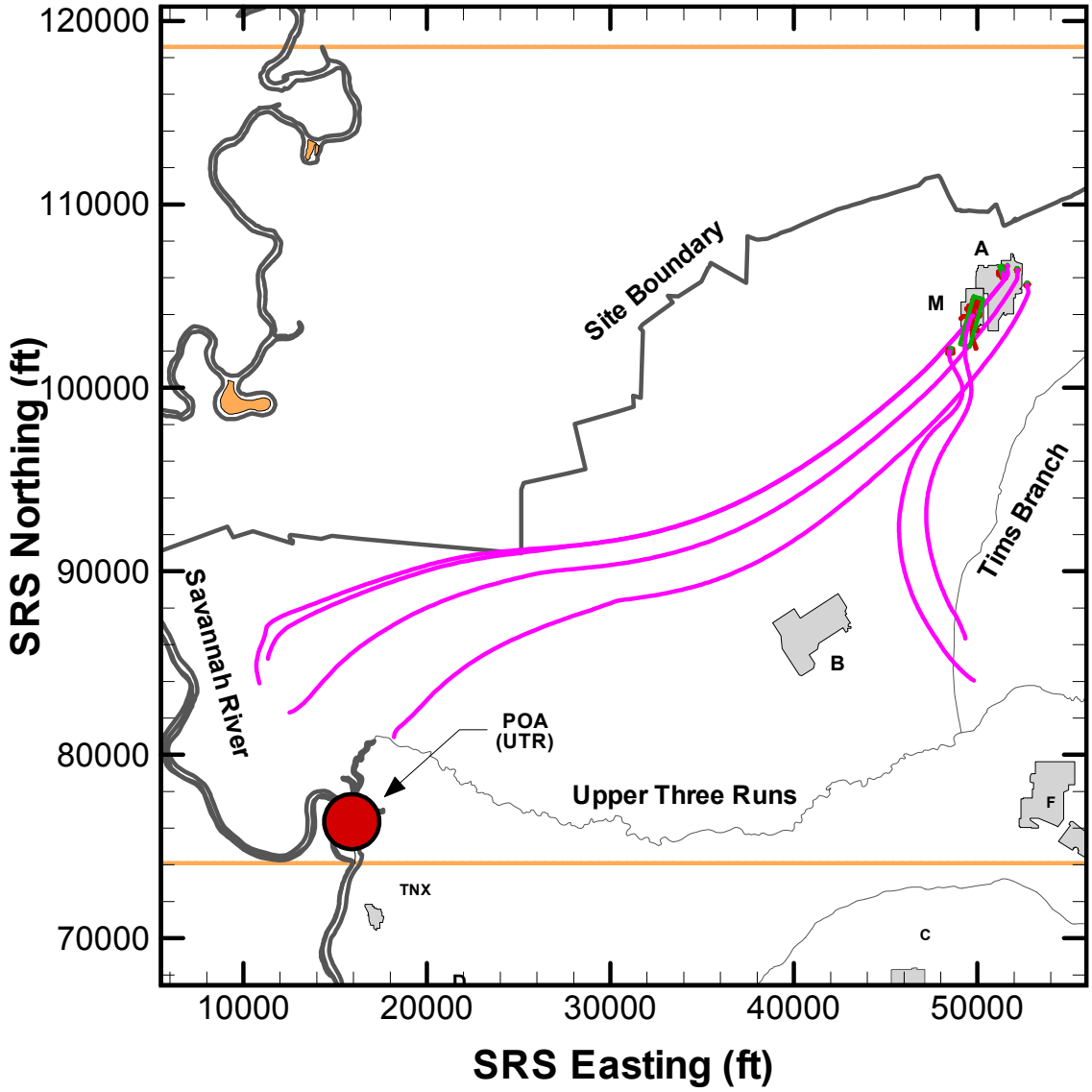
Table D-8. List of Inventory Fractions for Segment 1 Cells for those CA Units within the R Flow Model Domain (Hamm et al. 2009, Appendix D)

Table D-8. List of Inventory Fractions for Segment 1 Cells for those CA Units within the R Flow Model Domain (Hamm et al. 2009, Appendix D)				
Segment 1 Cell Number	R_Reactor_ Vessel	R_Reactor _Bld_Long	R_Reactor_ Bld_Short	R_SeepBasin
1	0.545703	0.176471	0.008979	0.026965
2	0.454297	0.147059	0.01459	0.047015
3	0	0.161765	0.016835	0.046278
4	0	0.132353	0.017957	0.045823
5	0	0.117647	0.01459	0.045089
6	0	0.088235	0.017957	0.04463
7	0	0	0.022447	0.044033
8	0	0	0.028058	0.043305
9	0	0	0.025814	0.043692
10	0	0	0.023569	0.088228
11	0	0	0.024691	0.120074
12	0	0	0.022447	0.121075
13	0	0	0.022447	0.121702
14	0	0	0.023569	0.111555
15	0	0	0.023569	0.048956
16	0	0	0.022447	0.001582
17	0	0	0.024691	0
18	0	0	0.025814	0
19	0	0	0.029181	0
20	0	0	0.034792	0
21	0	0	0.040404	0
22	0	0	0.041526	0
23	0	0	0.041526	0
24	0	0	0.035915	0
25	0	0	0.037037	0
26	0	0	0.037037	0
27	0	0	0.037037	0
28	0	0	0.039282	0
29	0	0	0.041526	0
30	0	0	0.037037	0
31	0	0.029412	0.035915	0
32	0	0.044118	0.03367	0
33	0	0.029412	0.034792	0
34	0	0.029412	0.029181	0
35	0	0.029412	0.023569	0
36	0	0.014706	0.010101	0
37	0	0	0	0
38	0	0	0	0
39	0	0	0	0
40	0	0	0	0
41	0	0	0	0
42	0	0	0	0
43	0	0	0	0
44	0	0	0	0
45	0	0	0	0
46	0	0	0	0
47	0	0	0	0
48	0	0	0	0
49	0	0	0	0
50	0	0	0	0
51	0	0	0	0
52	0	0	0	0
53	0	0	0	0
54	0	0	0	0

Table D-8. List of Inventory Fractions for Segment 1 Cells for those CA Units within the R Flow Model Domain (Hamm et al. 2009, Appendix D)

Segment 1 Cell Number	R_Reactor_ Vessel	R_Reactor _Bld_Long	R_Reactor_ Bld_Short	R_SeepBasin
55	0	0	0	0
56	0	0	0	0
57	0	0	0	0
58	0	0	0	0
59	0	0	0	0
60	0	0	0	0
61	0	0	0	0
62	0	0	0	0
63	0	0	0	0
64	0	0	0	0
65	0	0	0	0
66	0	0	0	0
67	0	0	0	0
68	0	0	0	0
69	0	0	0	0
70	0	0	0	0
71	0	0	0	0
72	0	0	0	0
73	0	0	0	0
74	0	0	0	0
75	0	0	0	0
76	0	0	0	0
77	0	0	0	0
78	0	0	0	0
79	0	0	0	0
80	0	0	0	0
81	0	0	0	0
82	0	0	0	0
83	0	0	0	0
84	0	0	0	0
85	0	0	0	0
86	0	0	0	0
87	0	0	0	0
88	0	0	0	0
89	0	0	0	0
90	0	0	0	0
91	0	0	0	0
92	0	0	0	0
93	0	0	0	0
94	0	0	0	0
95	0	0	0	0
96	0	0	0	0
97	0	0	0	0
98	0	0	0	0
99	0	0	0	0
100	0	0	0	0

D.2 AQUIFER FLOW FIGURES



Each unit's average stream trace from its footprint to its corresponding discharge point is shown. Also provided is the one POA within this flow model domain.

Figure D-1. View of those CA Units within the AM Flow Model Domain

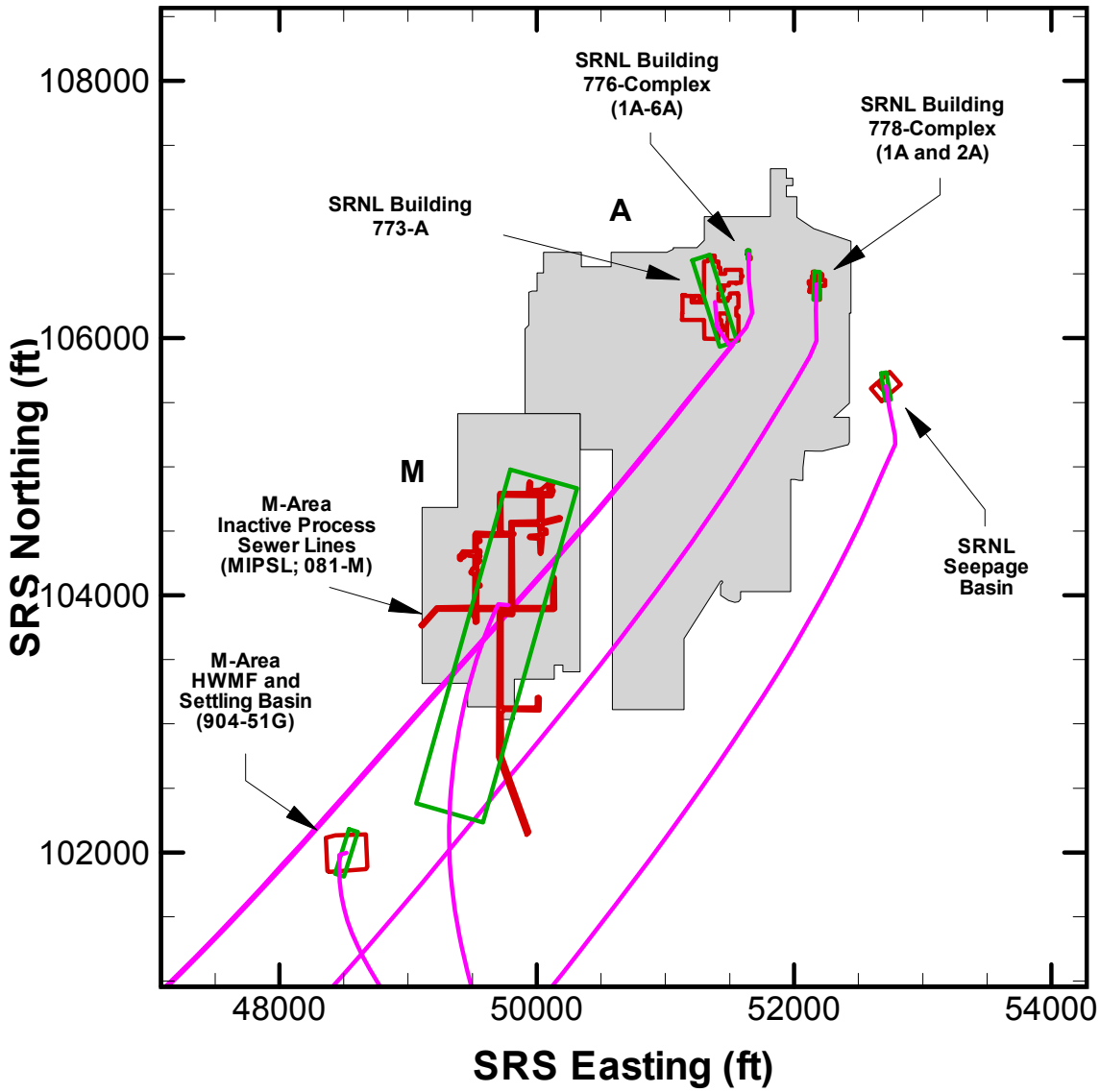
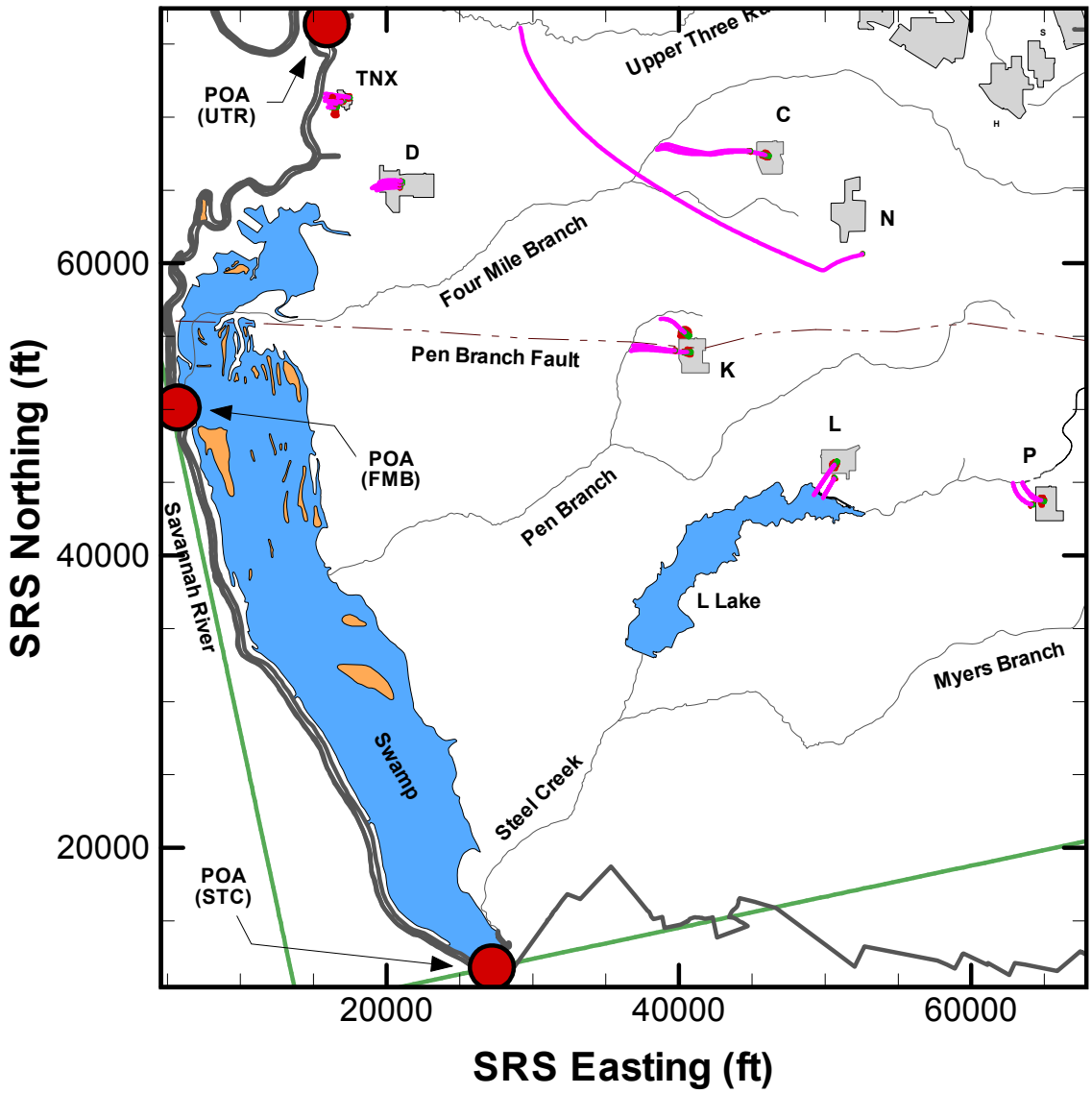


Figure D-2. Close-Up View of the Actual Footprints of the CA Units within the AM Flow Model Domain



Each unit's average stream trace from its footprint to its corresponding discharge point is shown. Also provided are the three POAs within this flow model domain.

Figure D-3. View of those CA Units within the CKLP Flow Model Domain

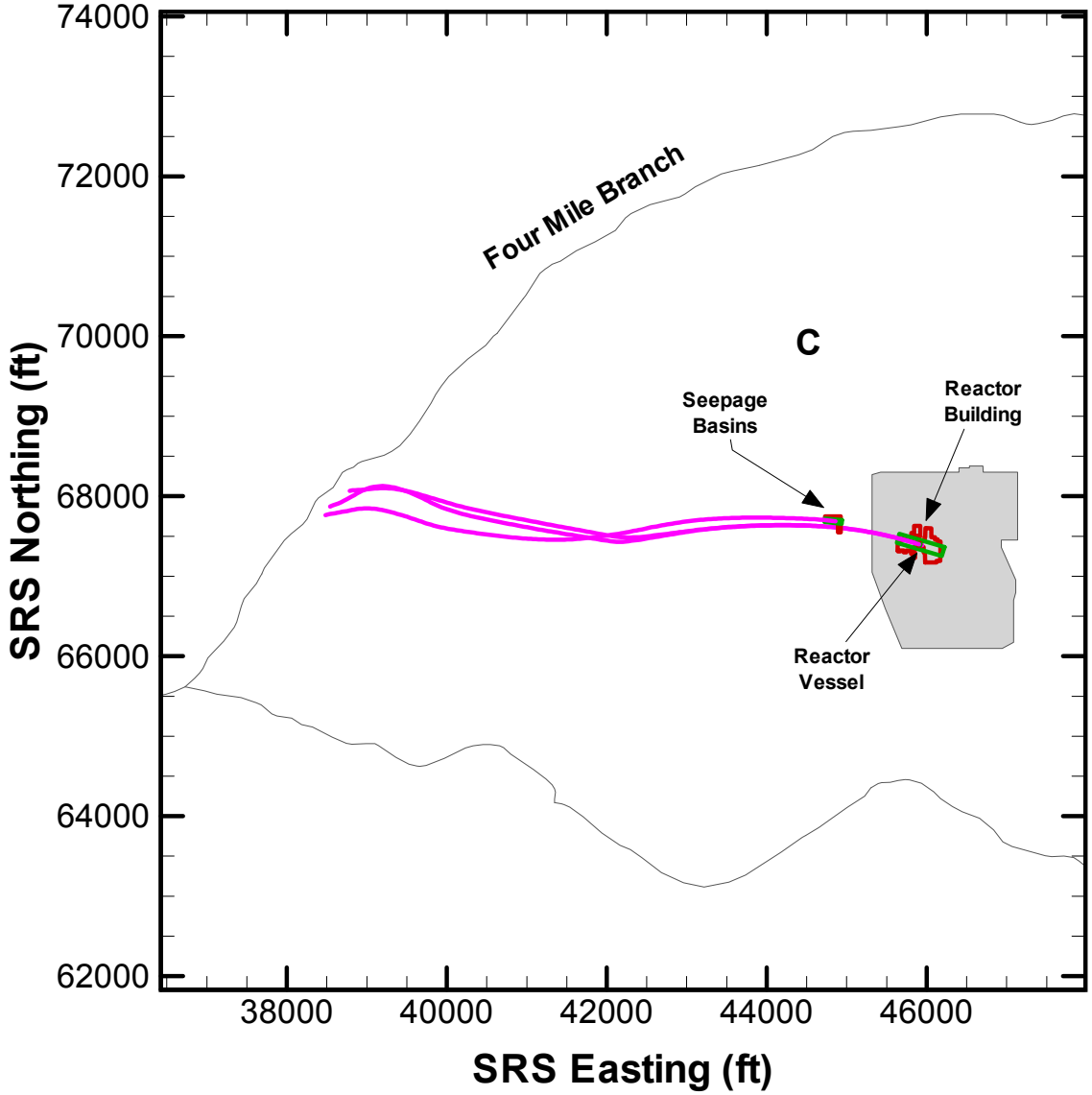


Figure D-4. Close-Up view of the C-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

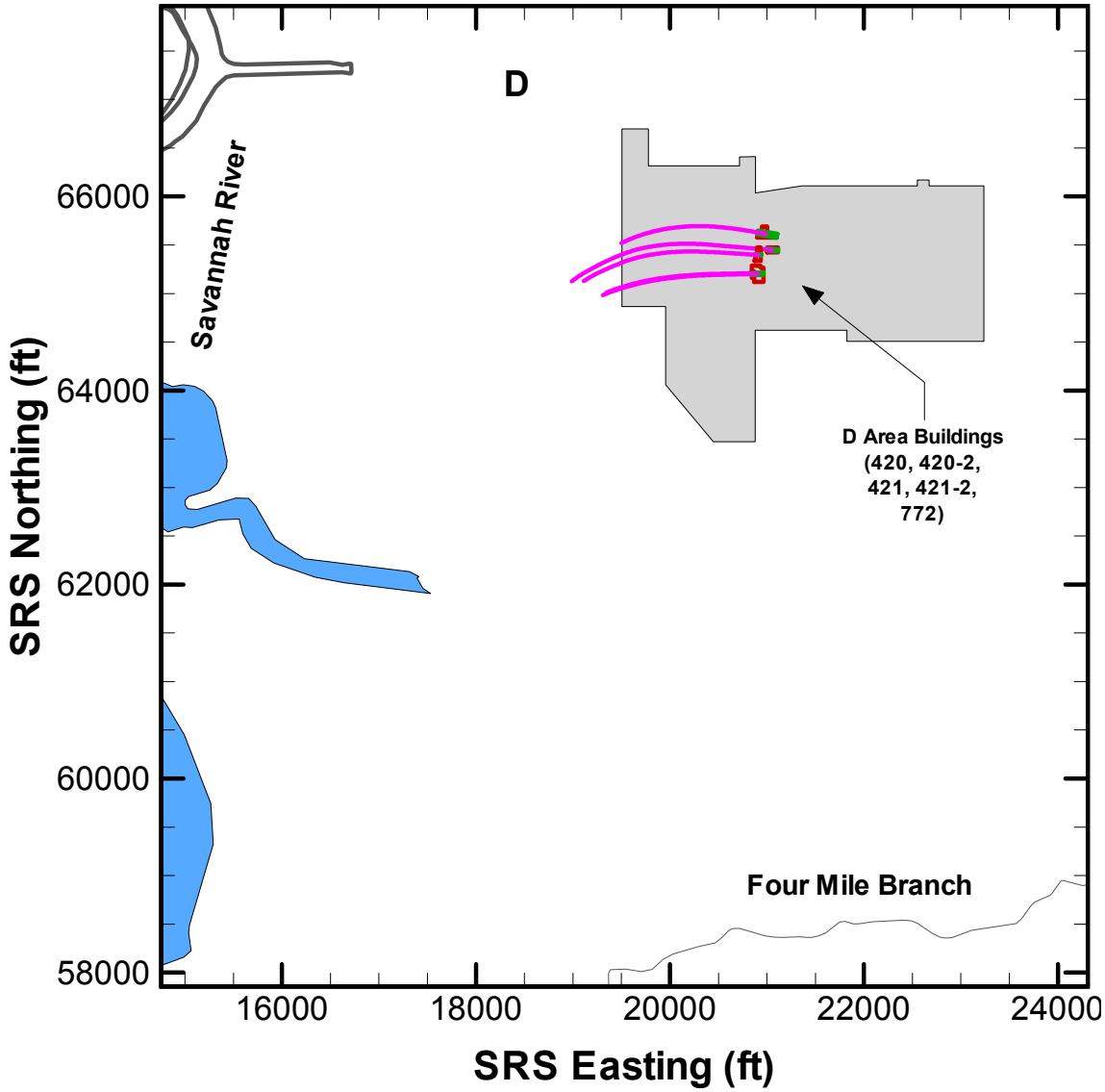


Figure D-5. Close-Up view of the D-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

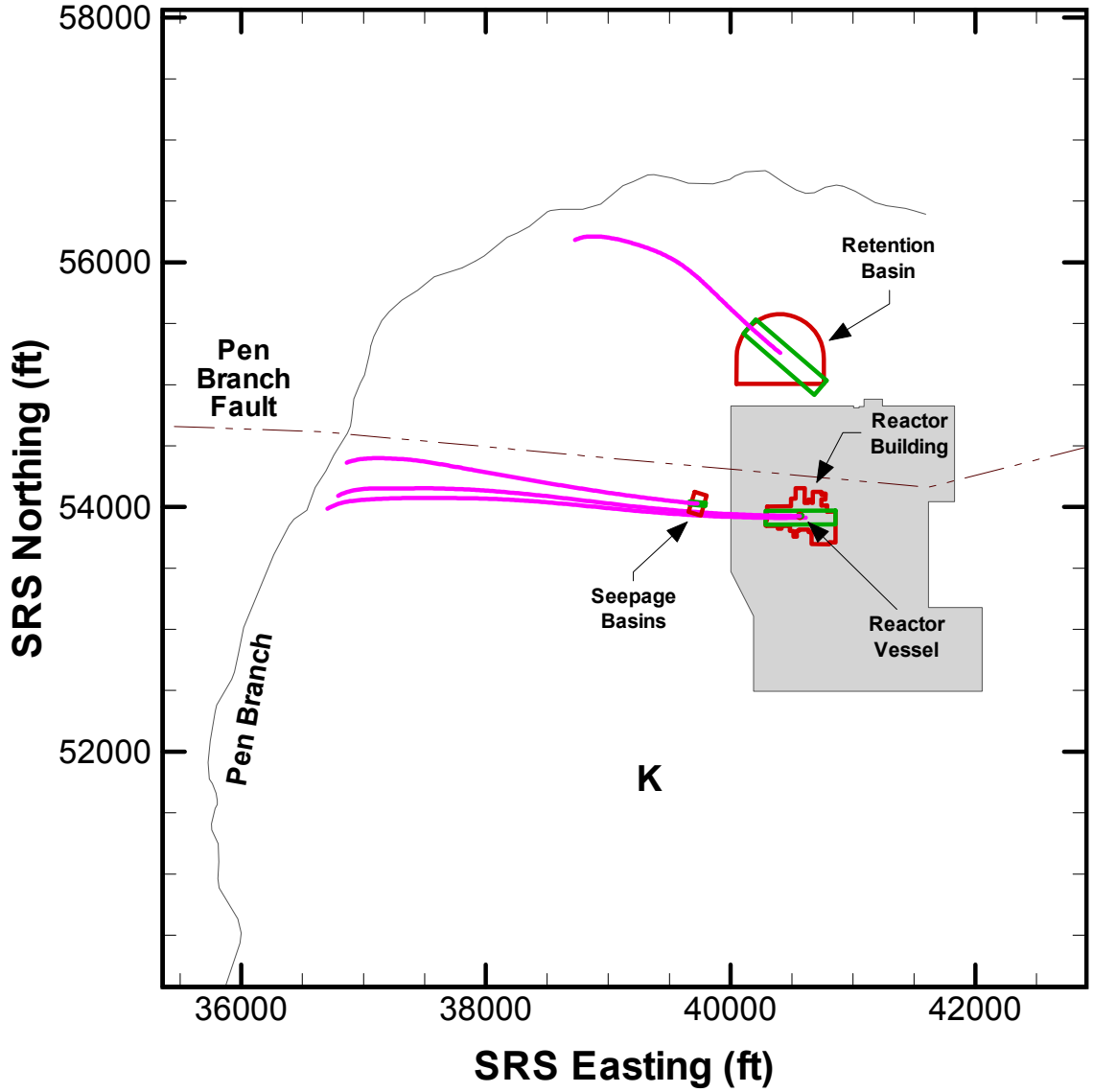


Figure D-6. Close-Up view of the K-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

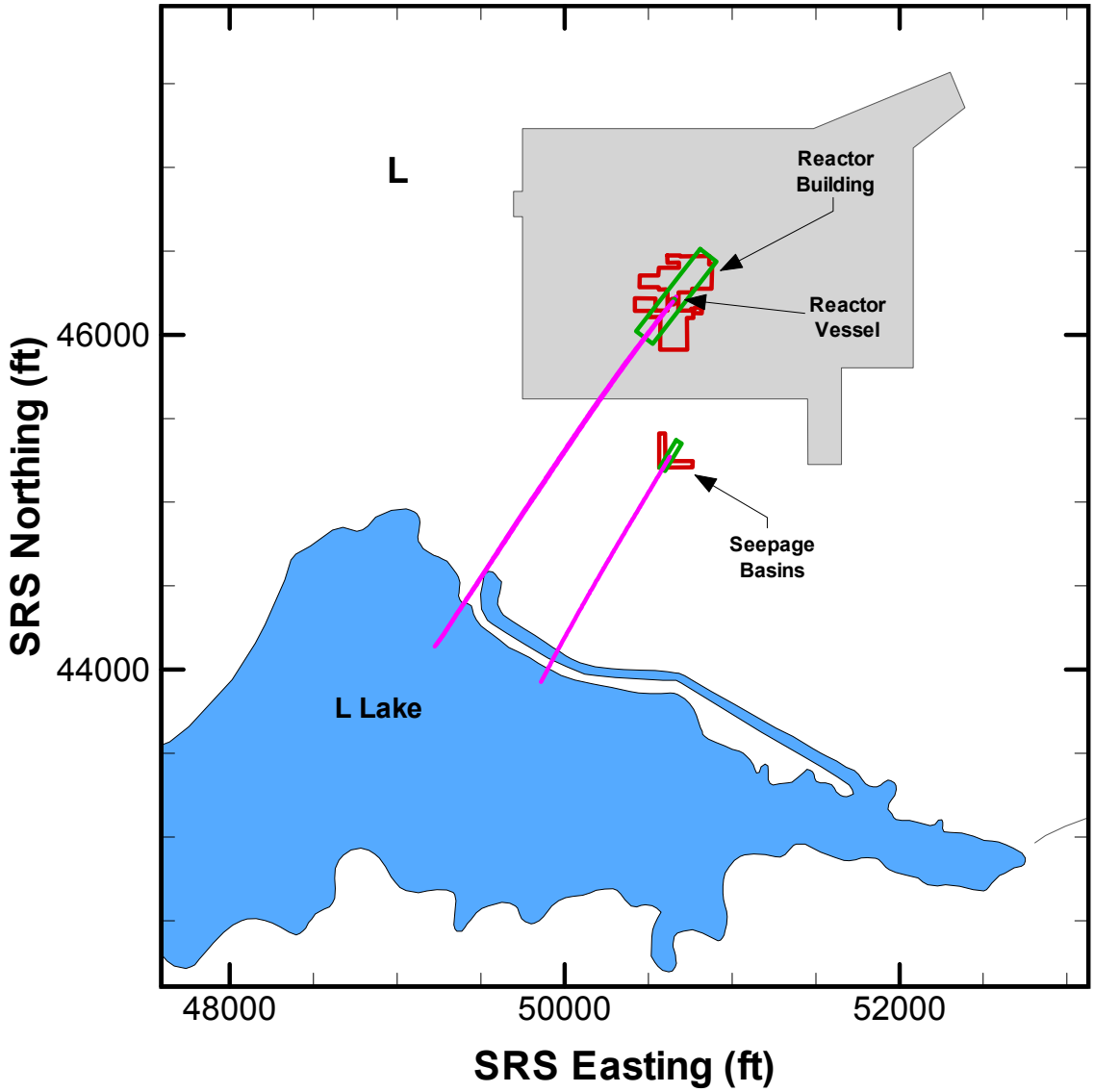
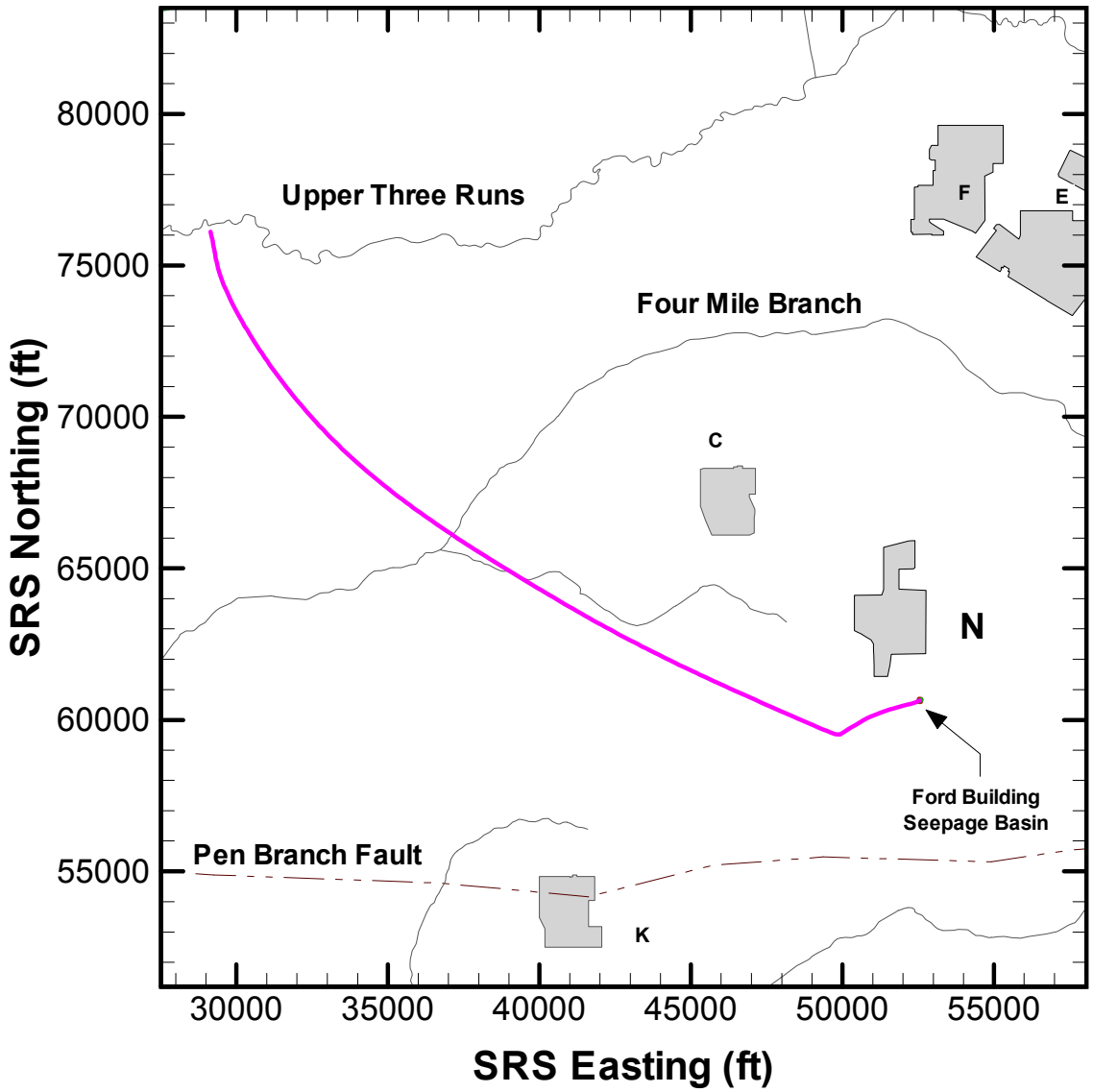


Figure D-7. Close-Up view of the L-Area CA Units Showing Each Unit's Footprint and Average Stream Trace



Note that stream trace changes direction due to dropping into Gordon Aquifer Unit and then travels up to UTR.

Figure D-8. Close-Up view of the N-Area CA Unit Showing its Footprint and Average Stream Trace

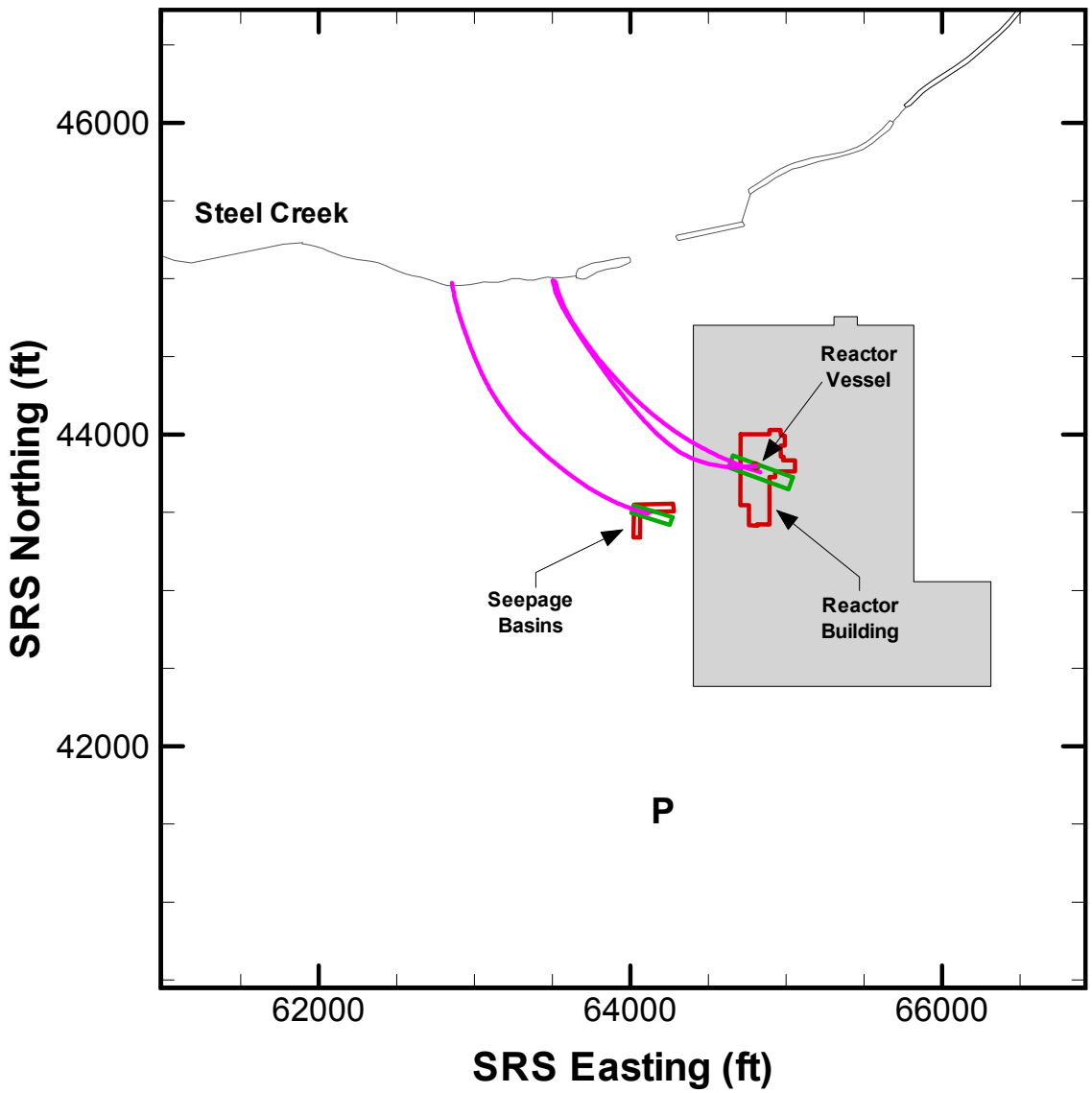


Figure D-9. Close-Up View of the P-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

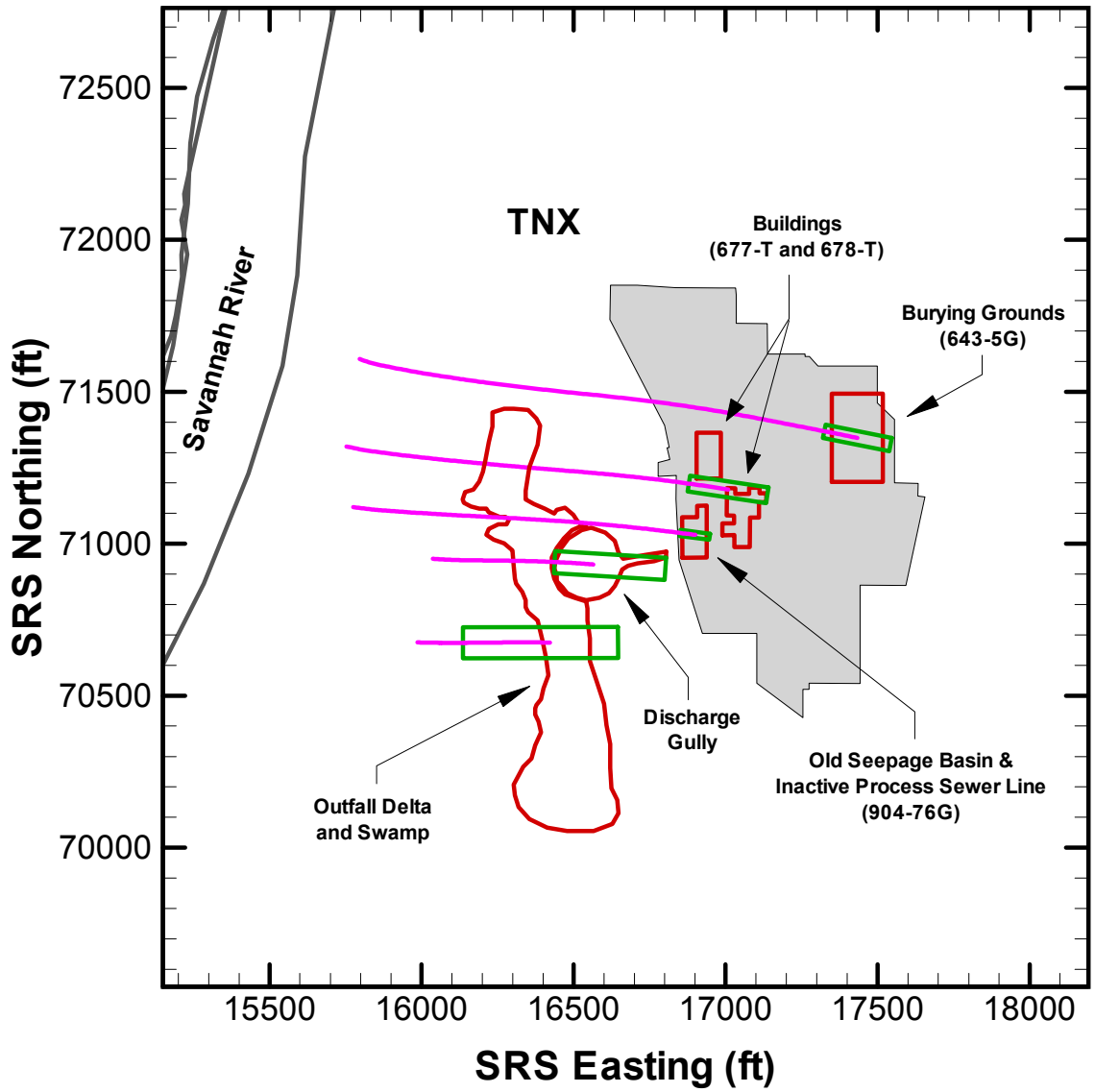
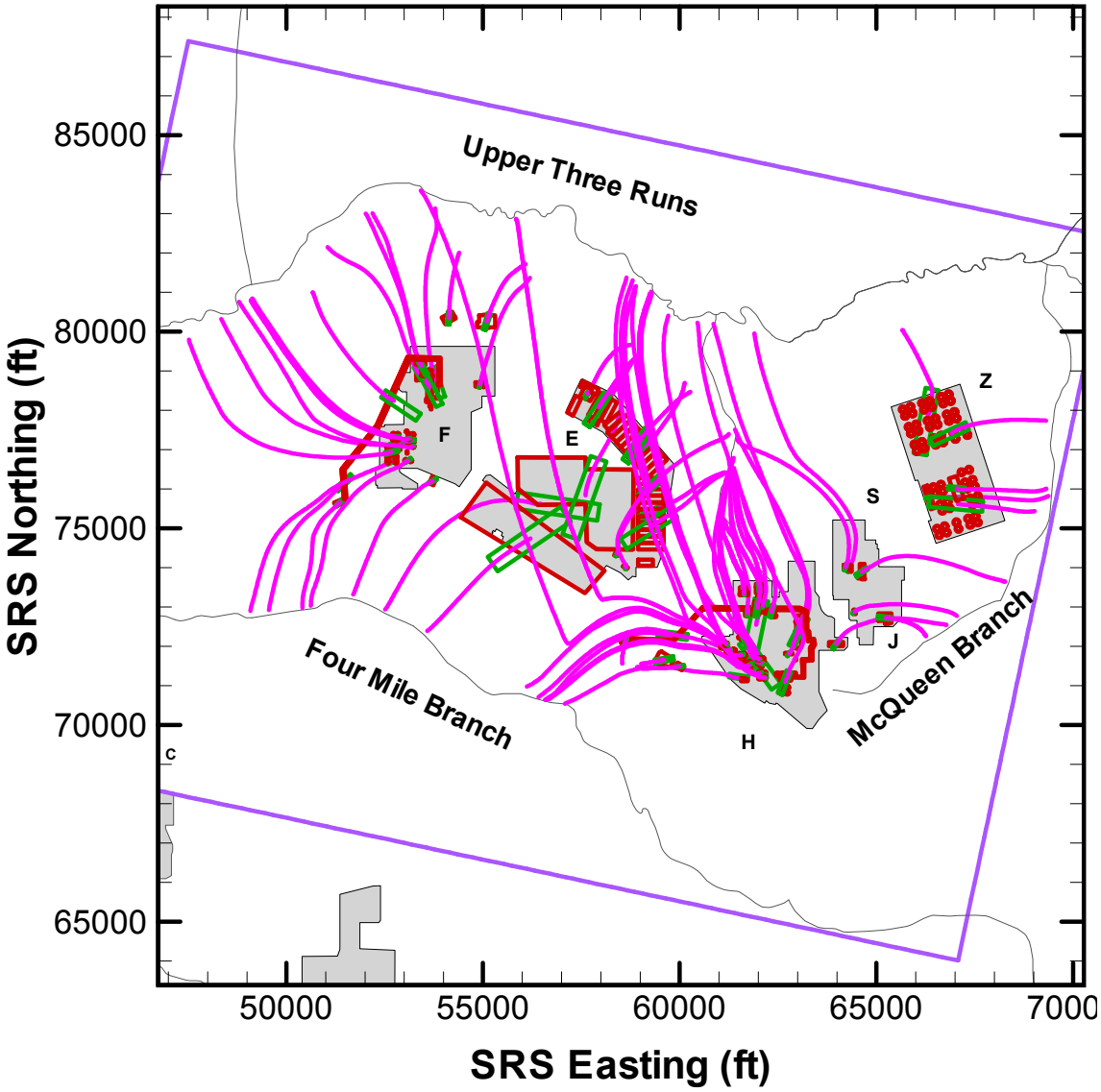


Figure D-10. Close-Up View of the T-Area CA Units Showing Each Unit's Footprint and Average Stream Trace



Each unit's average stream trace from its footprint to its corresponding discharge point is shown.

Figure D-11. View of those CA Units within the GSA Flow Model Domain

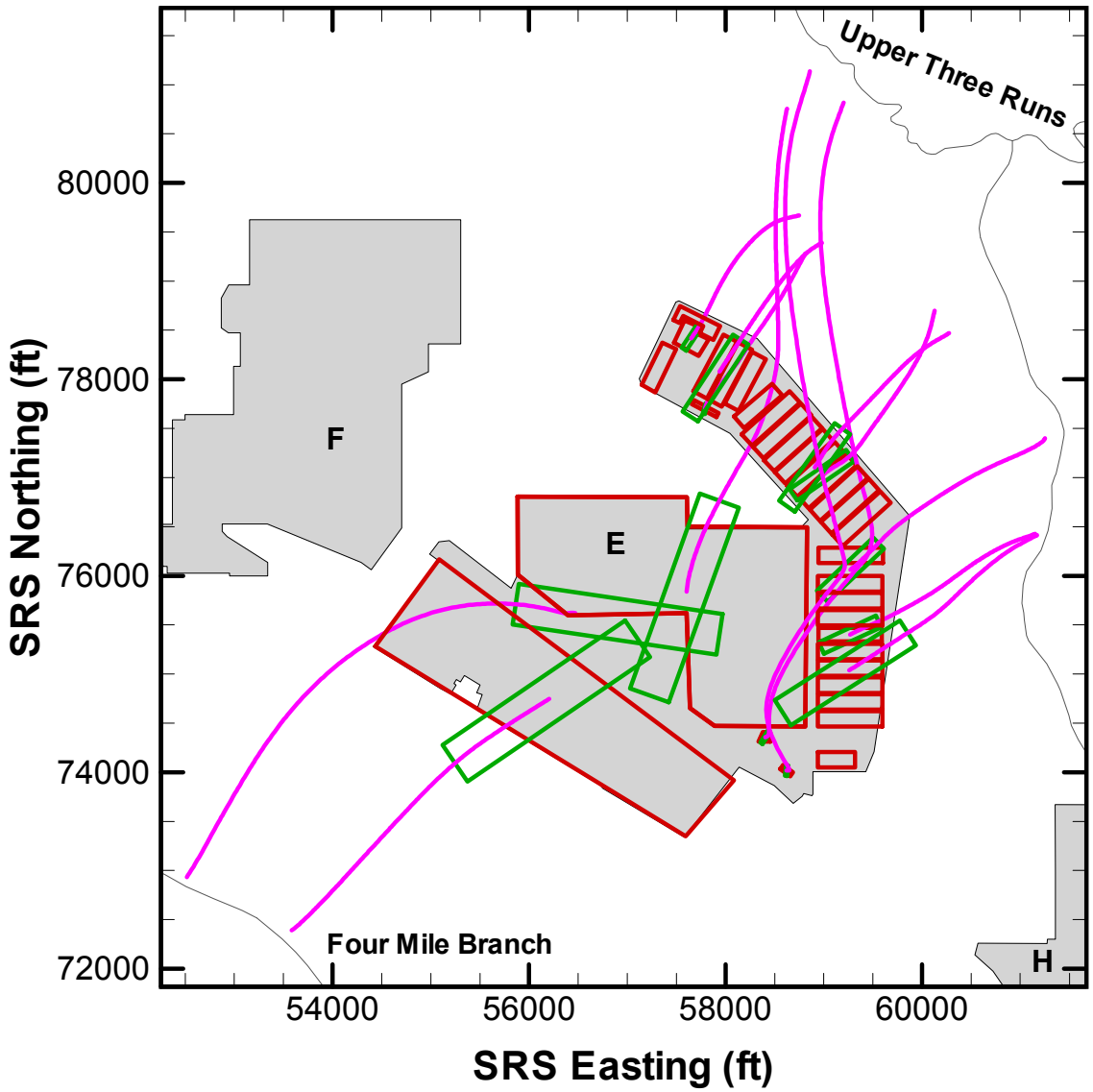


Figure D-12. Close-Up View of the T-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

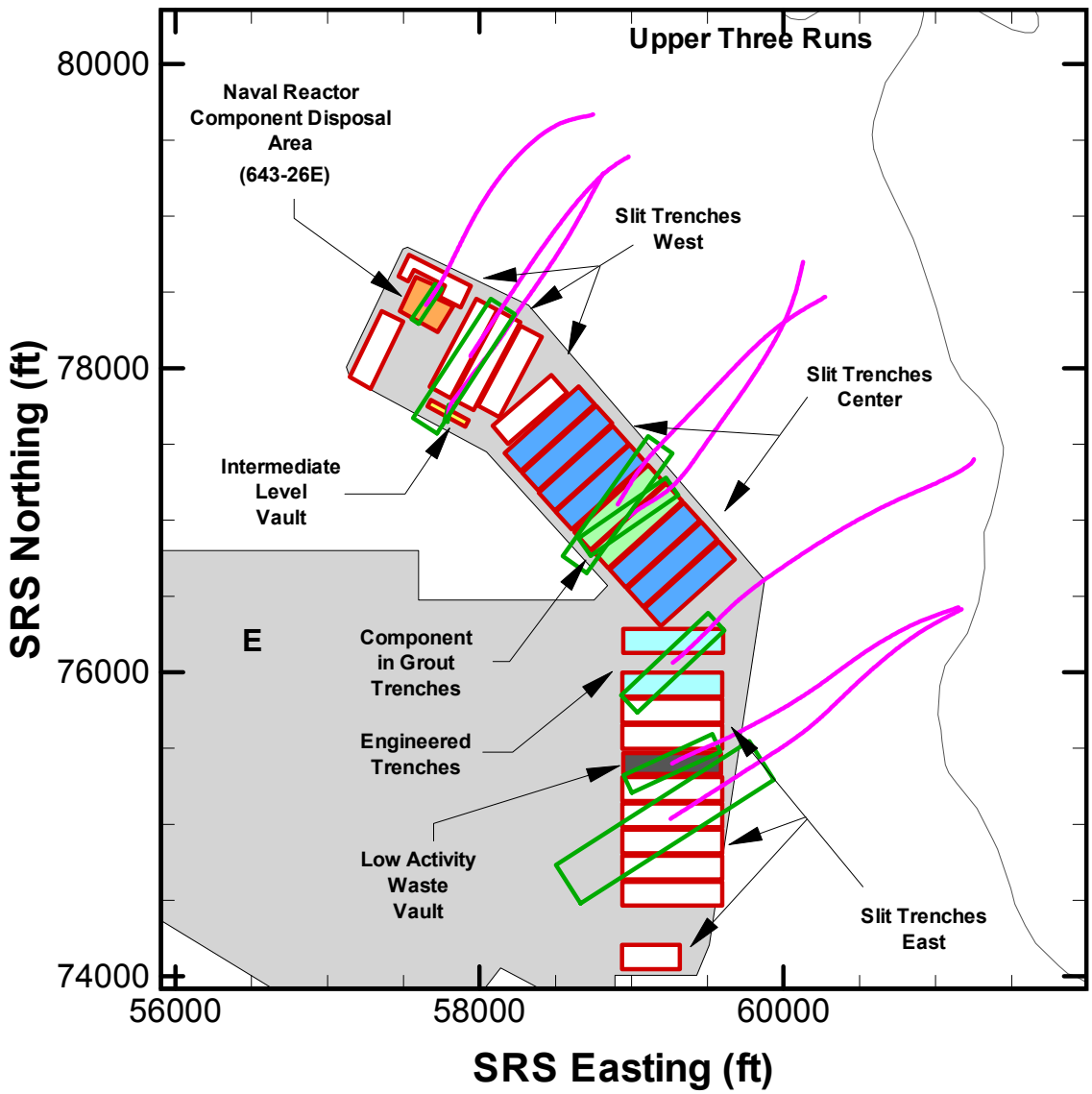


Figure D-13. First Close-Up View of the E-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

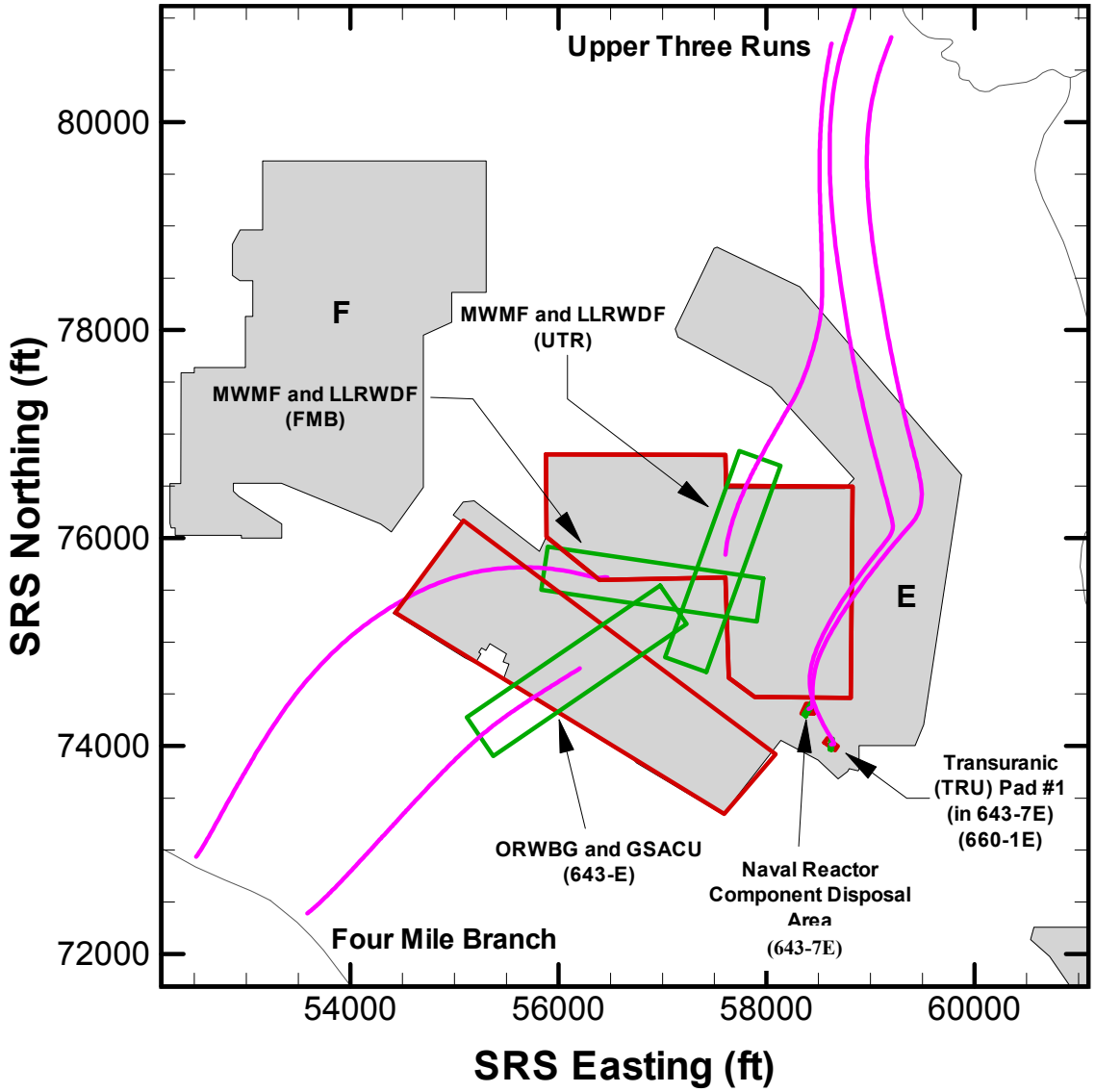


Figure D-14. Second Close-Up View of the E-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

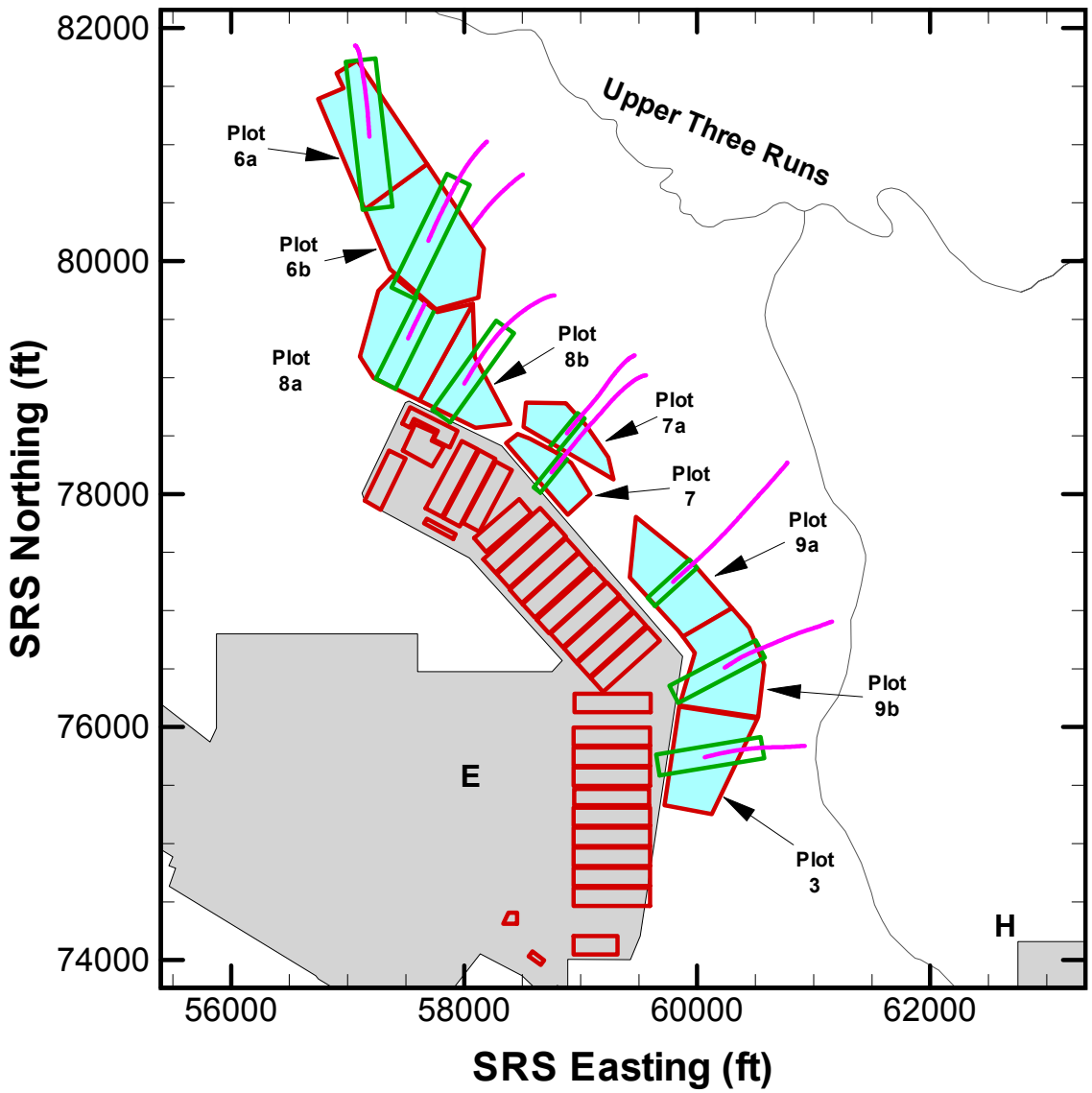


Figure D-15. Close-Up View of the E-Area CA Units (Proposed Expansion) Showing Each Unit's Footprint and Average Stream Trace

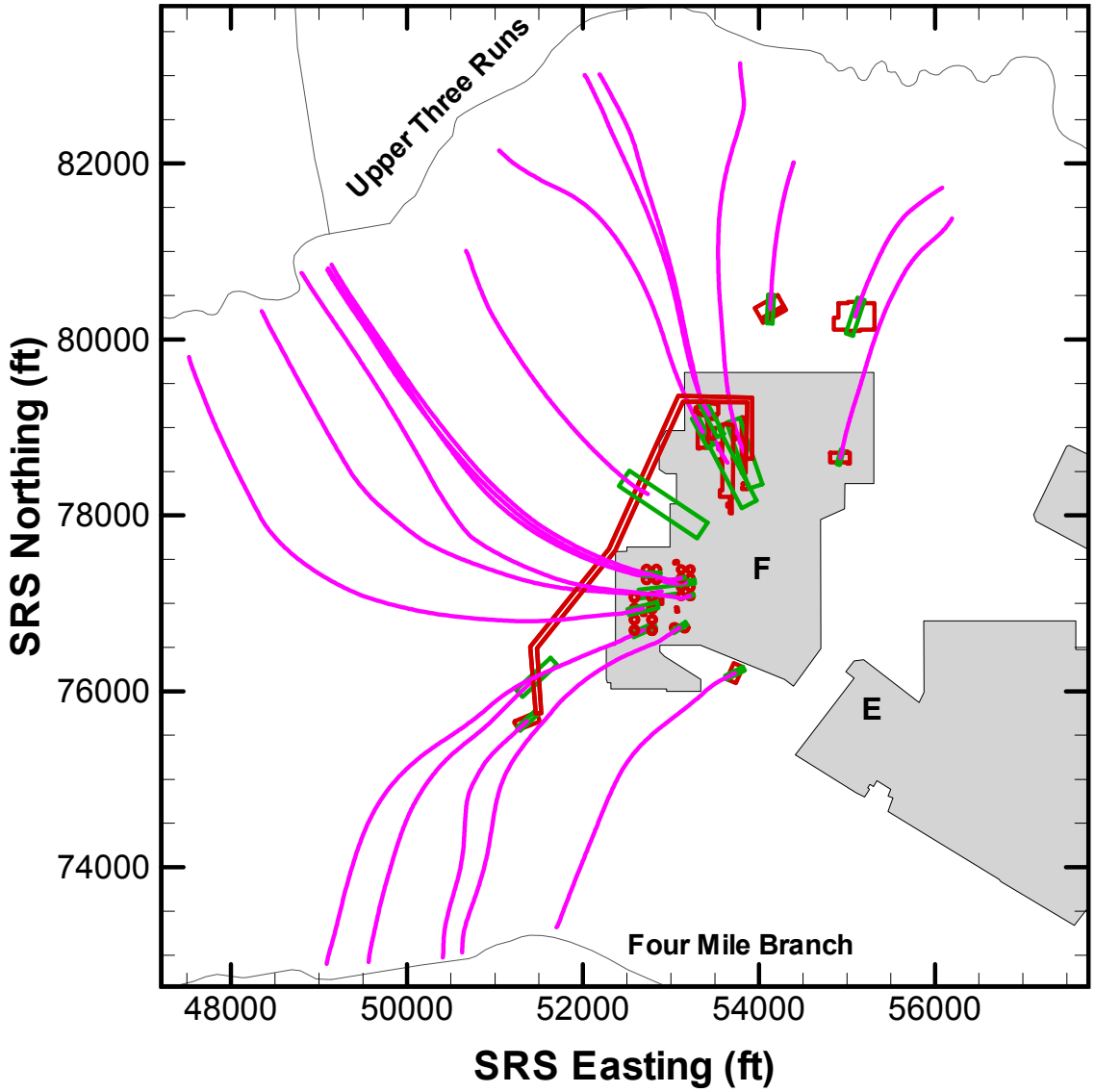


Figure D-16. Close-Up View of the F-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

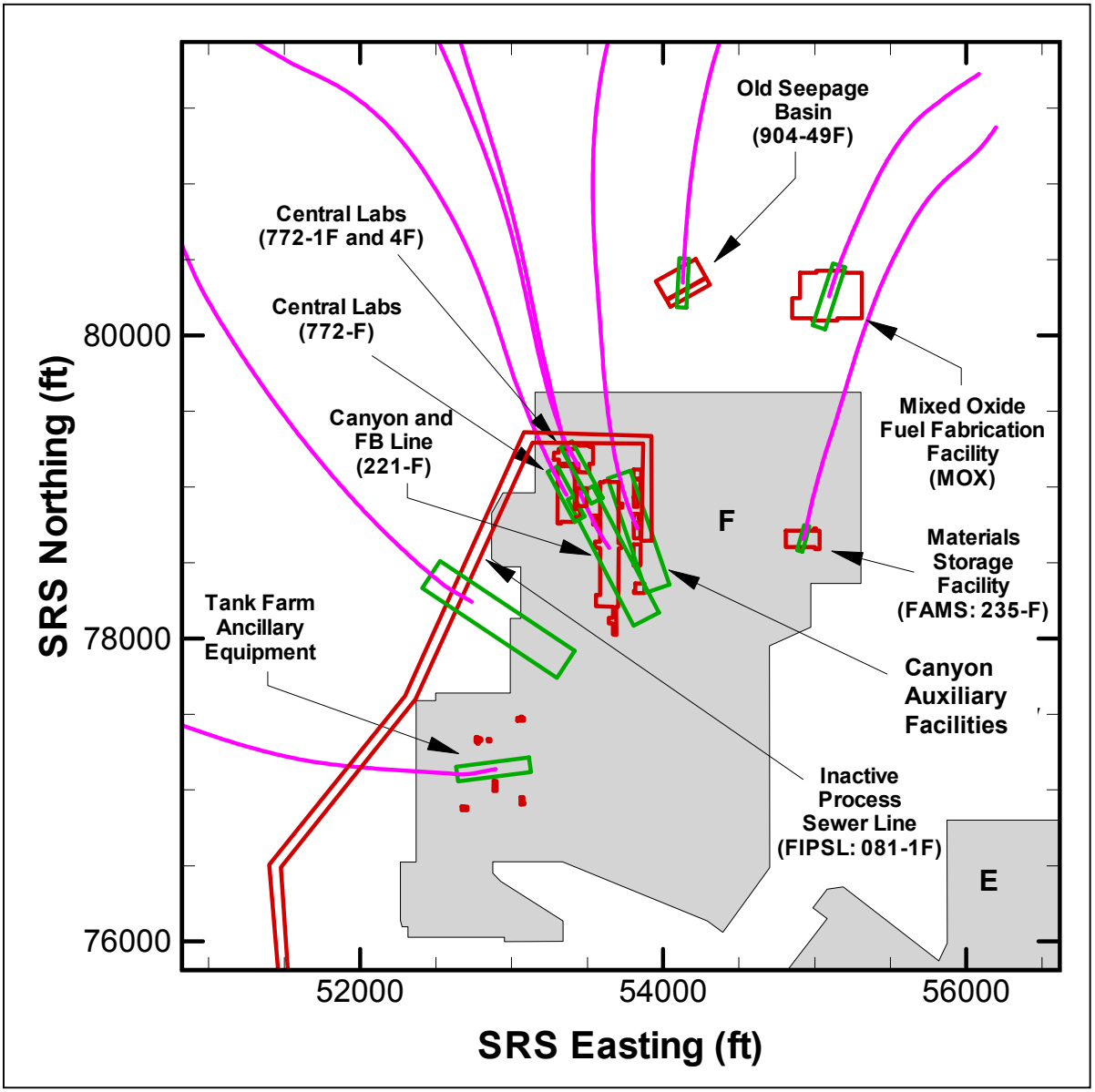


Figure D-17. First Close-Up View of the Actual Footprints of the CA Units within F-Area

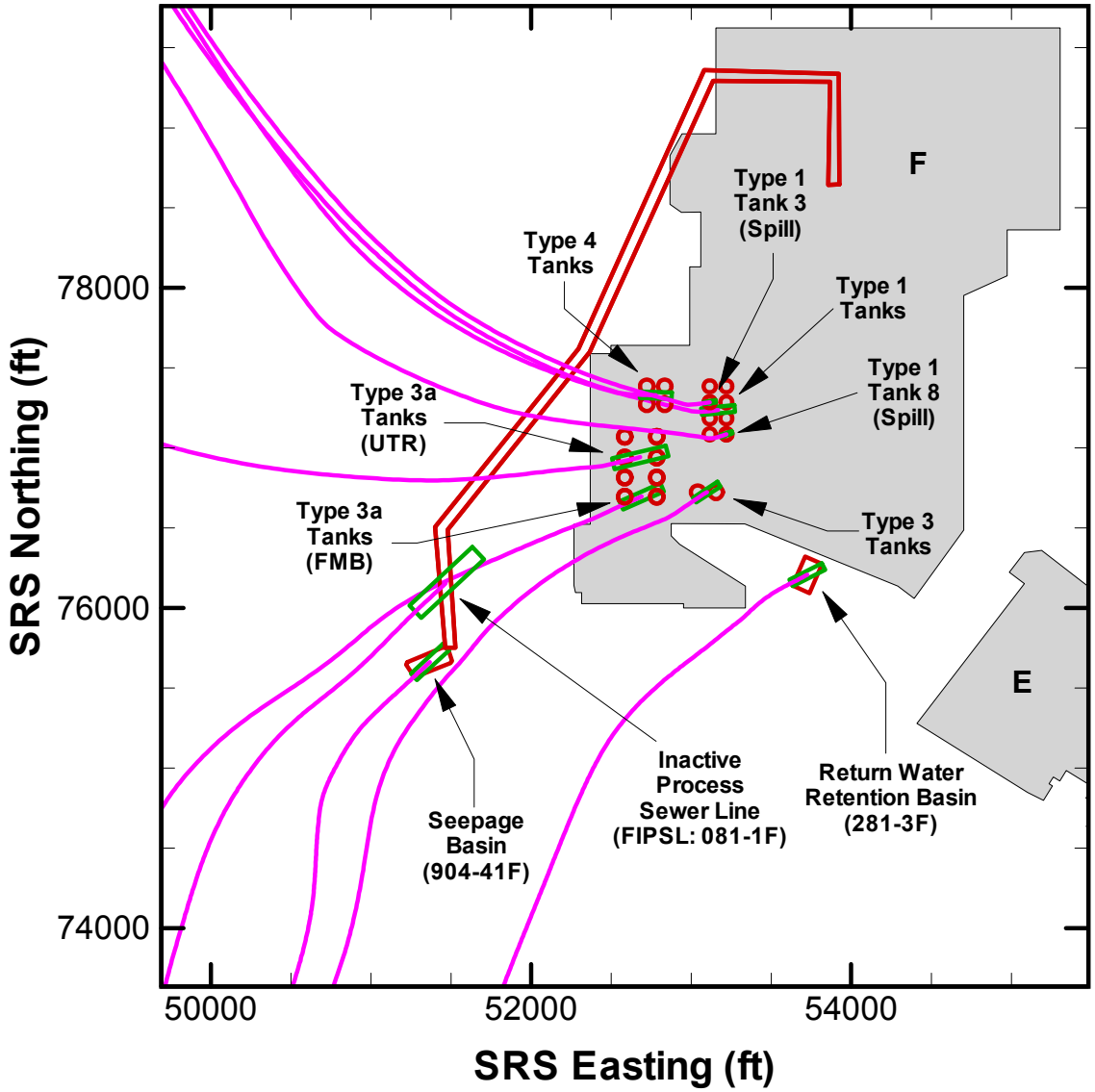


Figure D-18. Second Close-Up View of the Actual Footprints of the CA Units within F-Area

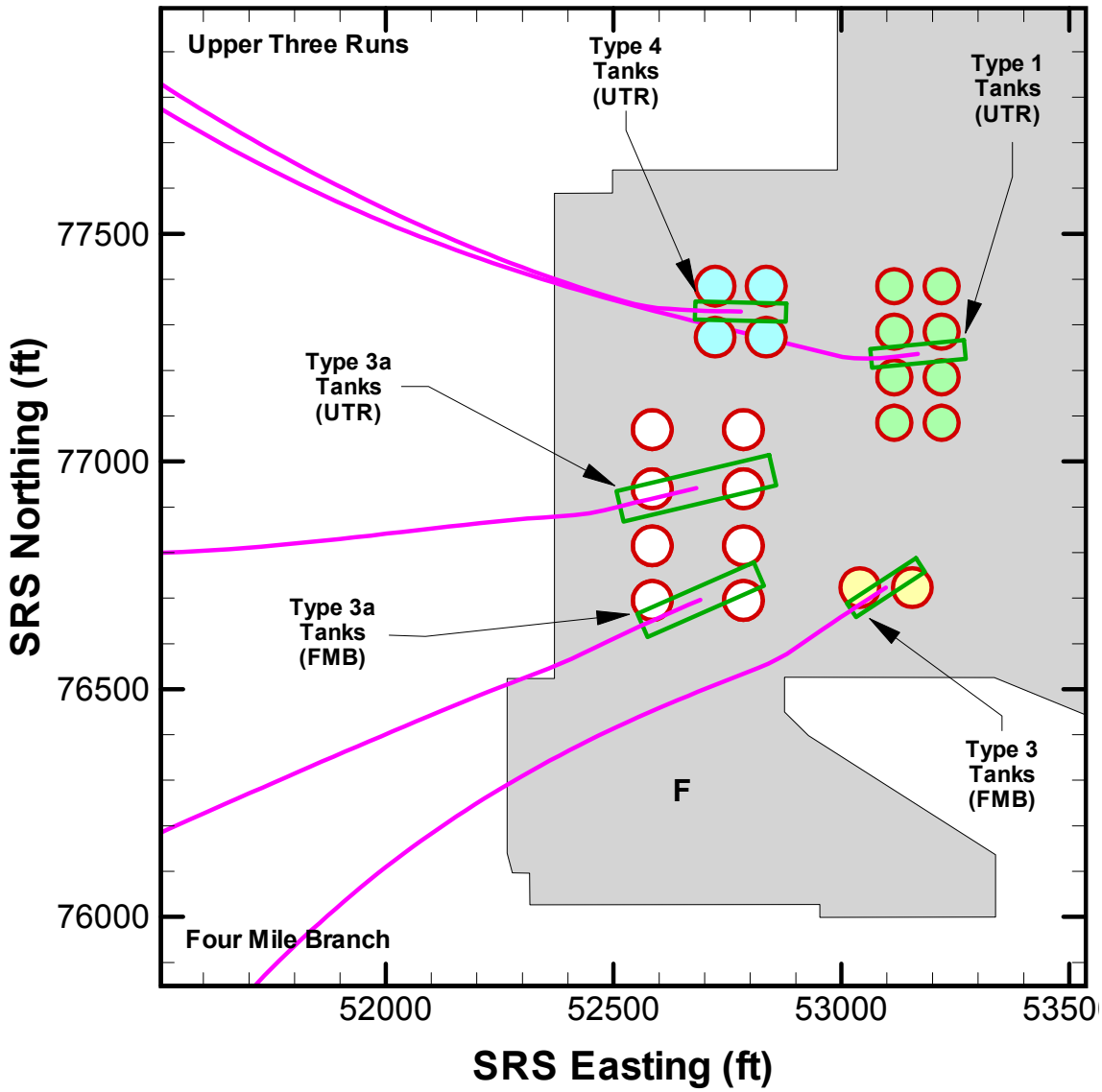


Figure D-19. Third Close-Up View of the Actual Footprints of the CA Tank Units within F-Area

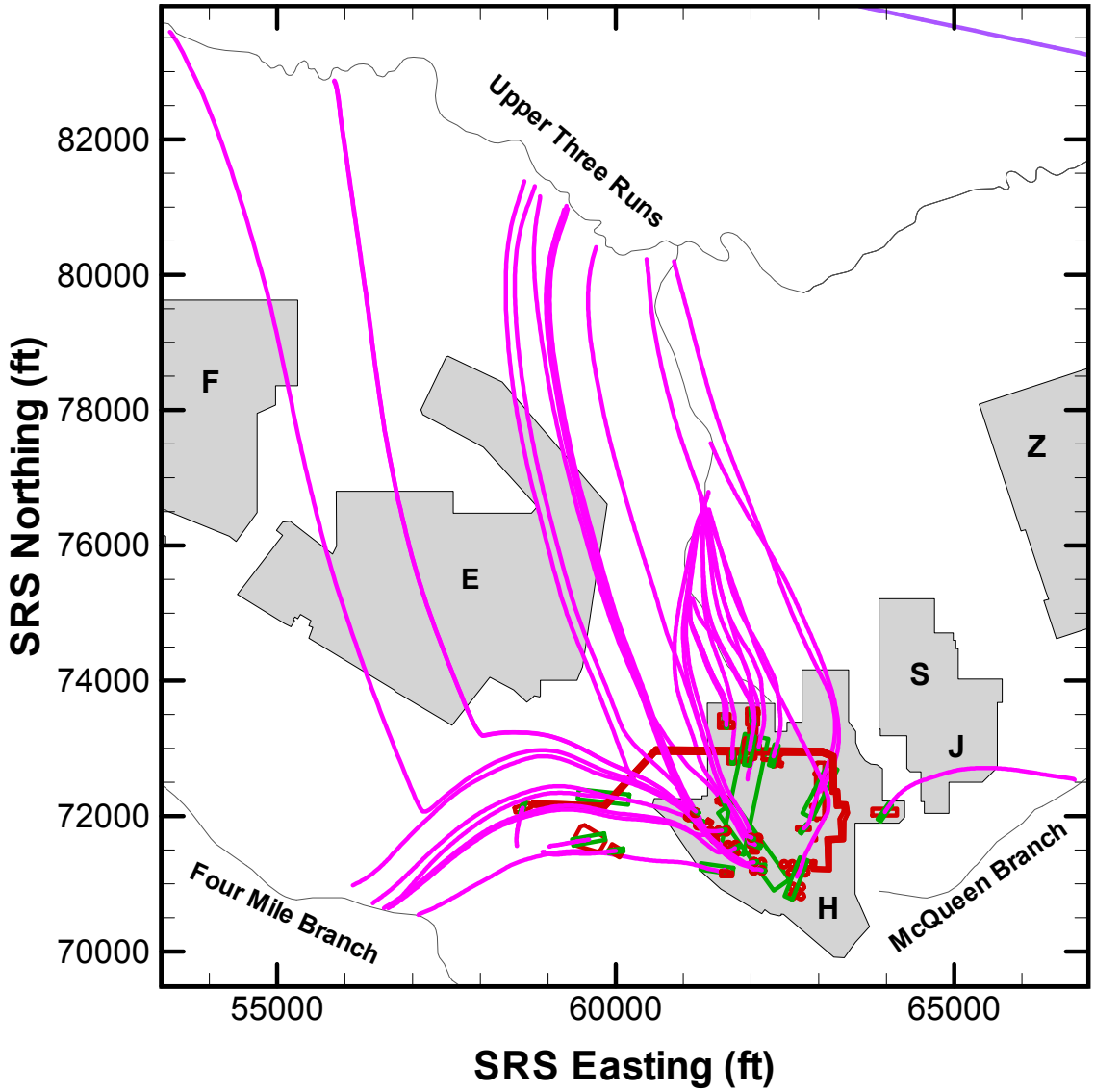


Figure D-20. Close-Up view of the H-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

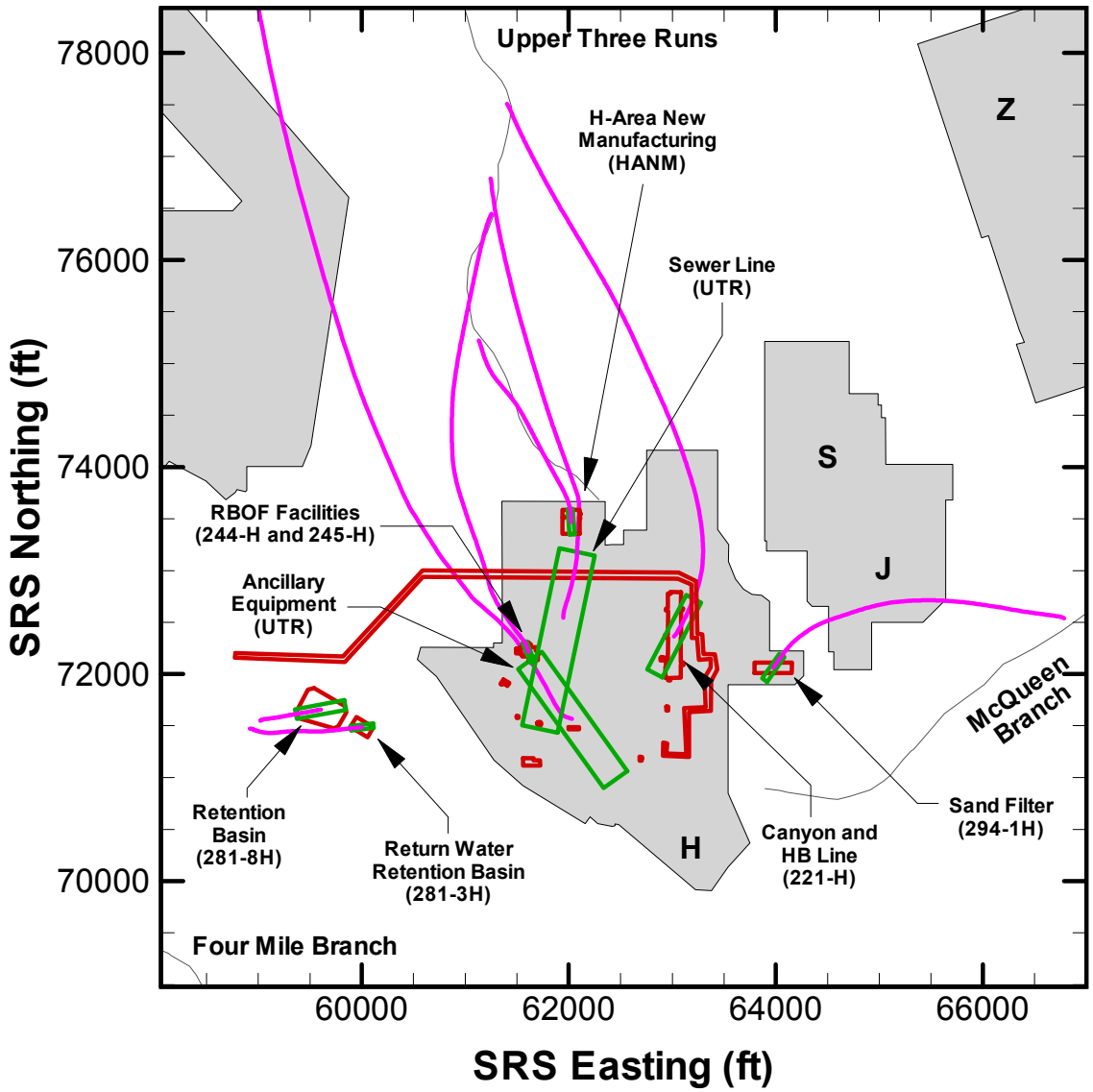


Figure D-21. First Close-Up View of the Actual Footprints of the CA Units within H-Area

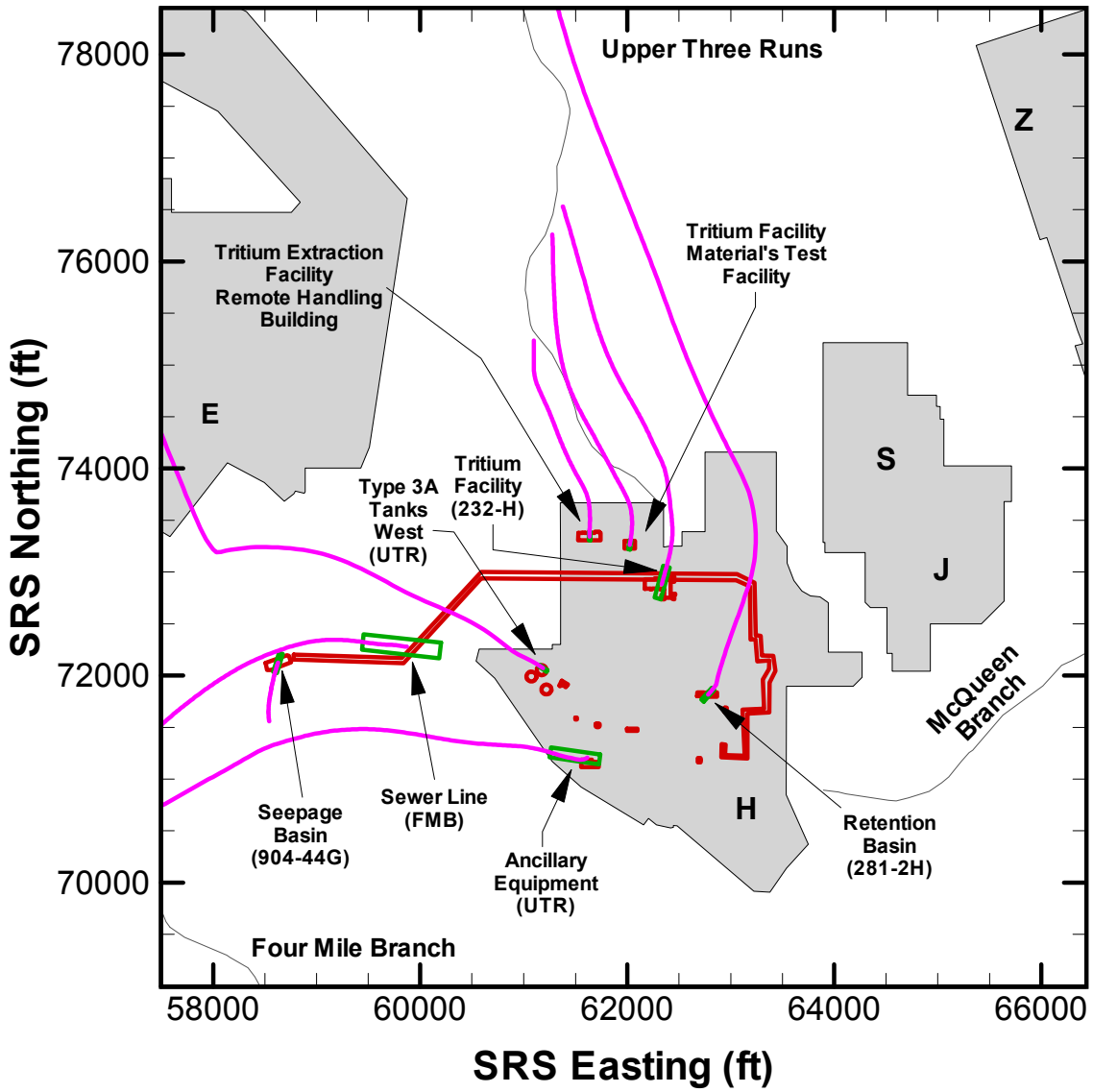


Figure D-22. Second Close-Up View of the Actual Footprints of the CA Units within H-Area

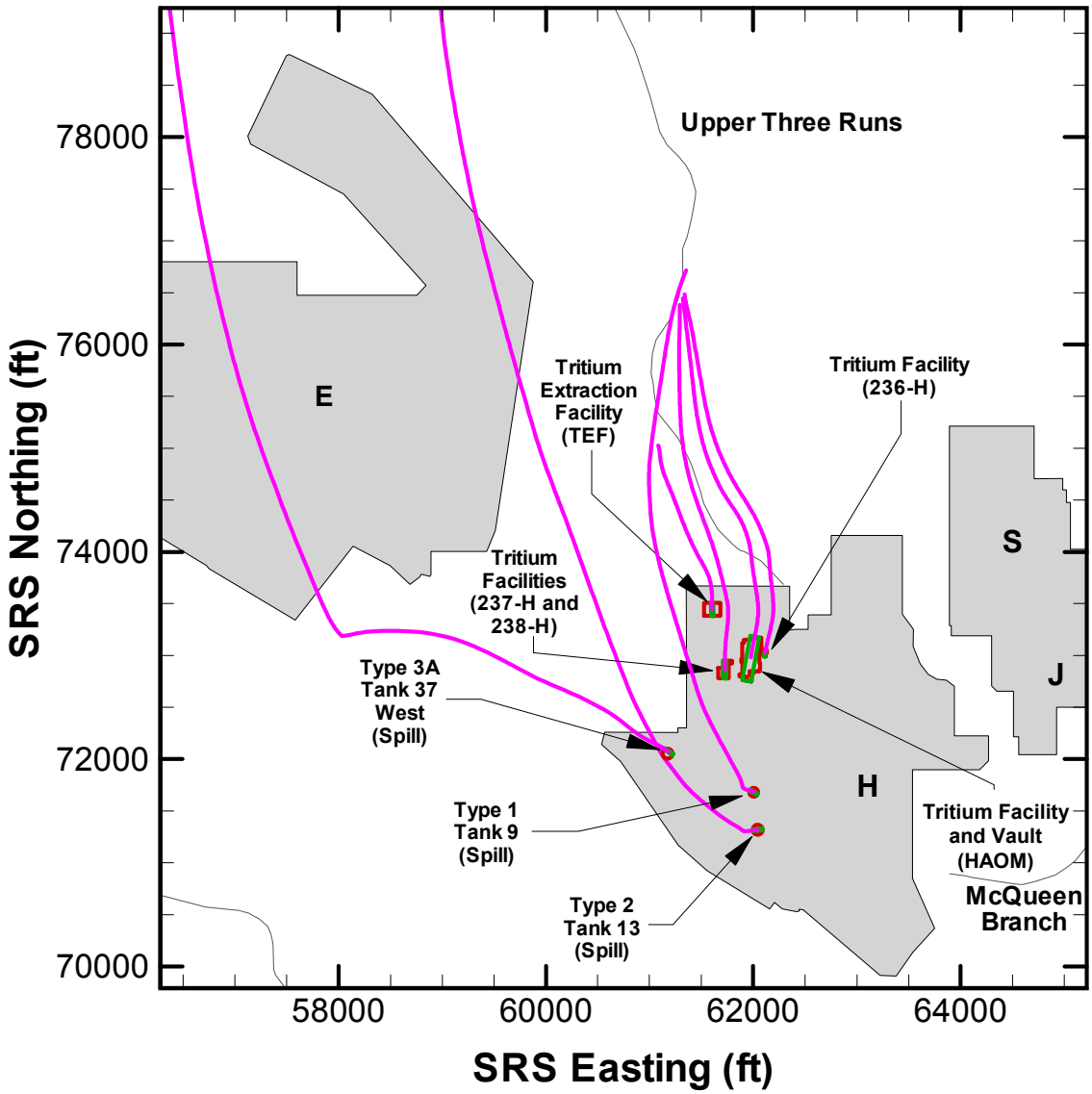


Figure D-23. Third Close-Up View of the Actual Footprints of the CA Units within H-Area

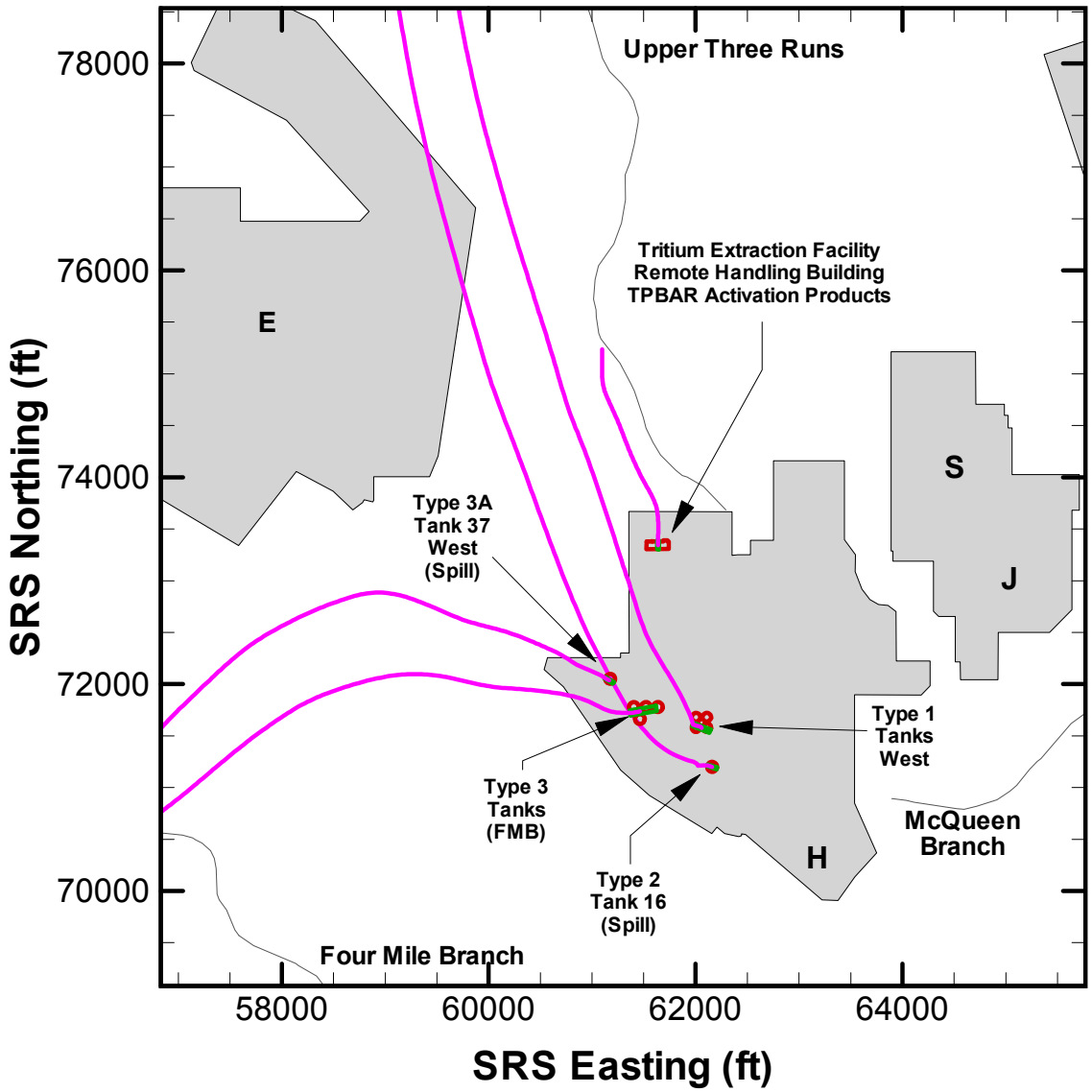


Figure D-24. Fourth Close-Up View of the Actual Footprints of the CA Units within H-Area

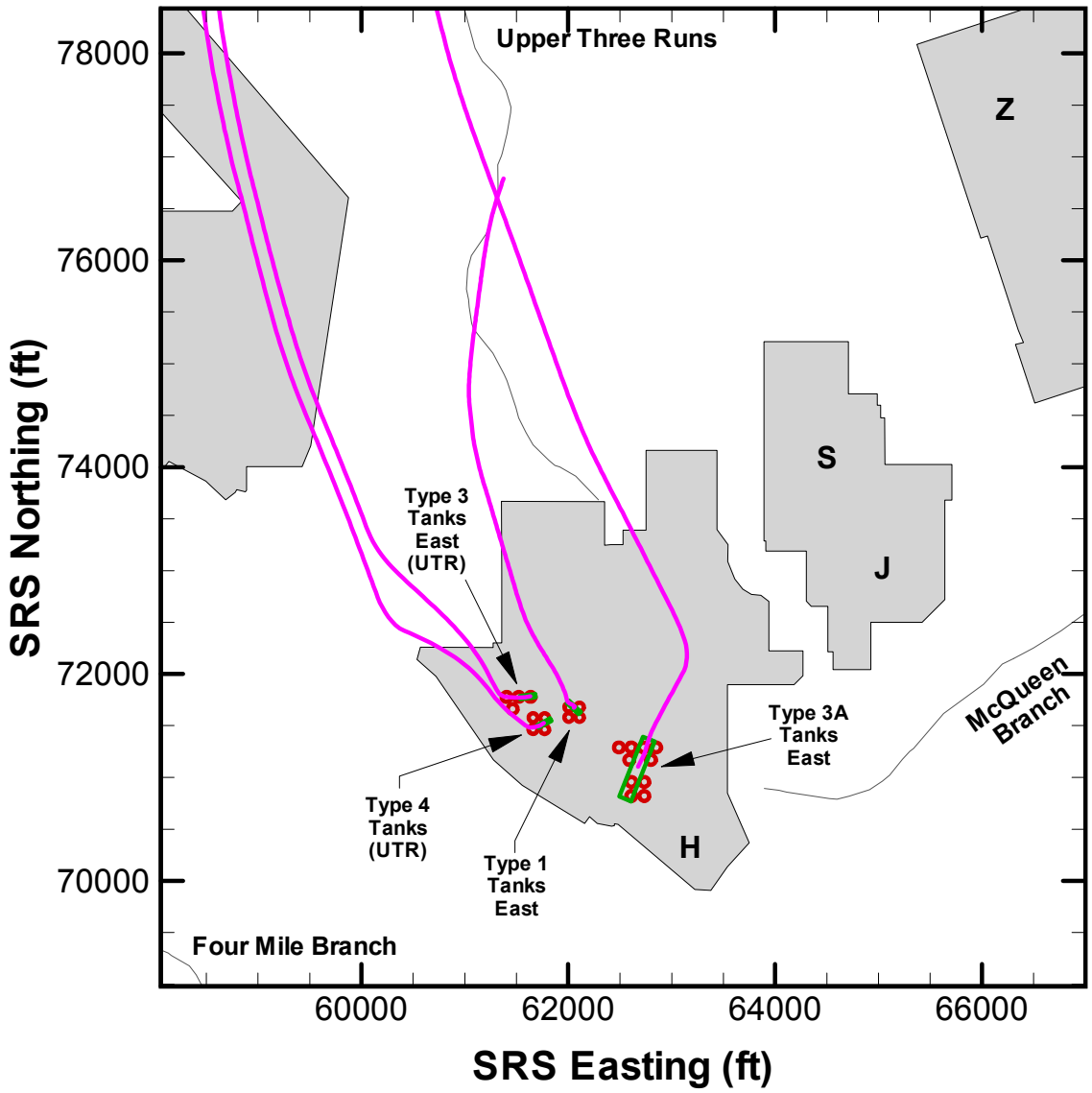


Figure D-25. Fifth Close-Up View of the Actual Footprints of the CA Units within H-Area

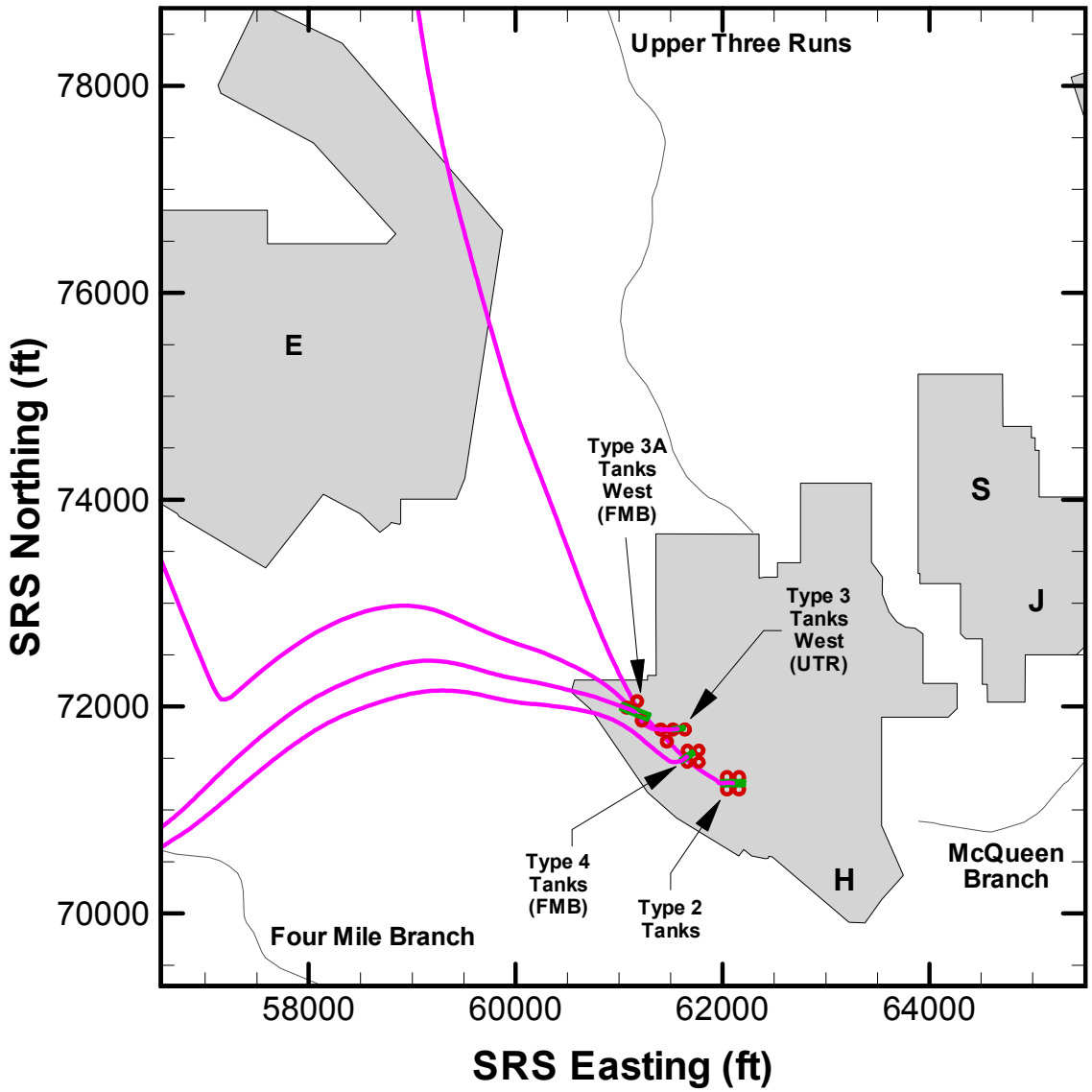


Figure D-26. Sixth Close-Up View of the Actual Footprints of the CA Units within H-Area

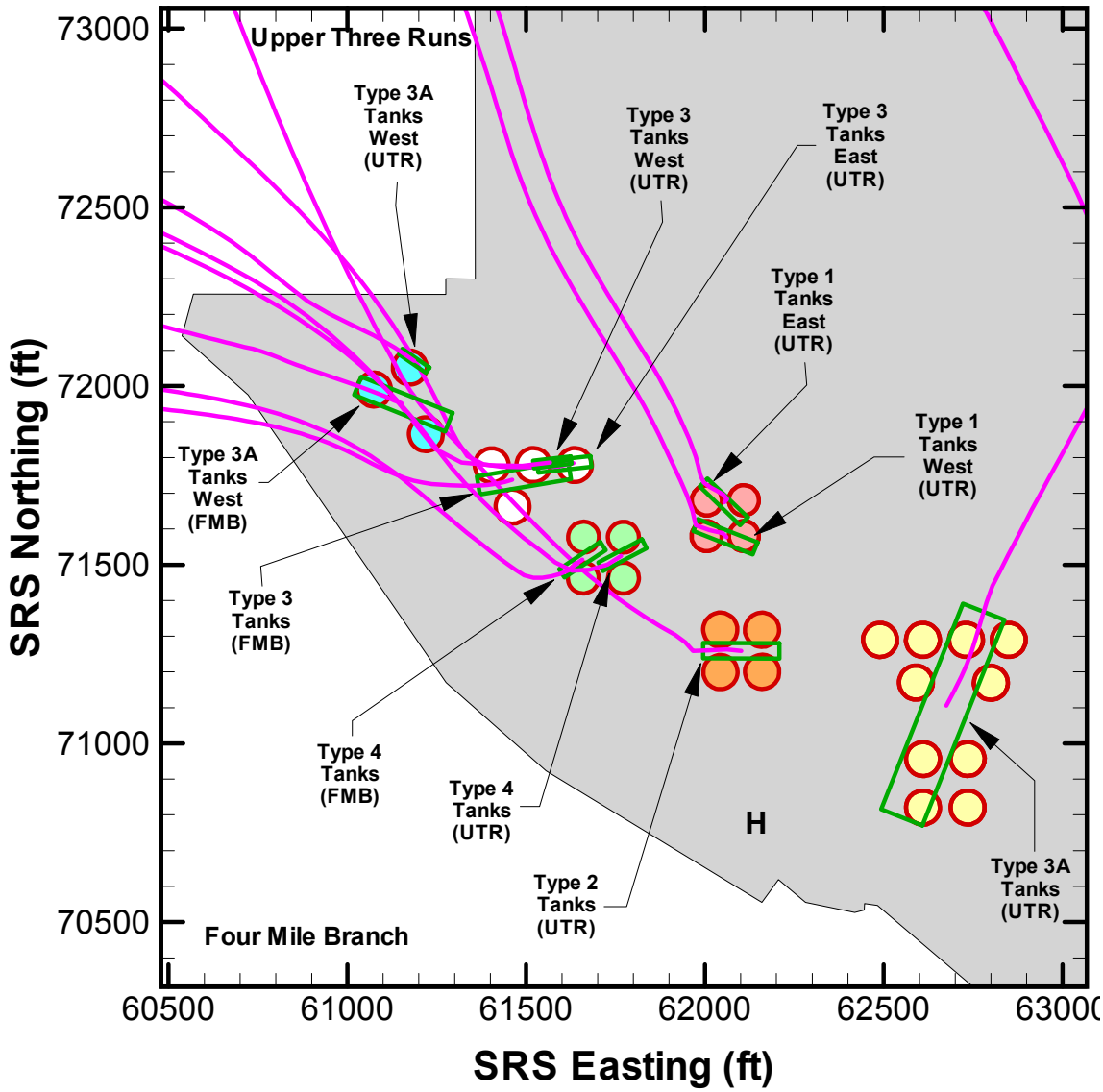


Figure D-27. Seventh Close-Up View of the Actual Footprints of the CA Tank Units within H-Area Tank Farm

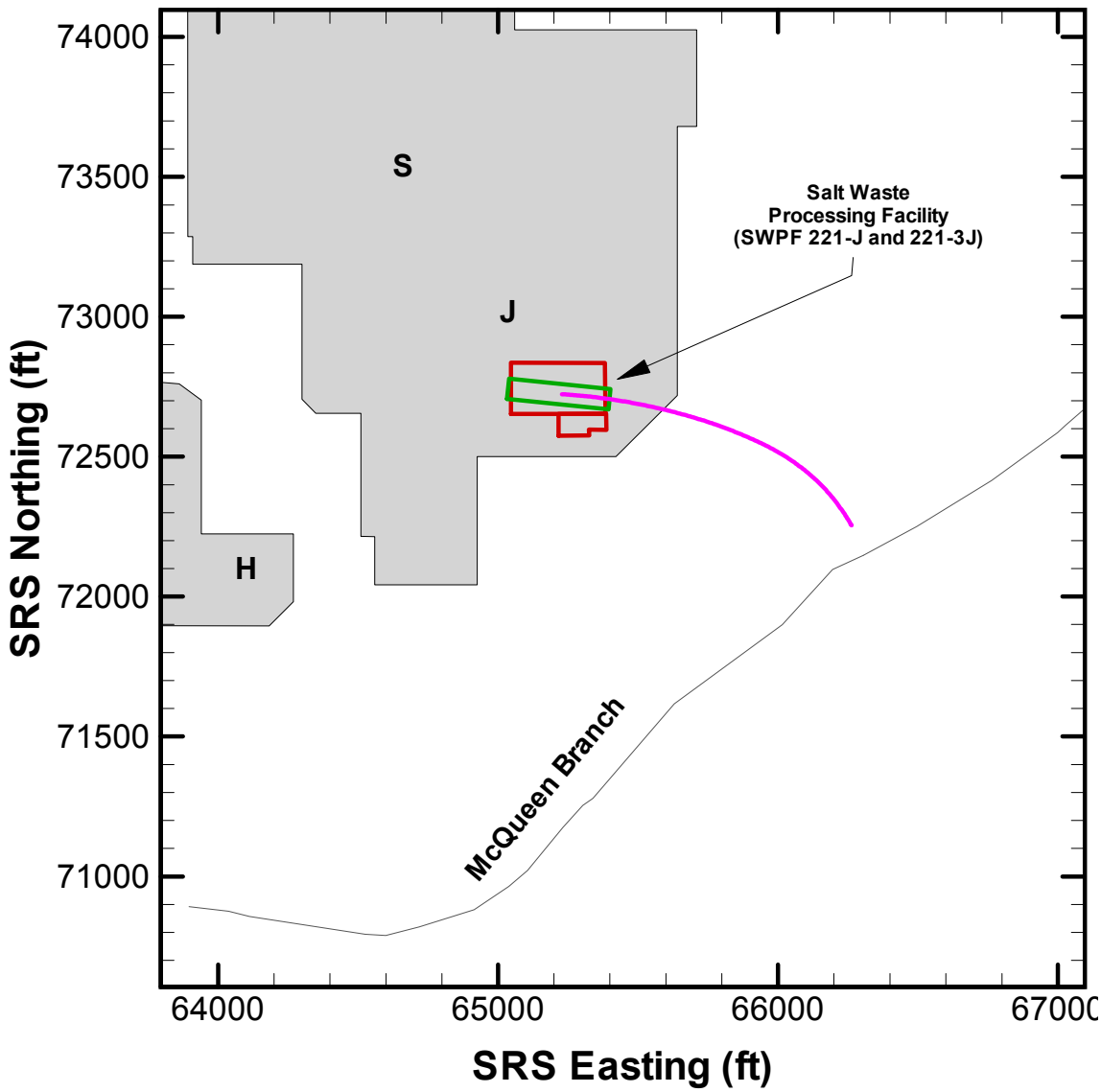


Figure D-28. Close-Up View of the J-Area CA Unit Showing its Footprint and Average Stream Trace

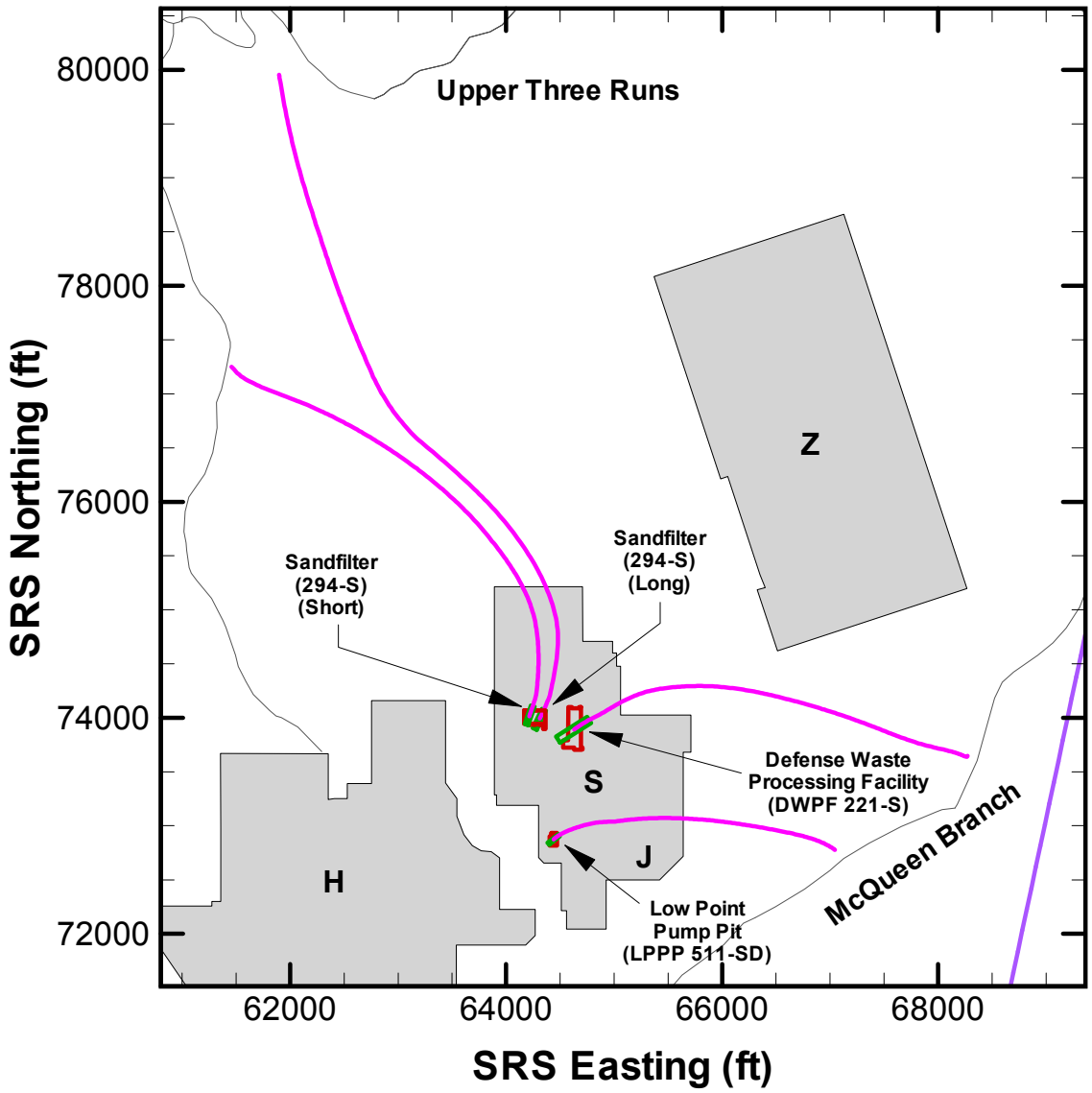


Figure D-29. Close-Up View of the S-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

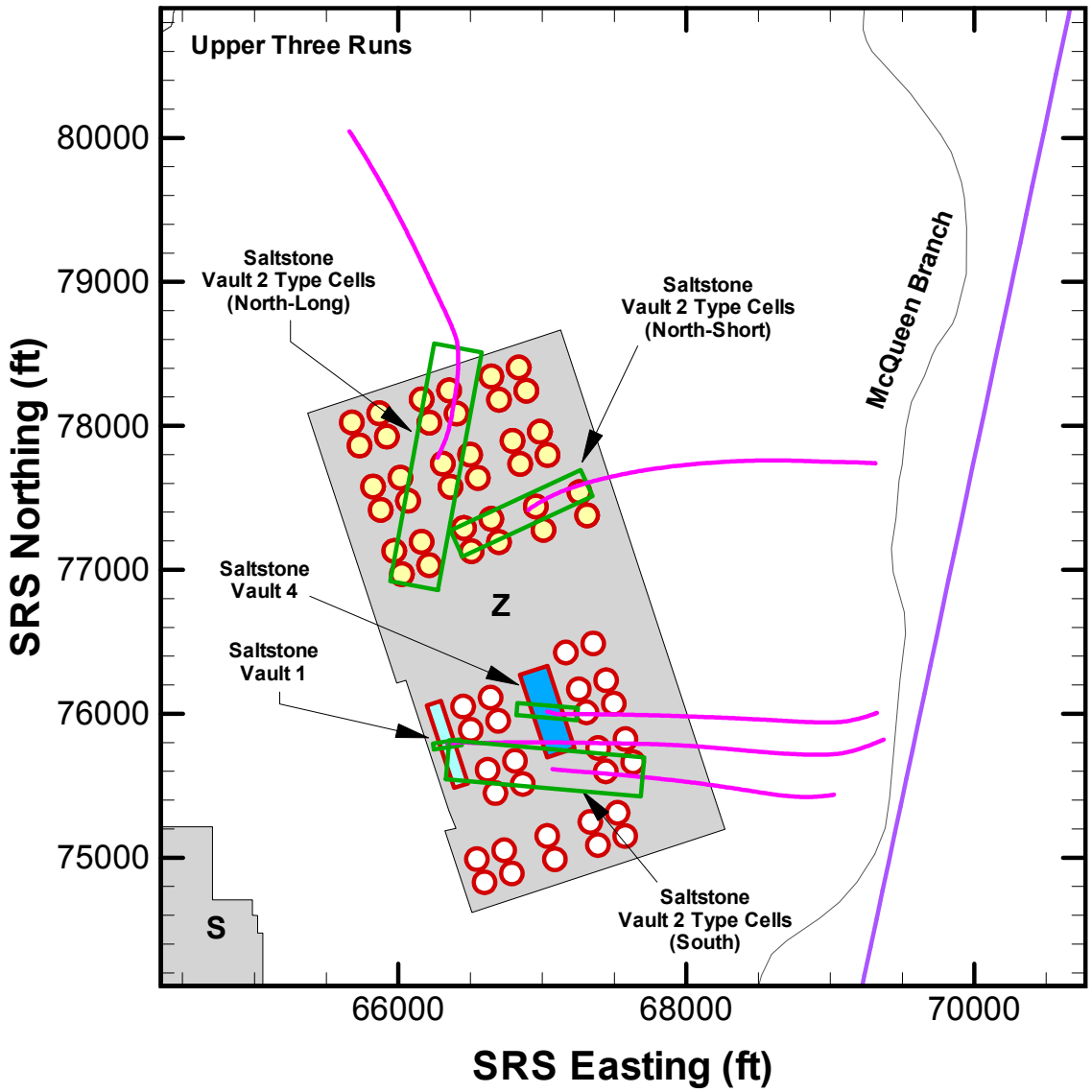
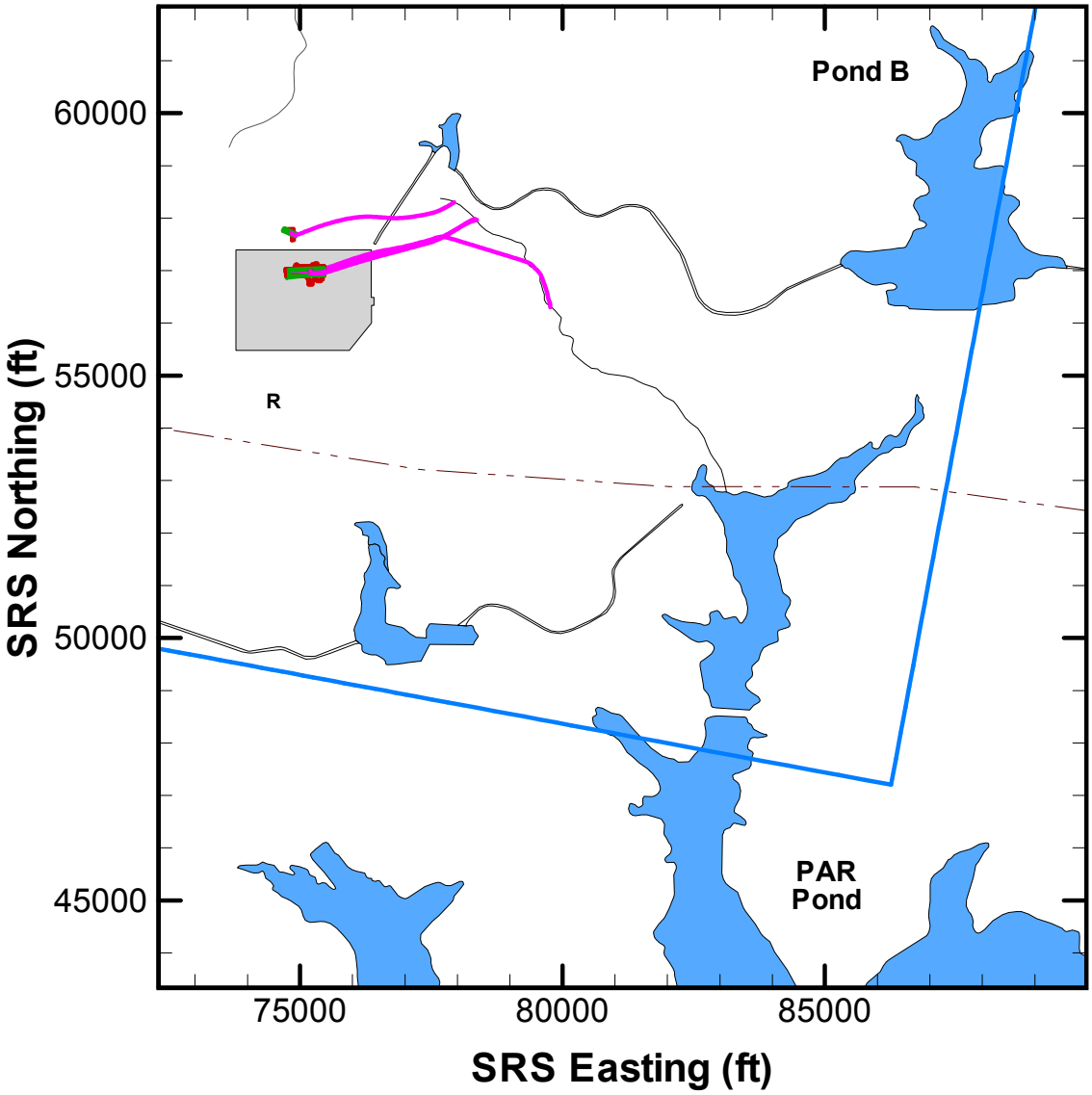


Figure D-30. Close-Up View of the Z-Area CA Units Showing Each Unit's Footprint and Average Stream Trace



Each unit's average streamtrace from its footprint to its corresponding discharge point is shown.

Figure D-31. Close-Up View of Those CA Units within the R Flow Model Domain

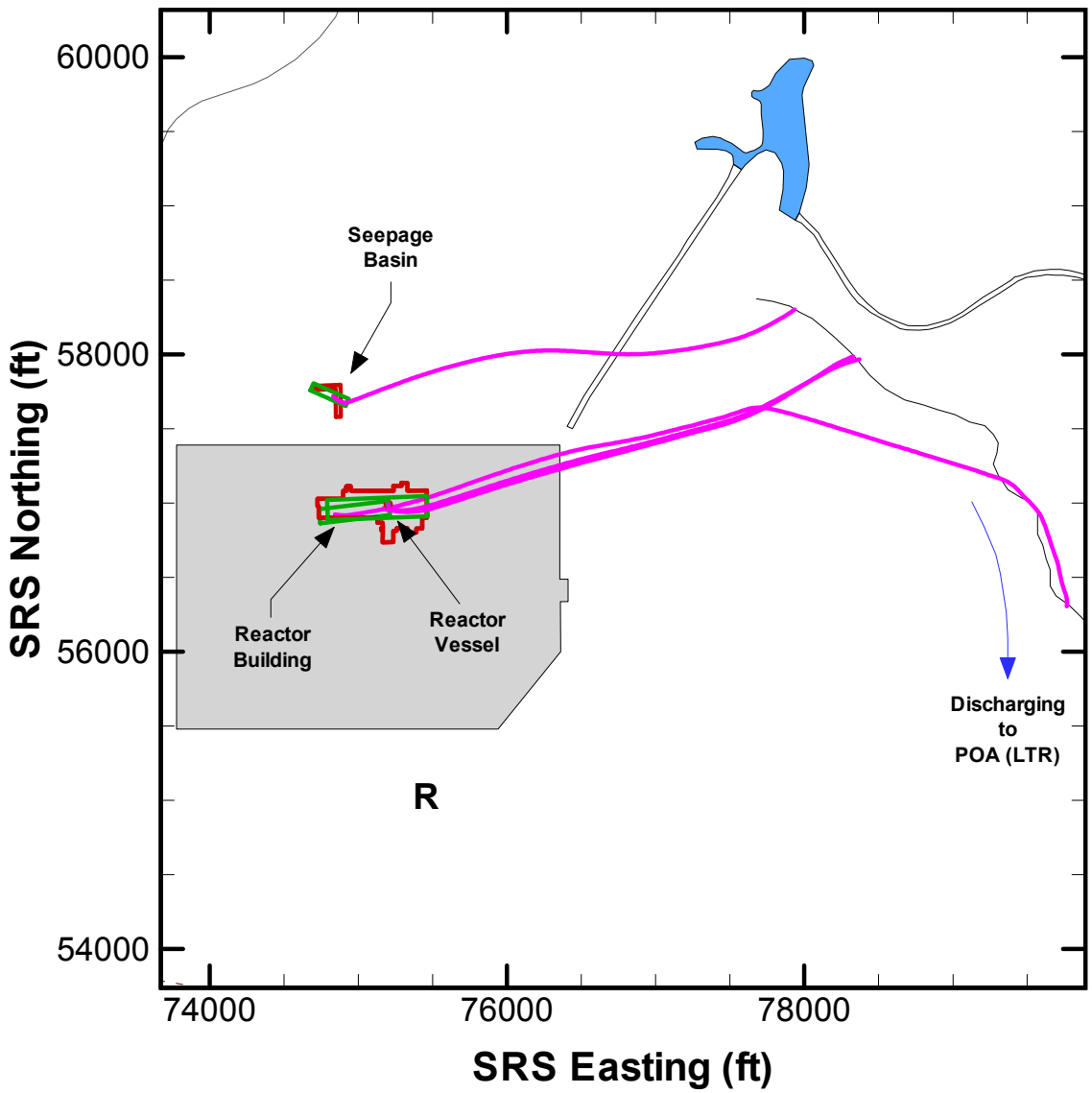


Figure D-32. Close-Up View of the R-Area CA Units Showing Each Unit's Footprint and Average Stream Trace

D.3 REFERENCES

Hamm, L. L., Smith, F. G. III, Phifer, M. A., and Collard, L. B. 2009. *Savannah River Site Composite Analysis: Aquifer Flow Path Parameters*, SRNL-STI-2009-00438, Rev. 0, Savannah River National Laboratory, Aiken, SC, August 2009. [OUO]

E. APPENDIX E: MODELING RESULTS FIGURES

E.1 DETERMINISTIC OUTPUT RESULTS (FROM SMITH ET AL. 2009A)

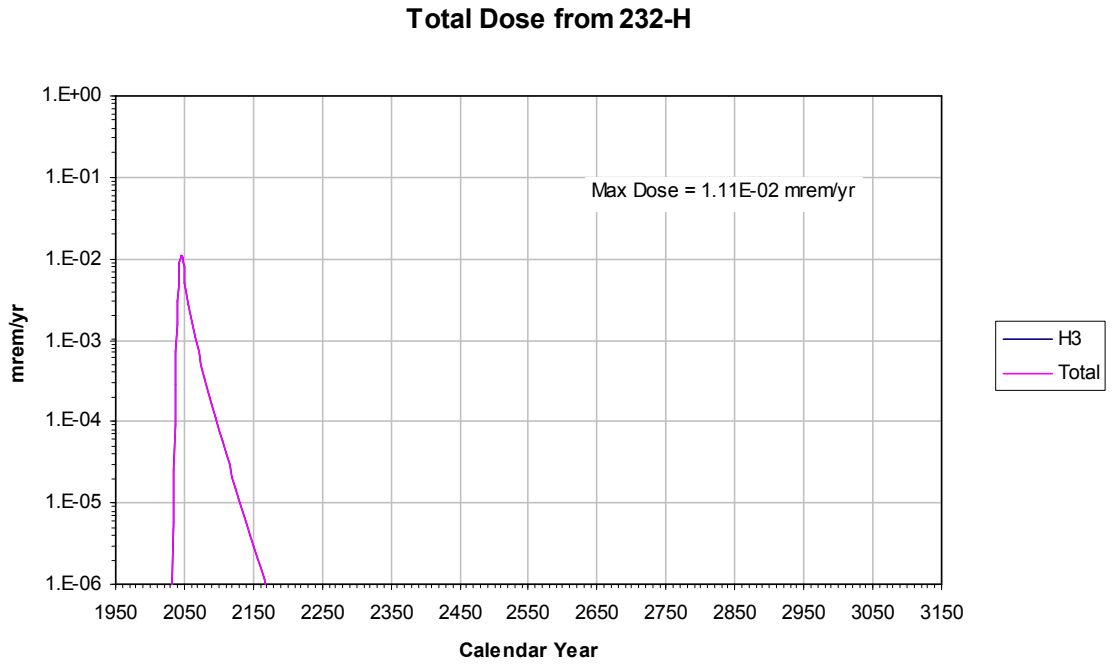


Figure E-1. 232-H Direct Dose by Radionuclide

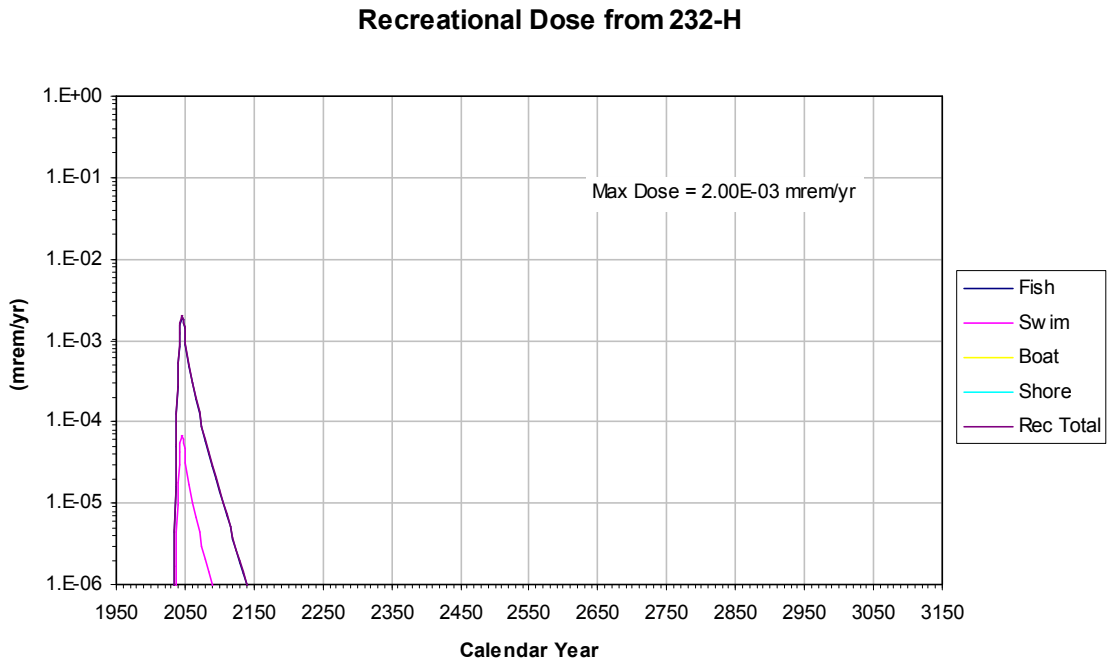


Figure E-2. 232-H Recreational Dose

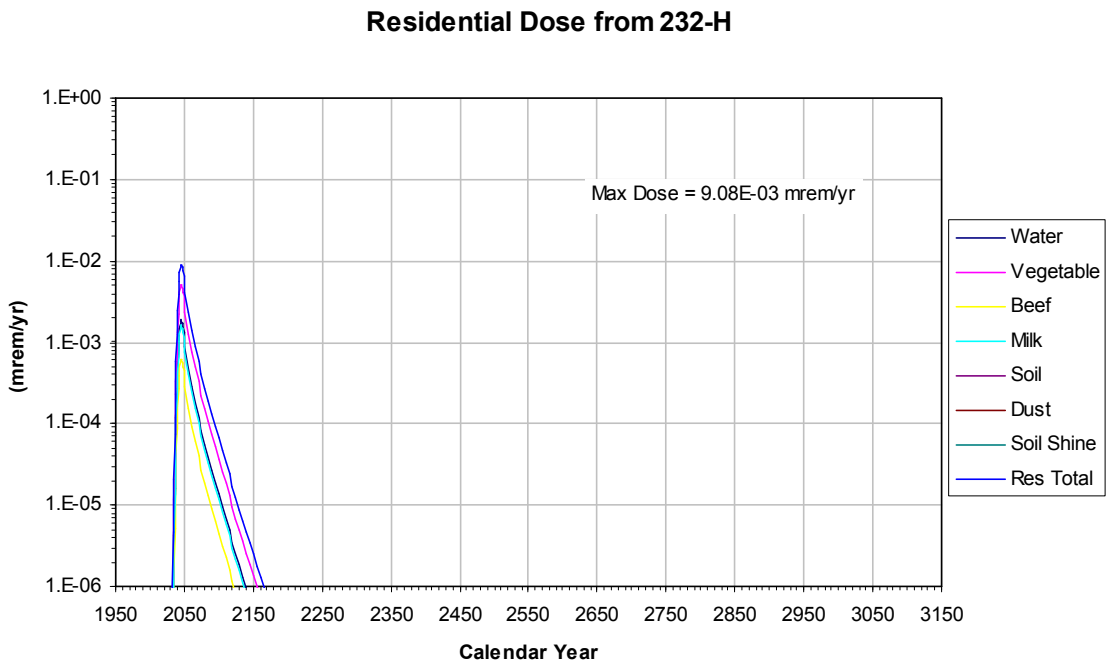


Figure E-3. 232-H Direct Residential Dose

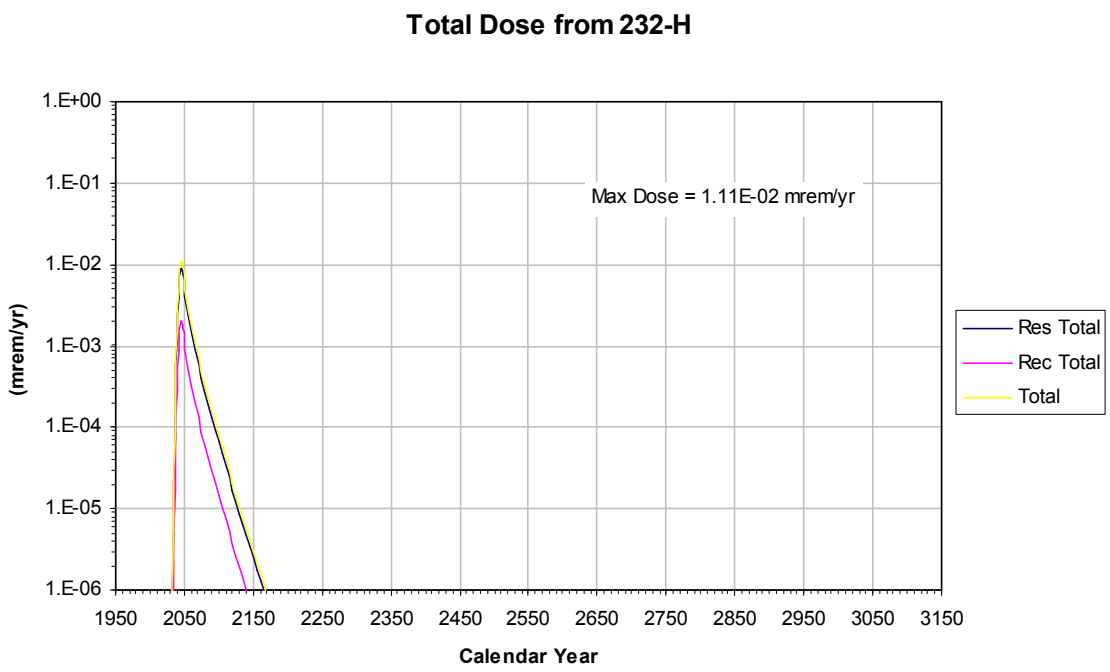


Figure E-4. 232-H Total Direct Dose

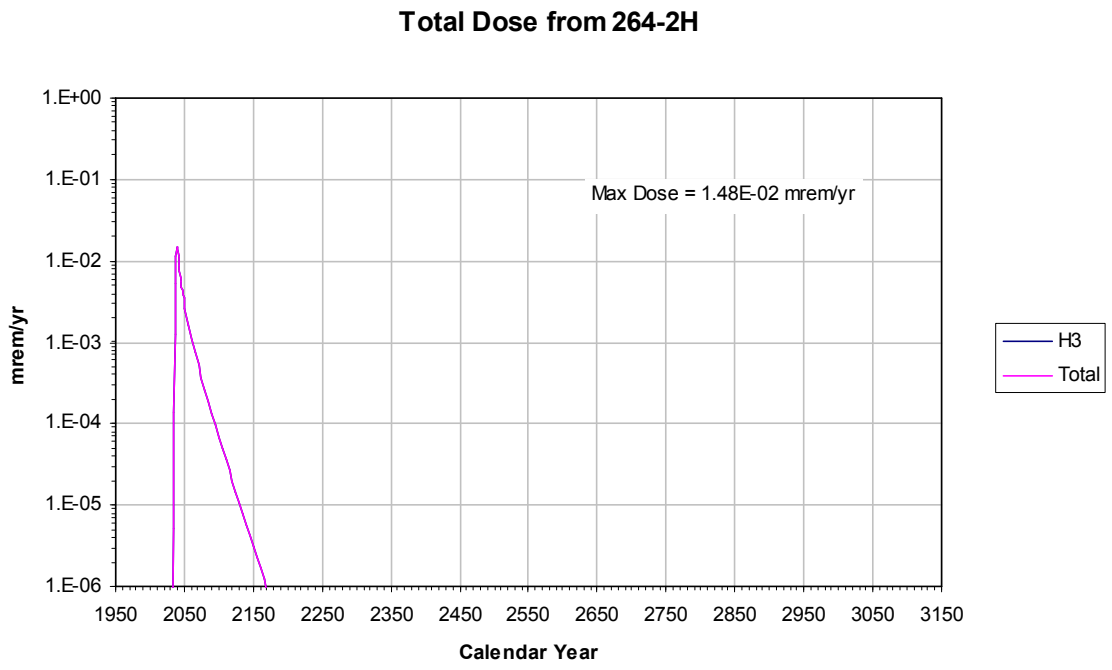


Figure E-5. 264-2H Direct Dose by Radionuclide

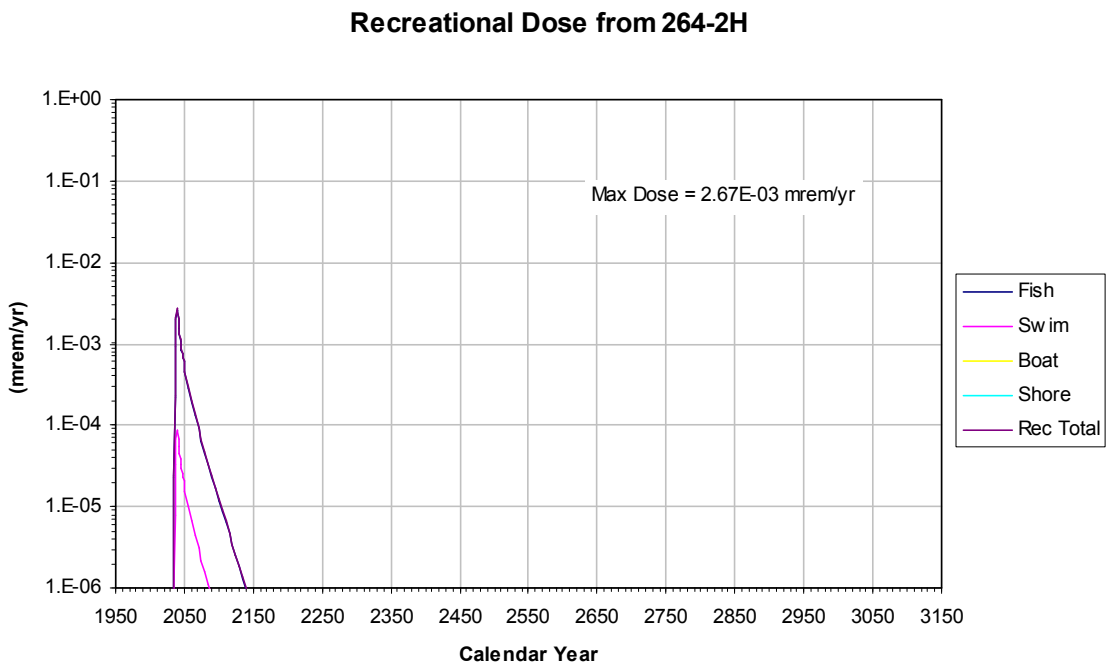


Figure E-6. 264-2H Recreational Dose

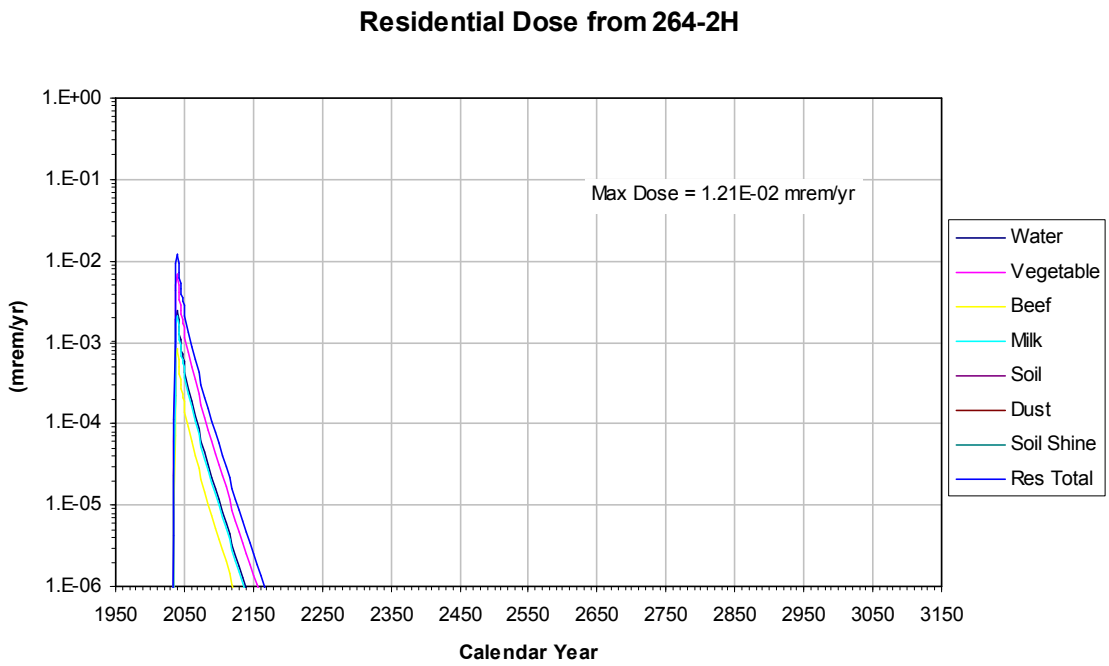


Figure E-7. 264-2H Direct Residential Dose

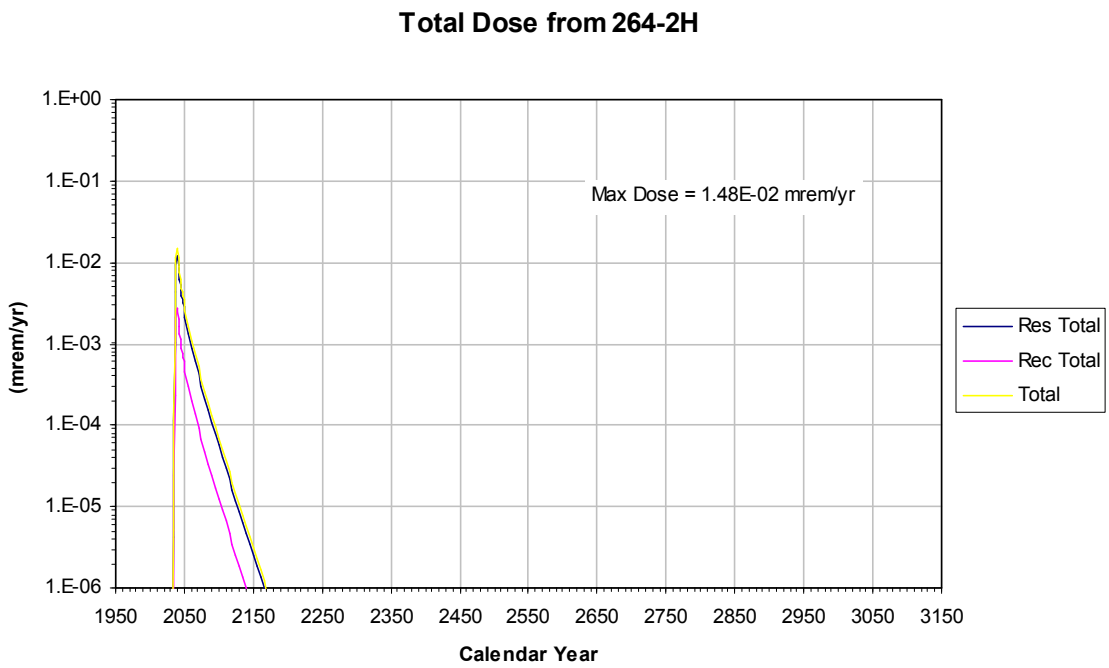


Figure E-8. 264-2H Total Direct Dose

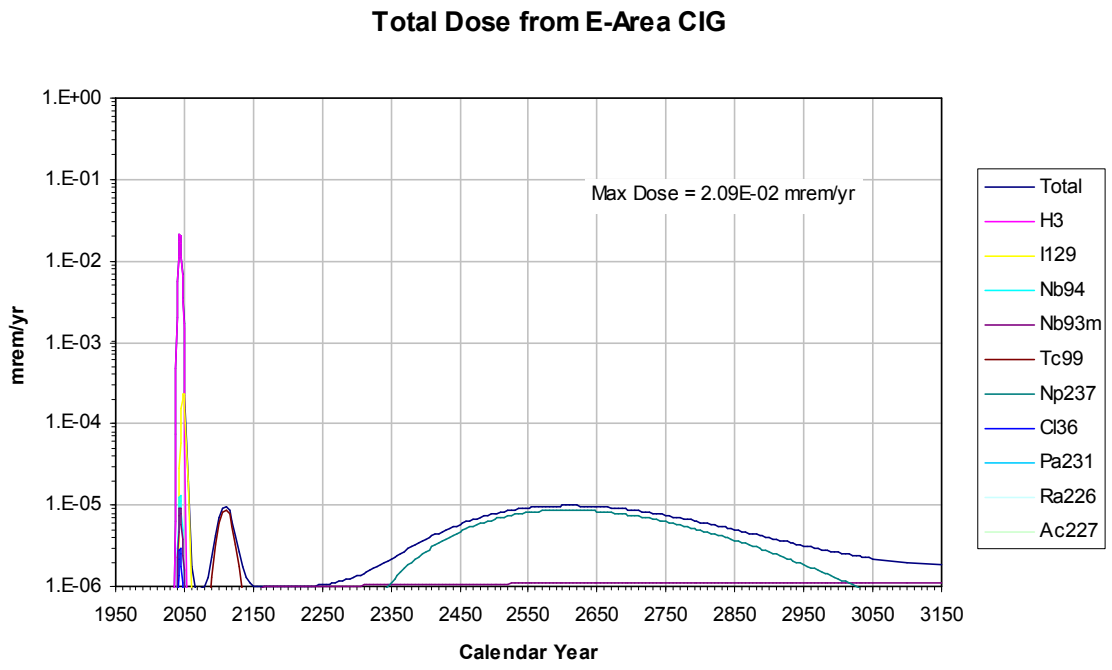


Figure E-9. E-Area CIG Direct Dose by Radionuclide

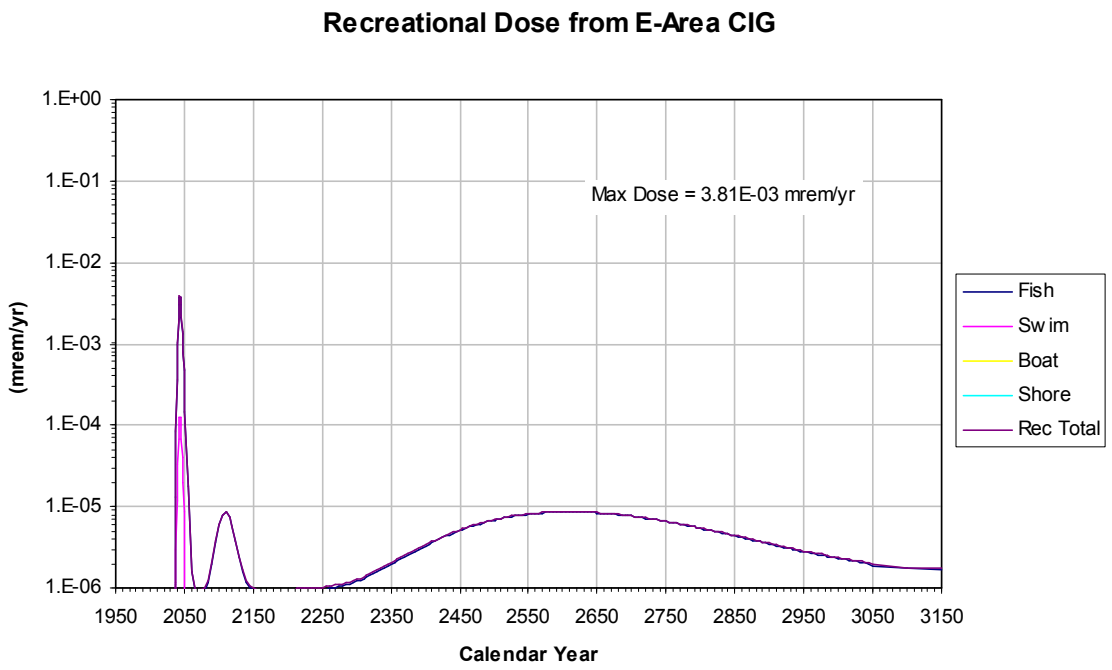


Figure E-10. E-Area CIG Recreational Dose

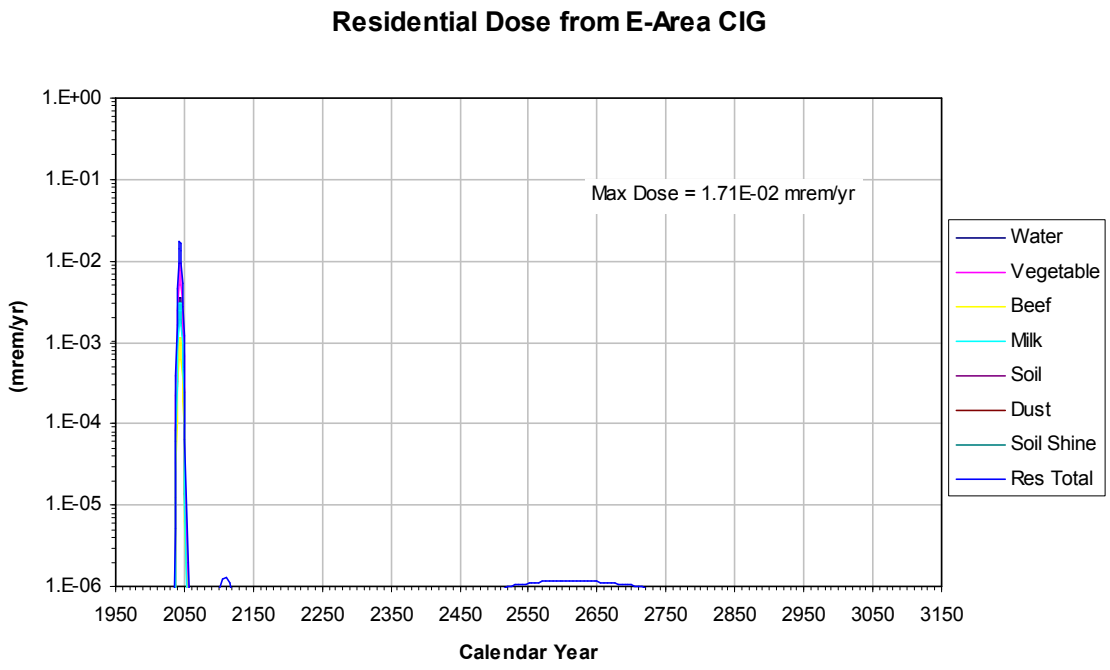


Figure E-11. E-Area CIG Direct Residential Dose

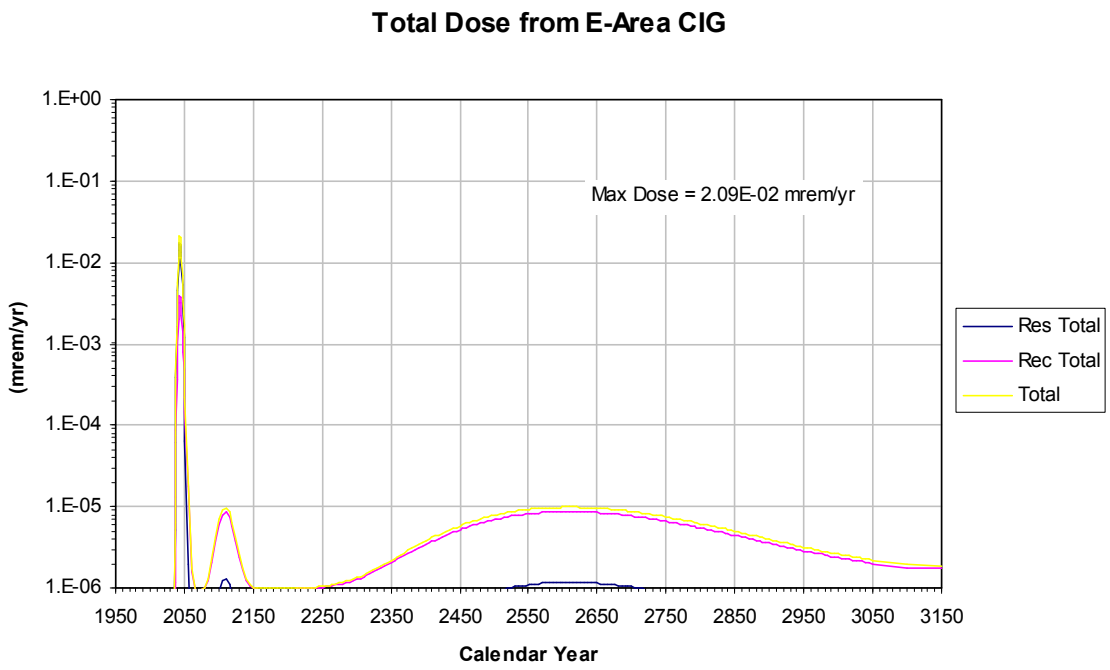


Figure E-12. E-Area CIG Total Direct Dose

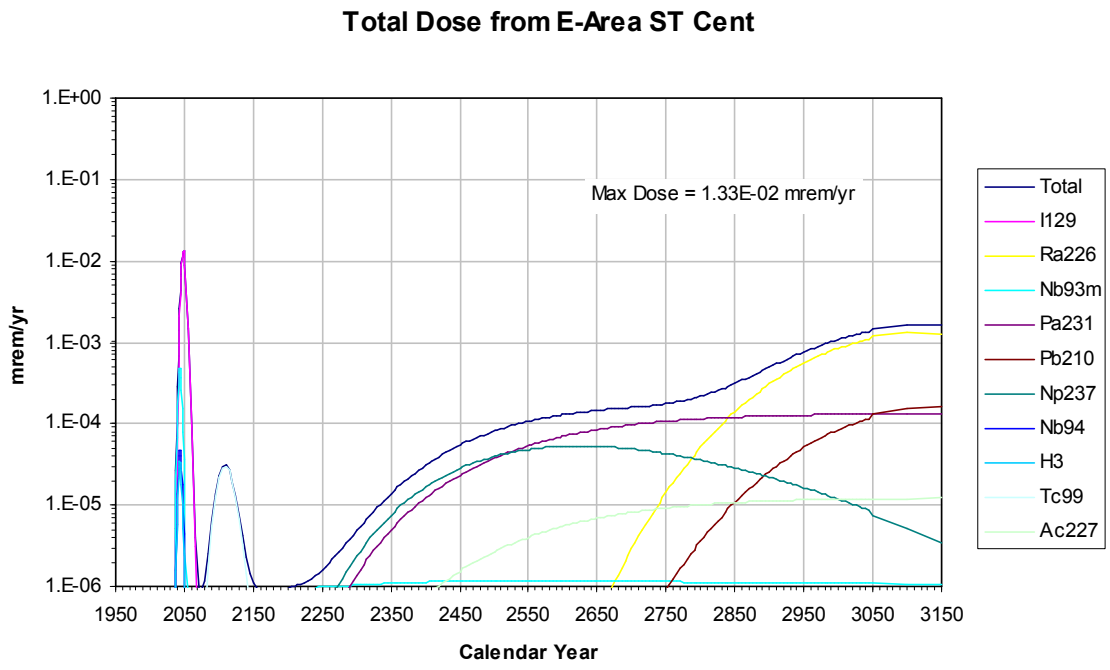


Figure E-13. E-Area ST Cent Direct Dose by Radionuclide

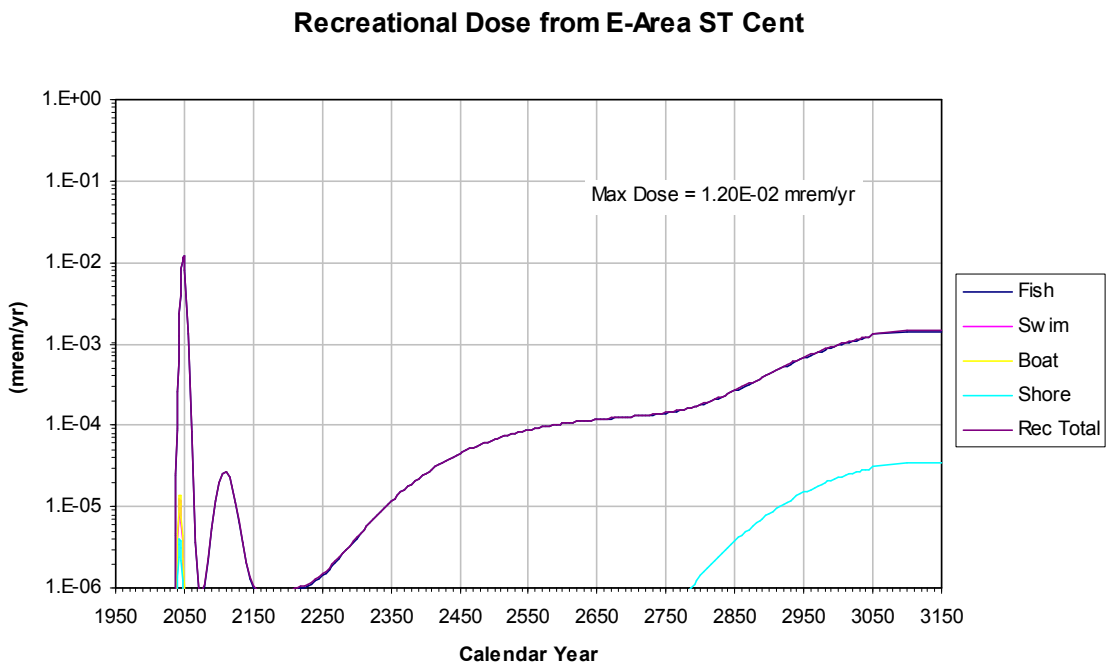


Figure E-14. E-Area ST Cent Recreational Dose

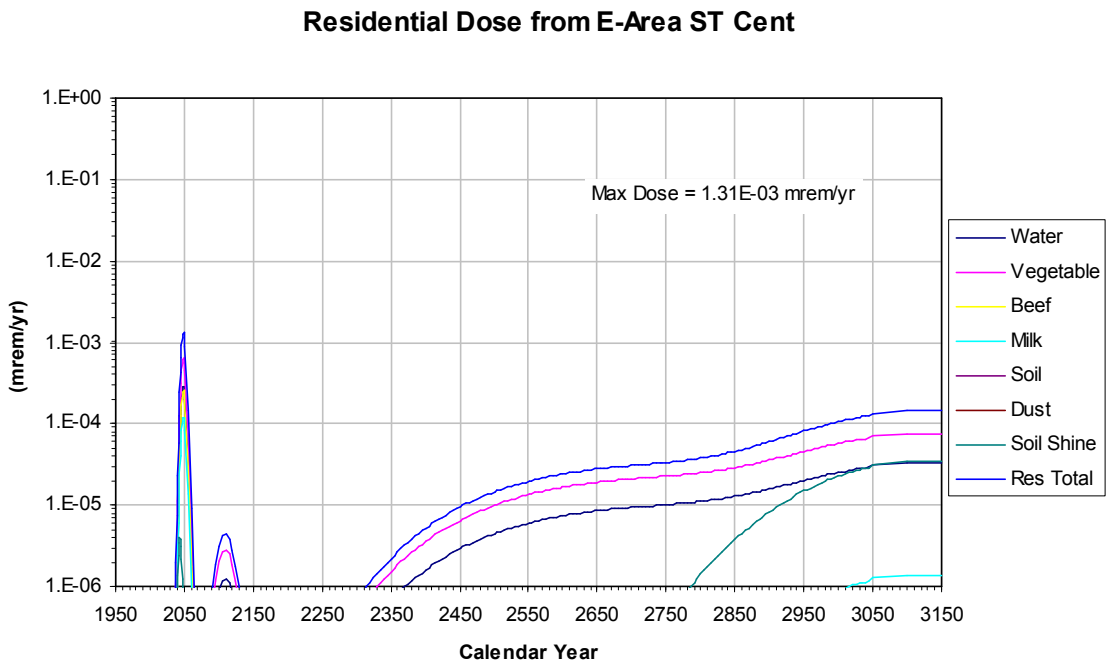


Figure E-15. E-Area ST Cent Direct Residential Dose

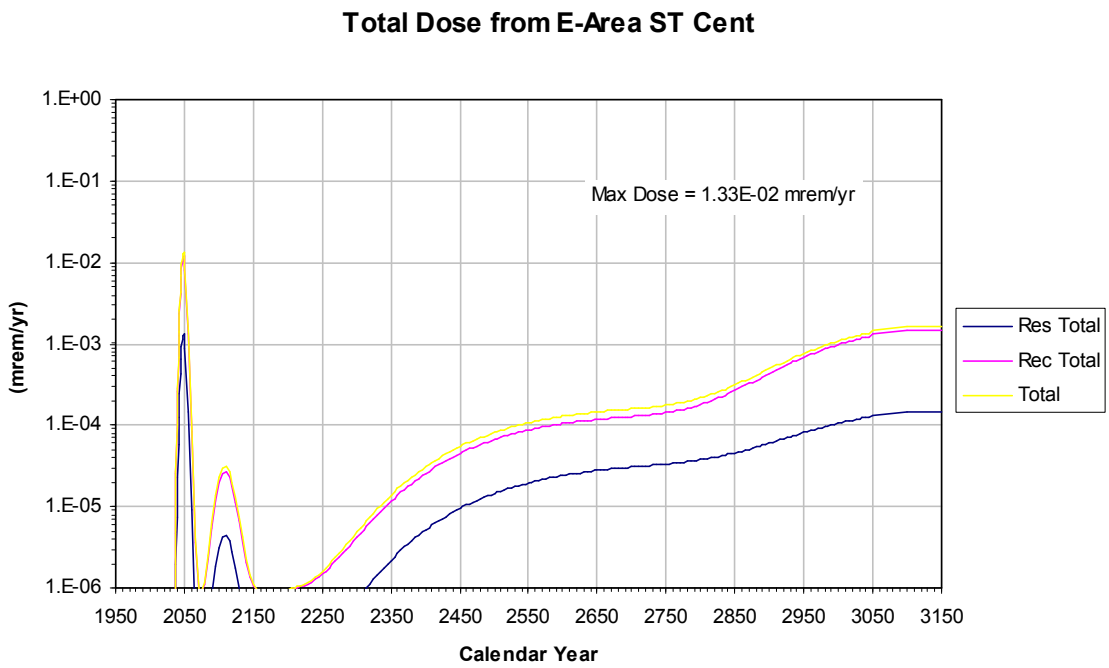


Figure E-16. E-Area ST Cent Total Direct Dose

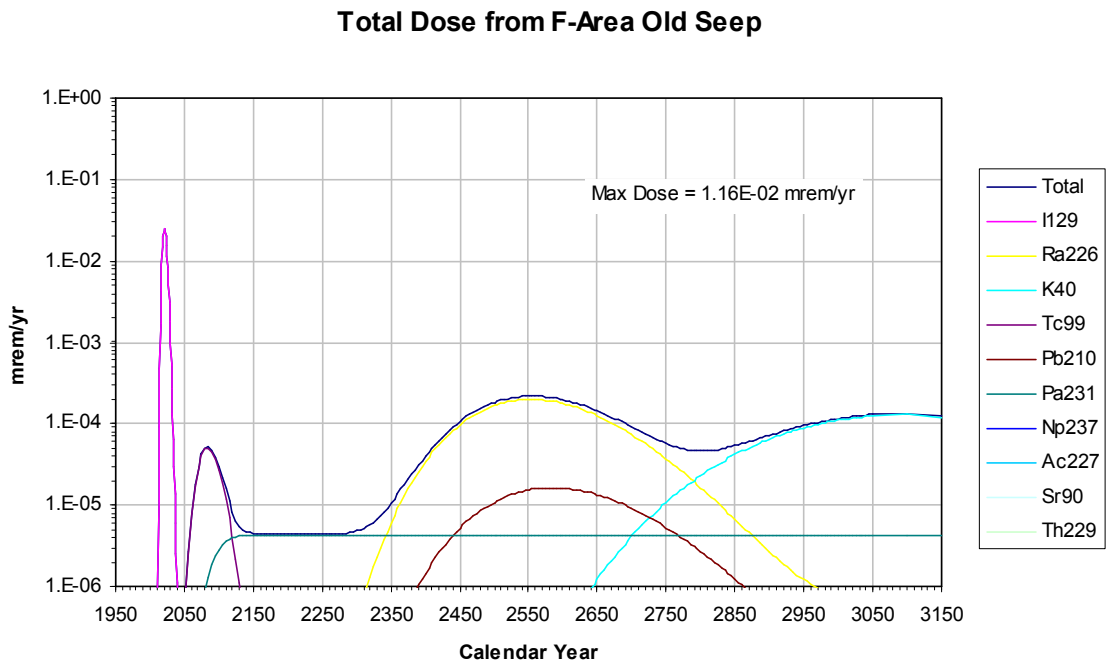


Figure E-17. F-Area Old Seep Direct Dose by Radionuclide

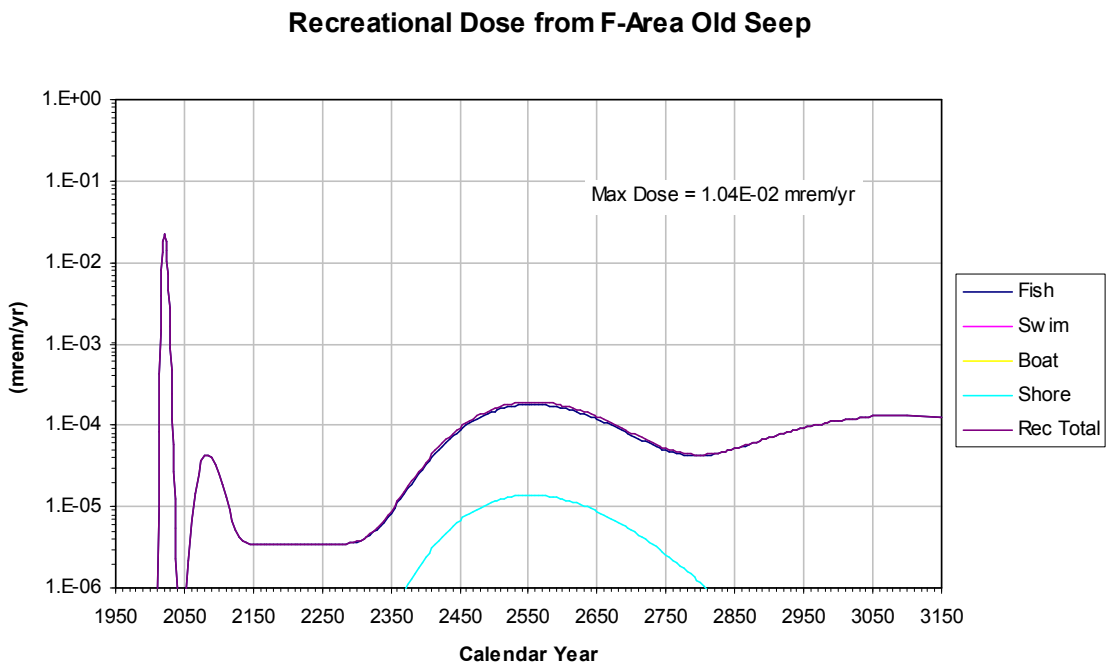


Figure E-18. F-Area Old Seep Recreational Dose

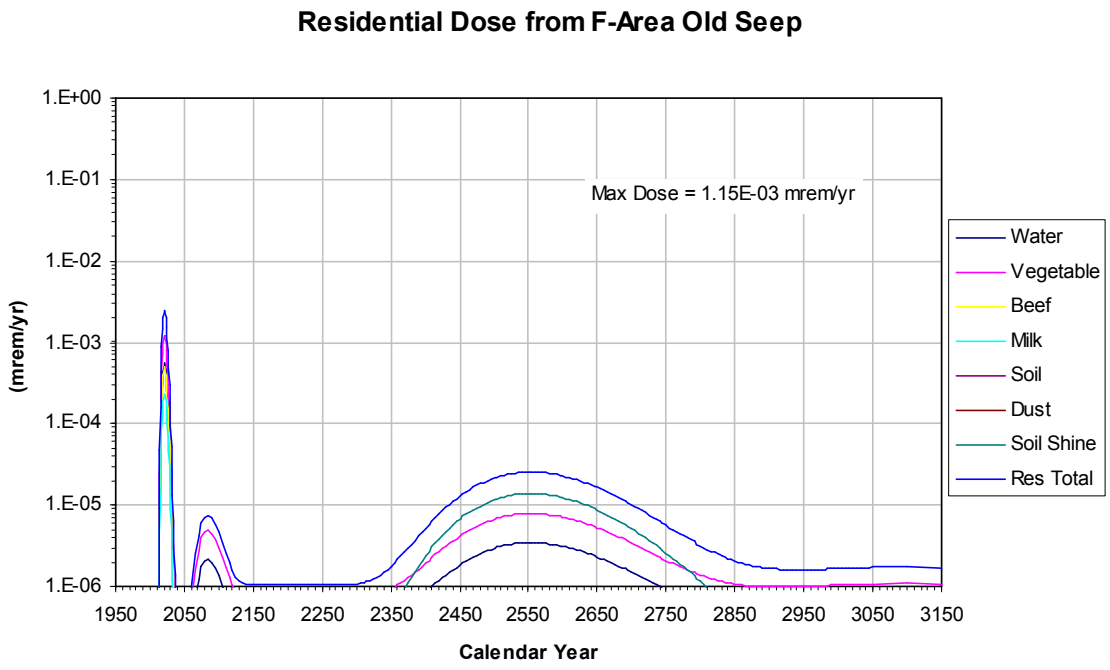


Figure E-19. F-Area Old Seep Direct Residential Dose

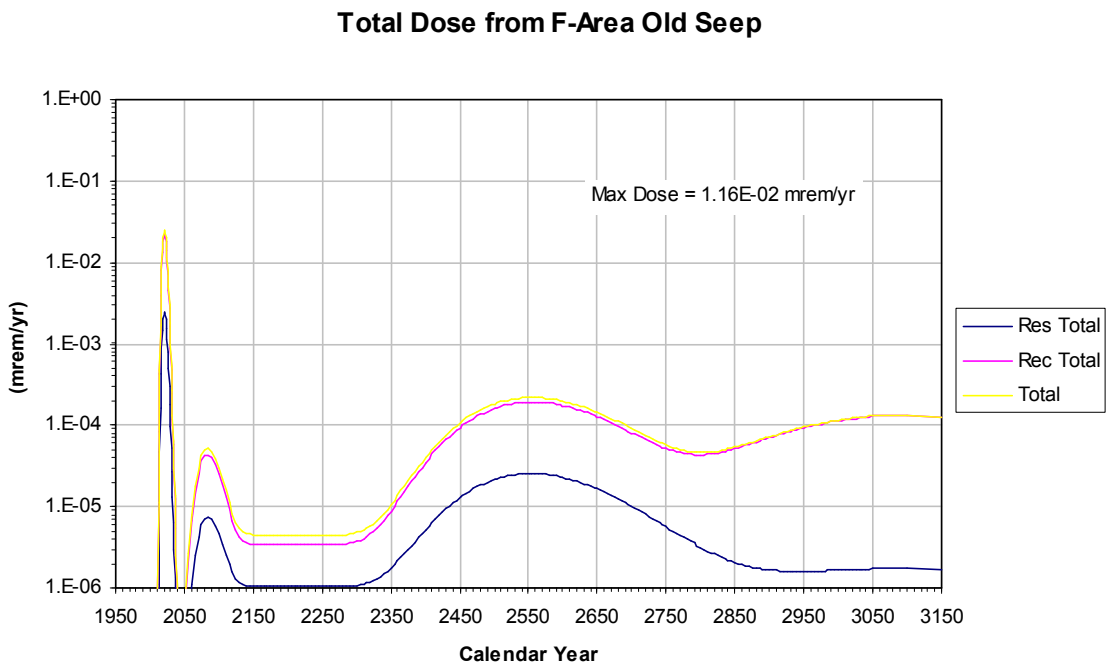


Figure E-20. F-Area Old Seep Total Direct Dose

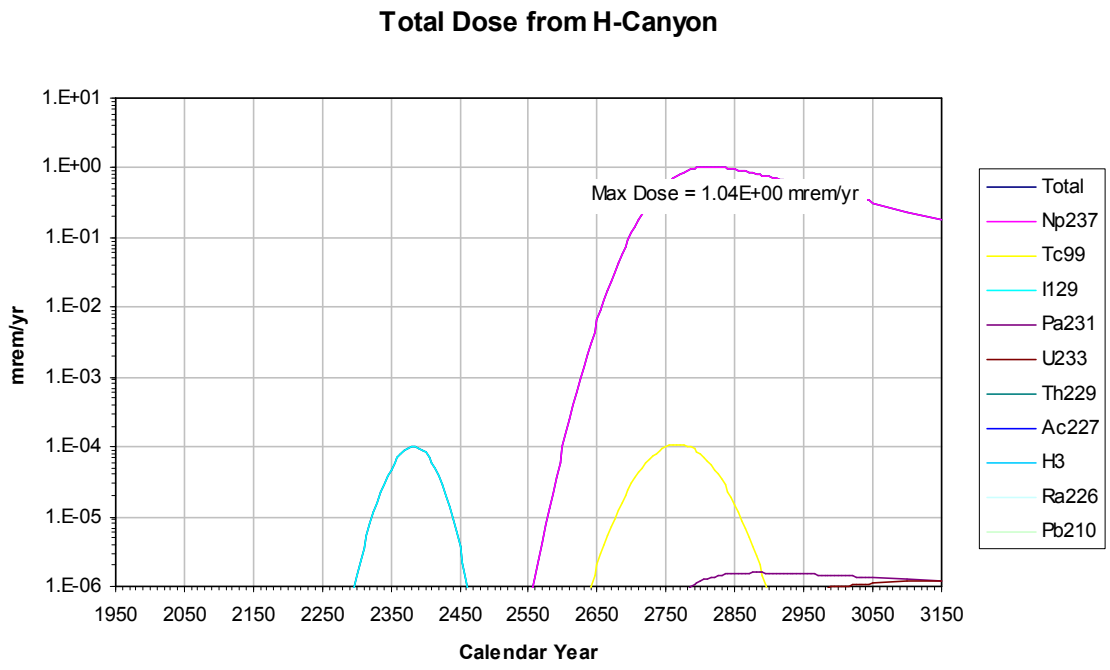


Figure E-21. H-Canyon Direct Dose by Radionuclide

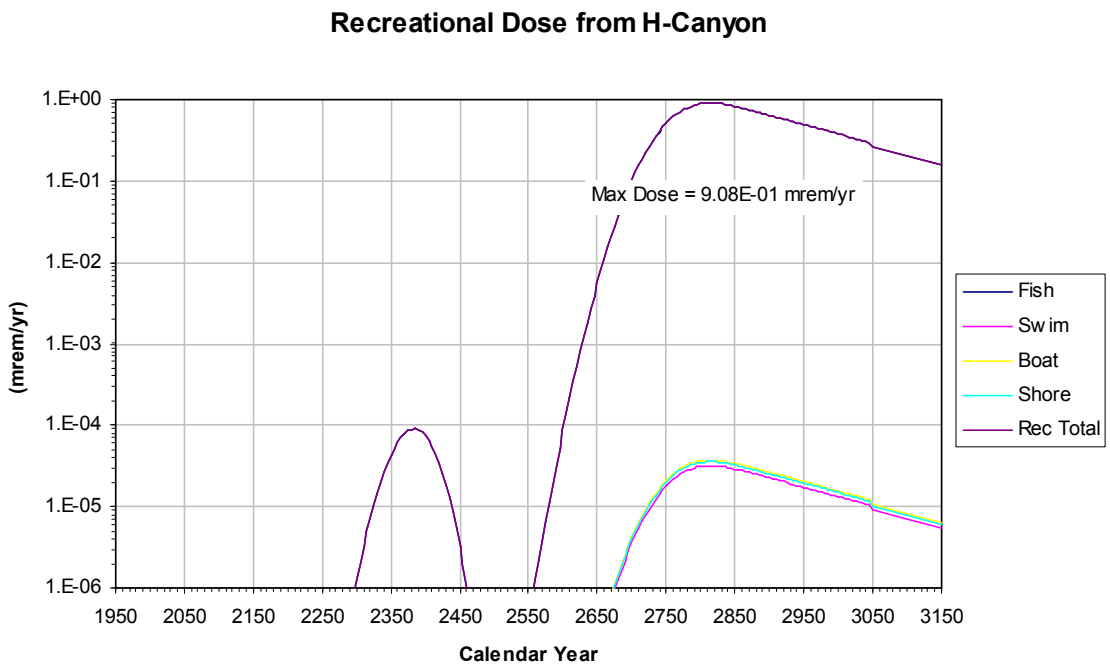


Figure E-22. H-Canyon Recreational Dose

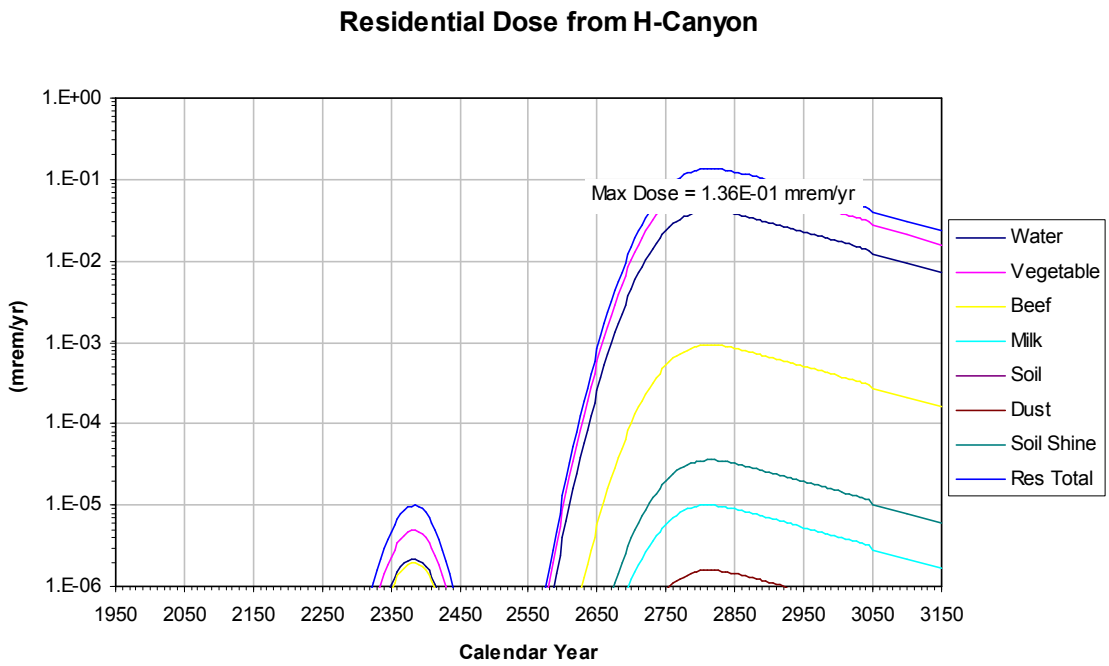


Figure E-23. H-Canyon Direct Residential Dose

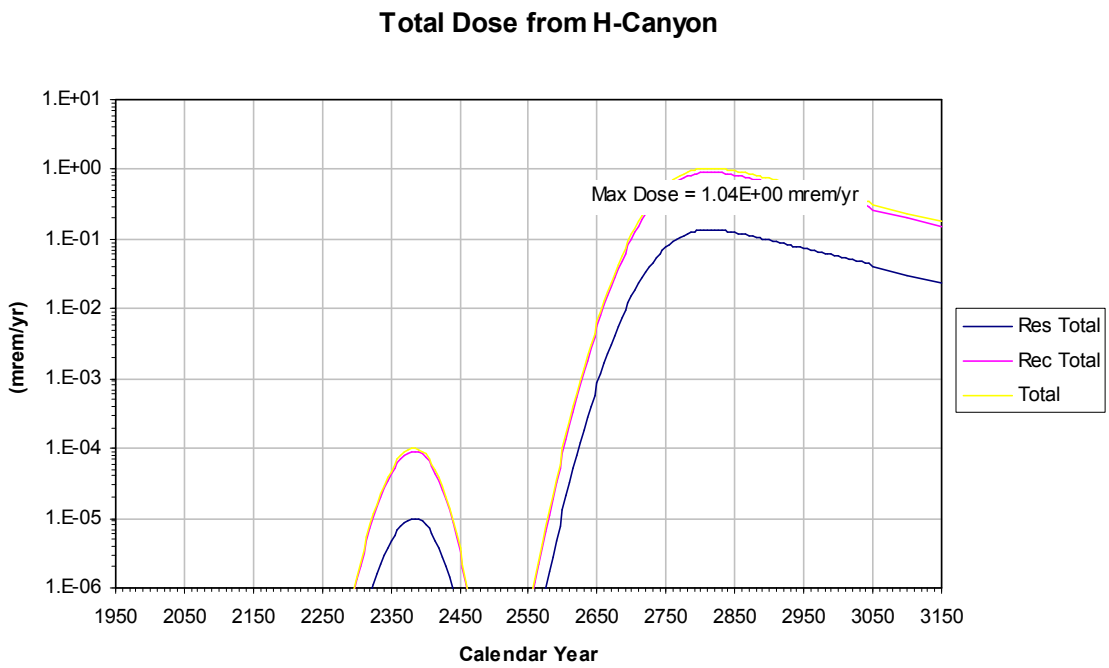


Figure E-24. H-Canyon Total Direct Dose

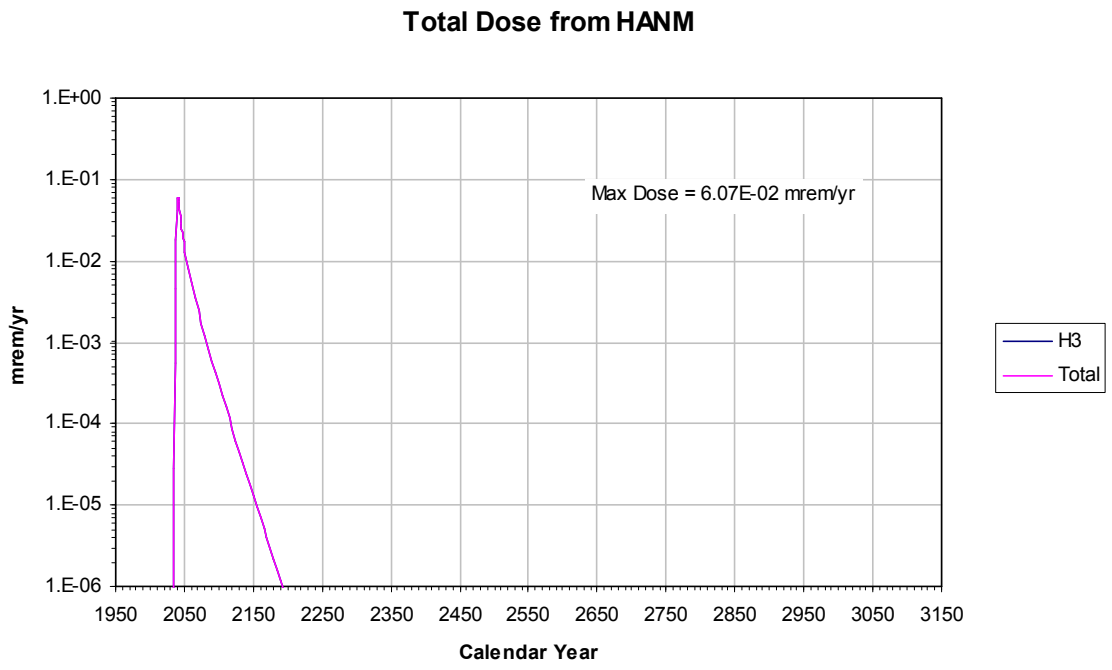


Figure E-25. HANM Direct Dose by Radionuclide

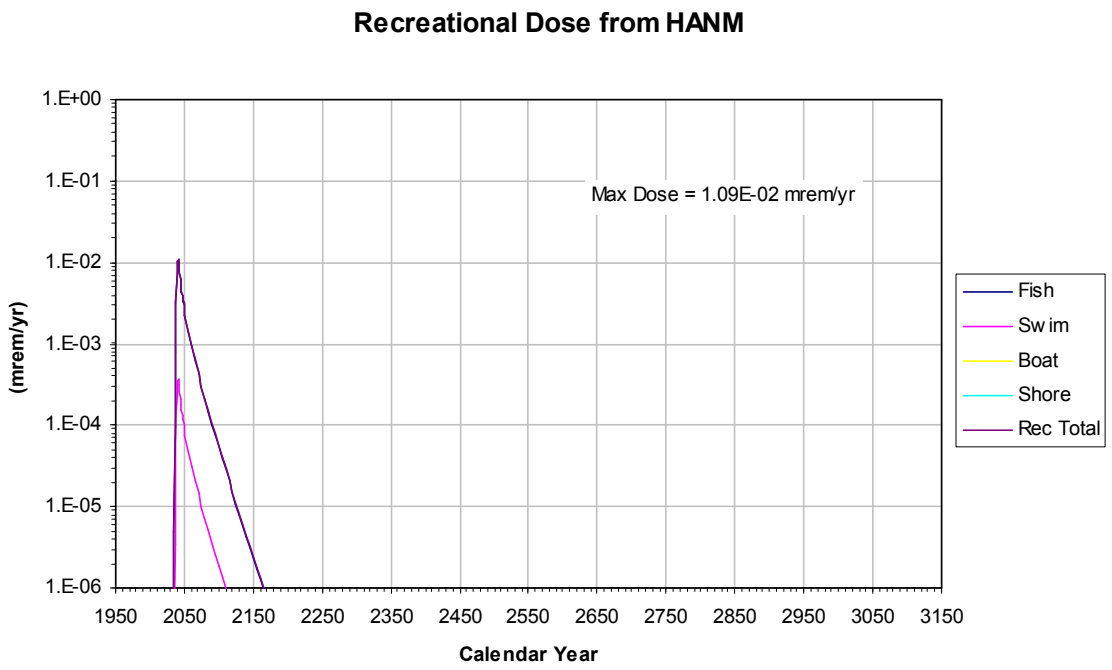


Figure E-26. HANM Recreational Dose

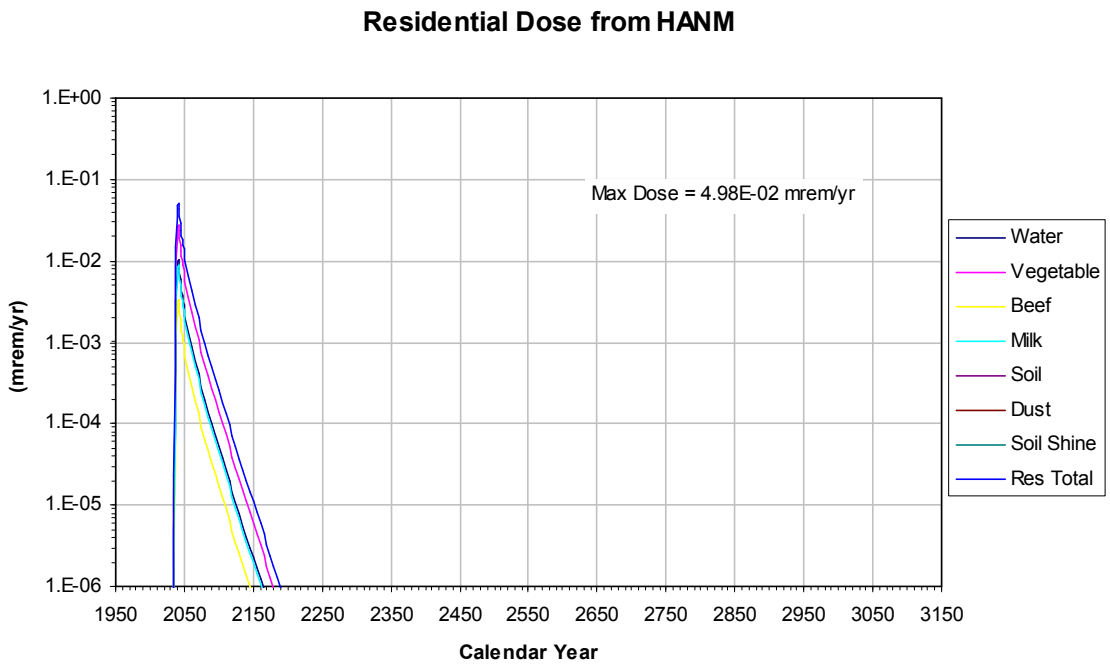


Figure E-27. HANM Direct Residential Dose

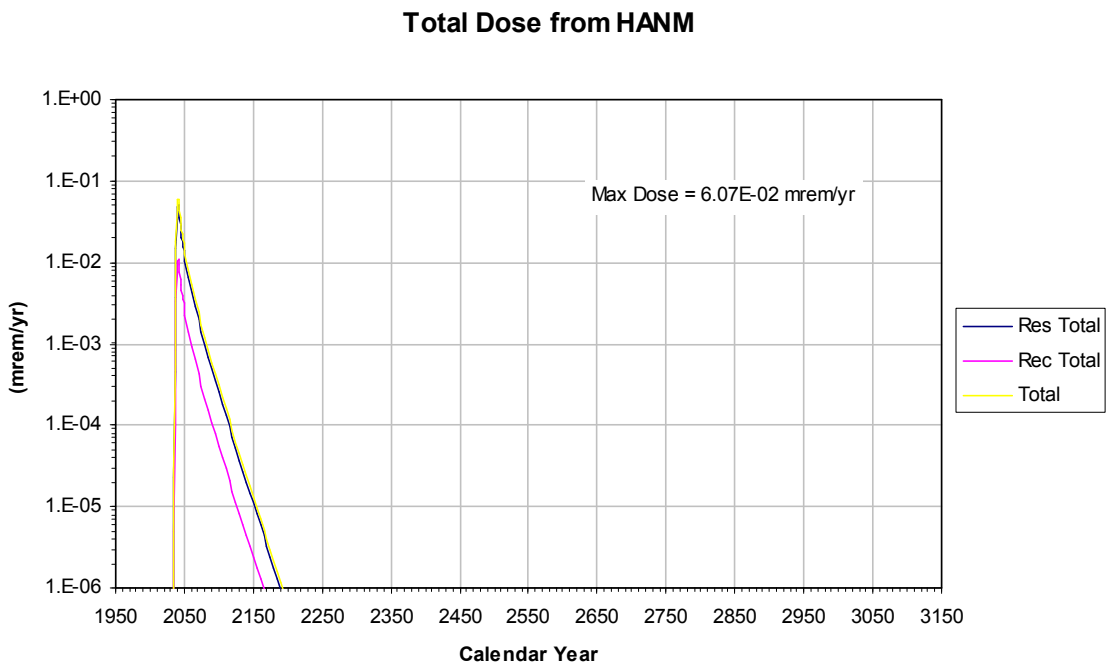


Figure E-28. HANM Total Direct Dose

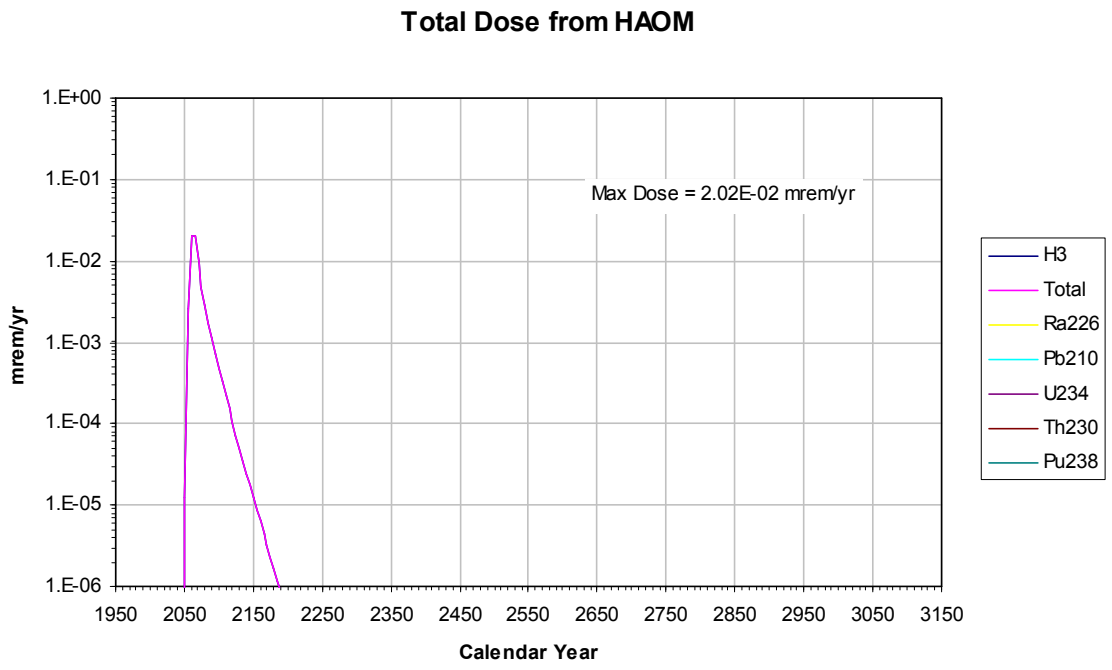


Figure E-29. HAOM Direct Dose by Radionuclide

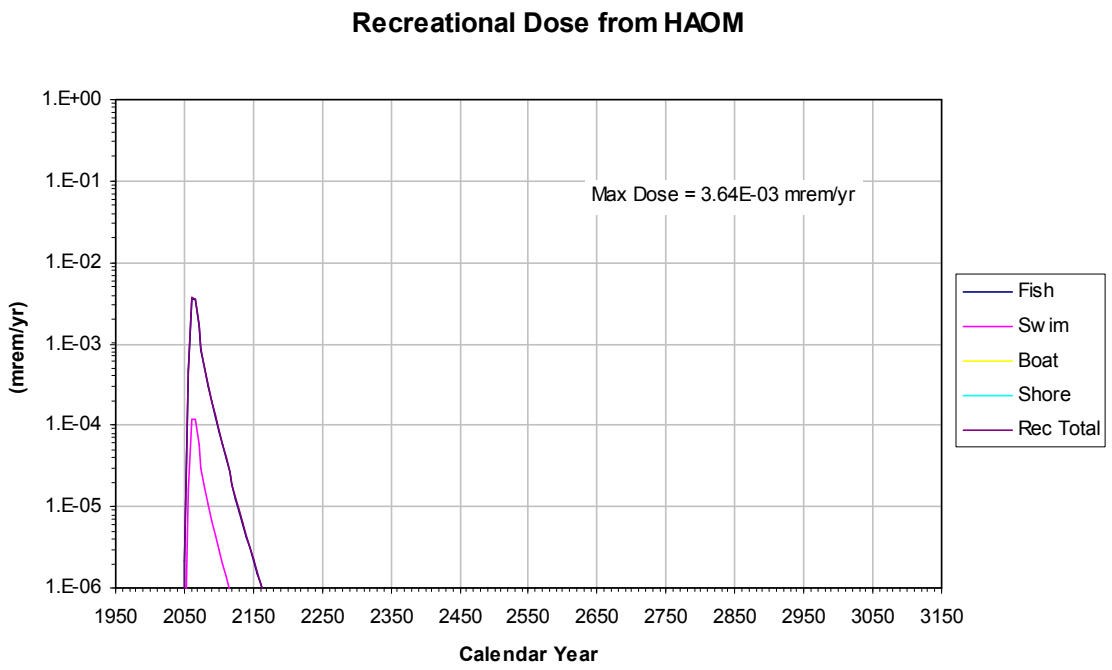


Figure E-30. HAOM Recreational Dose

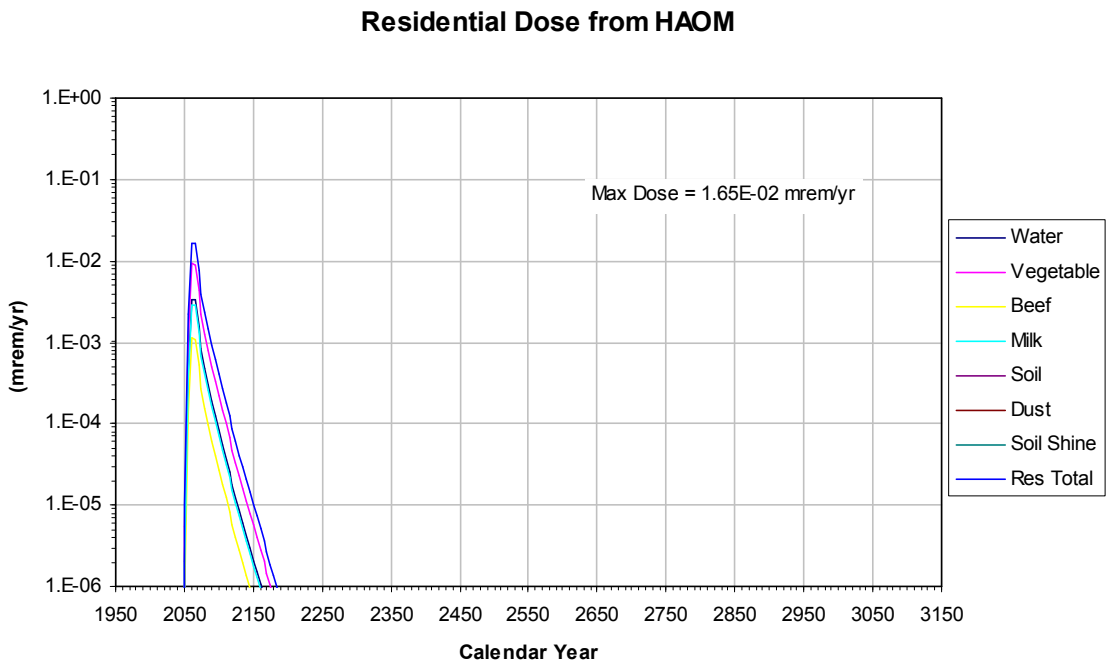


Figure E-31. HAOM Direct Residential Dose

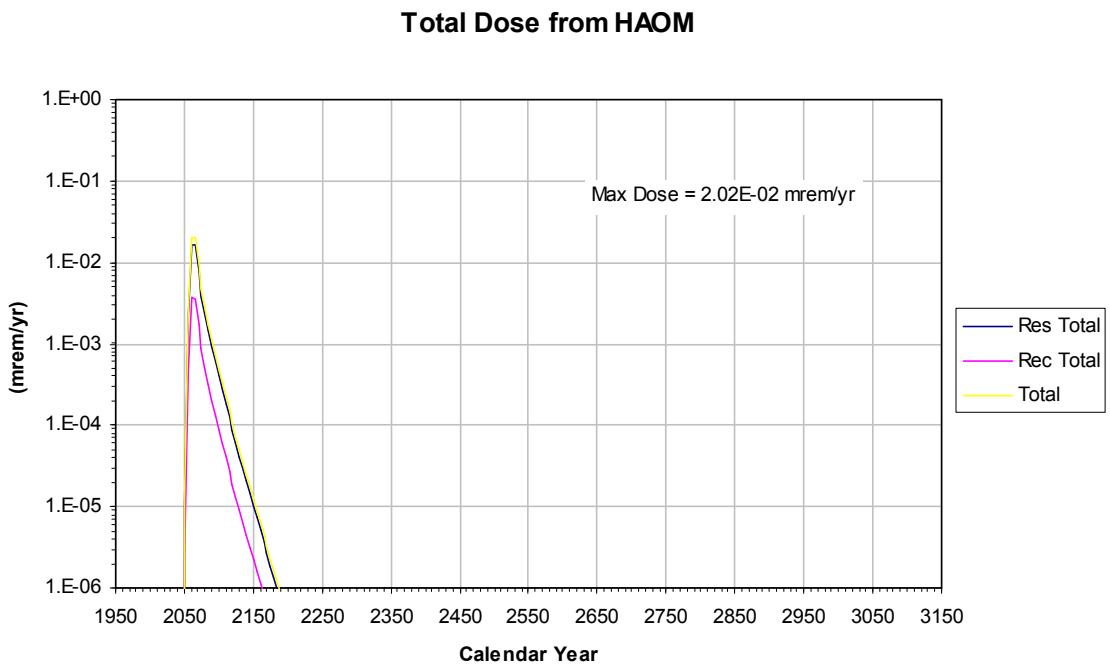


Figure E-32. HAOM Total Direct Dose

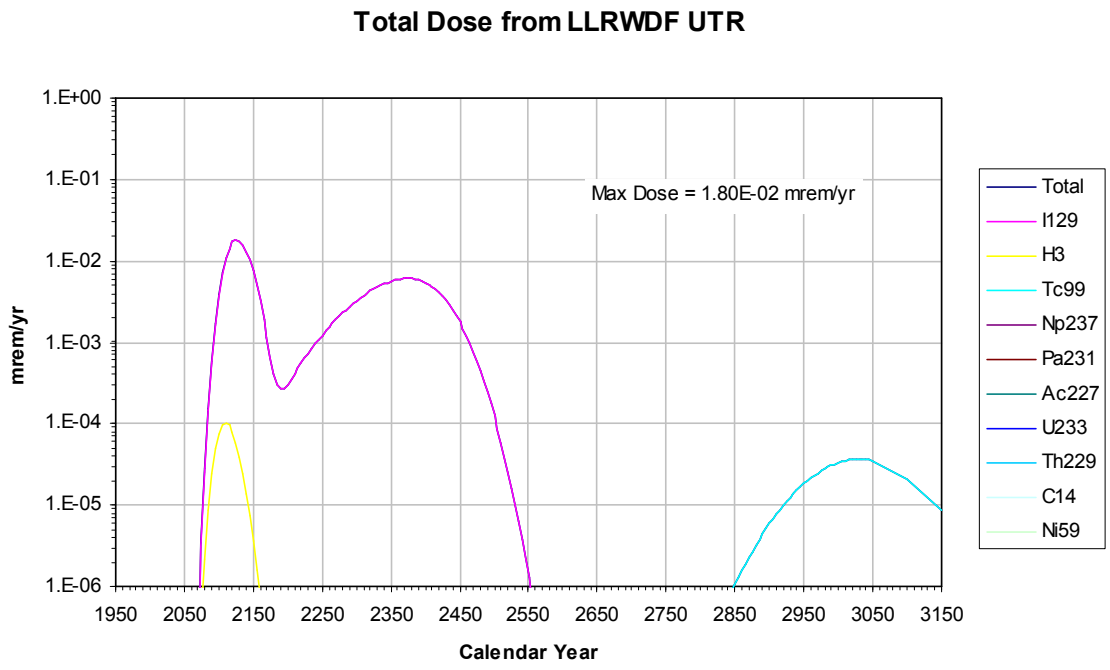


Figure E-33. LLRWDF UTR Direct Dose by Radionuclide

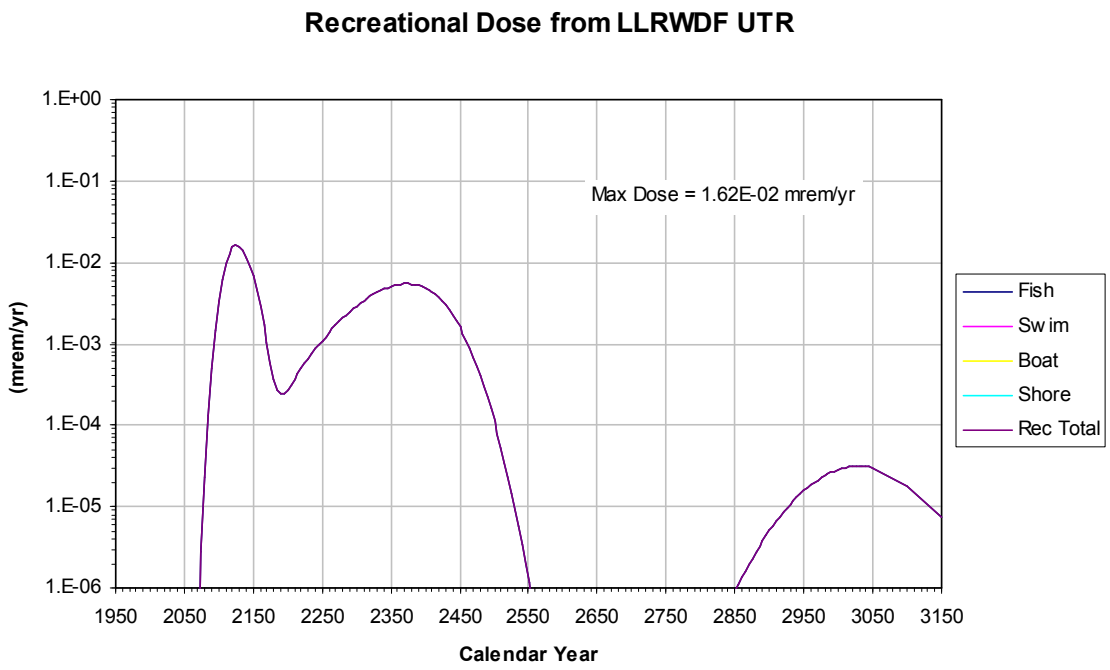


Figure E-34. LLRWDF UTR Recreational Dose

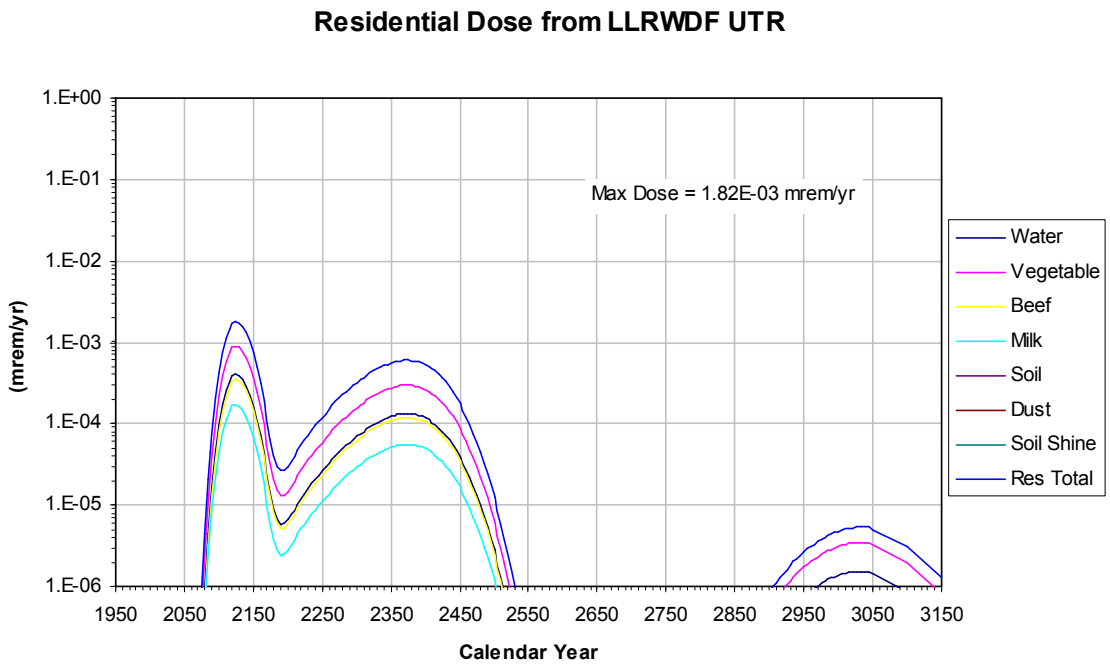


Figure E-35. LLRWDF UTR Direct Residential Dose

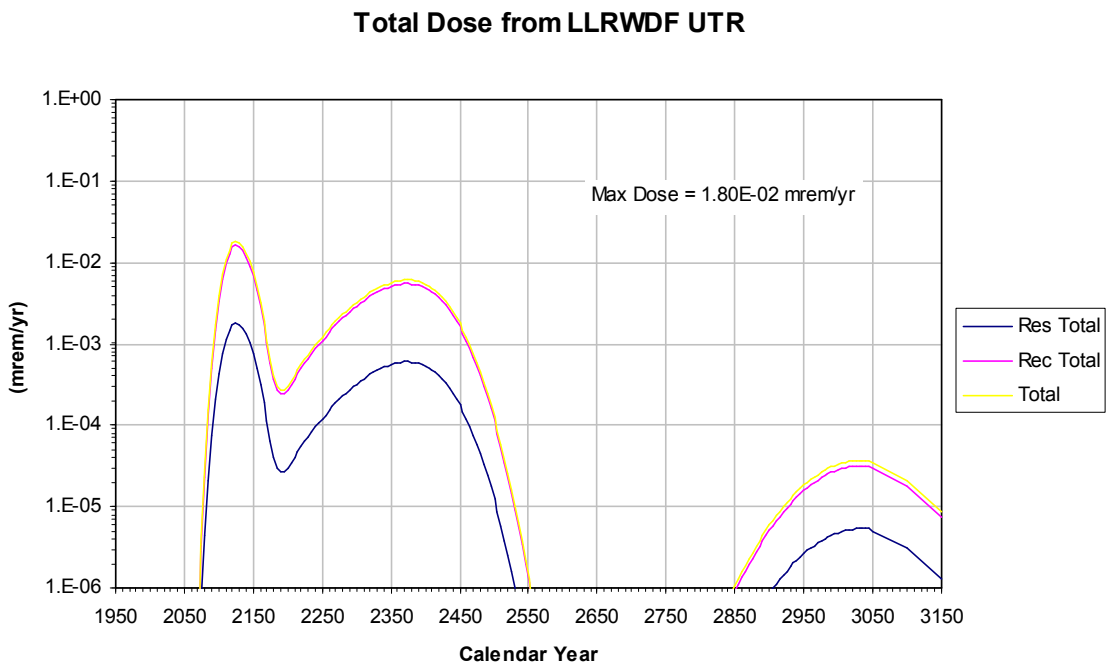


Figure E-36. LLRWDF UTR Total Direct Dose

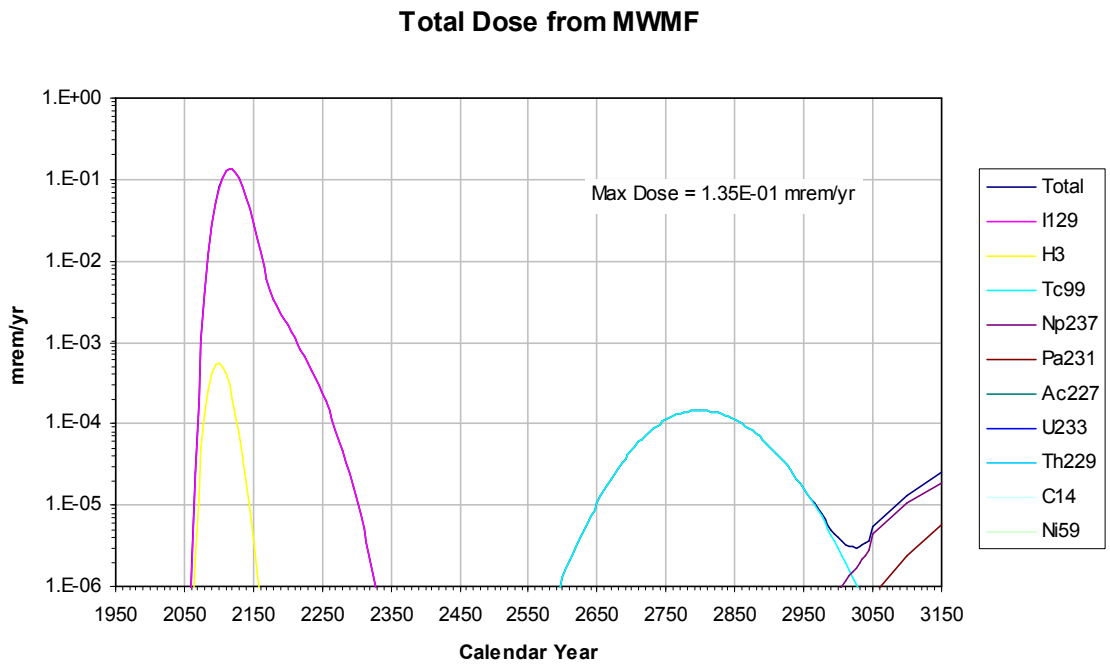


Figure E-37. MWMF Direct Dose by Radionuclide

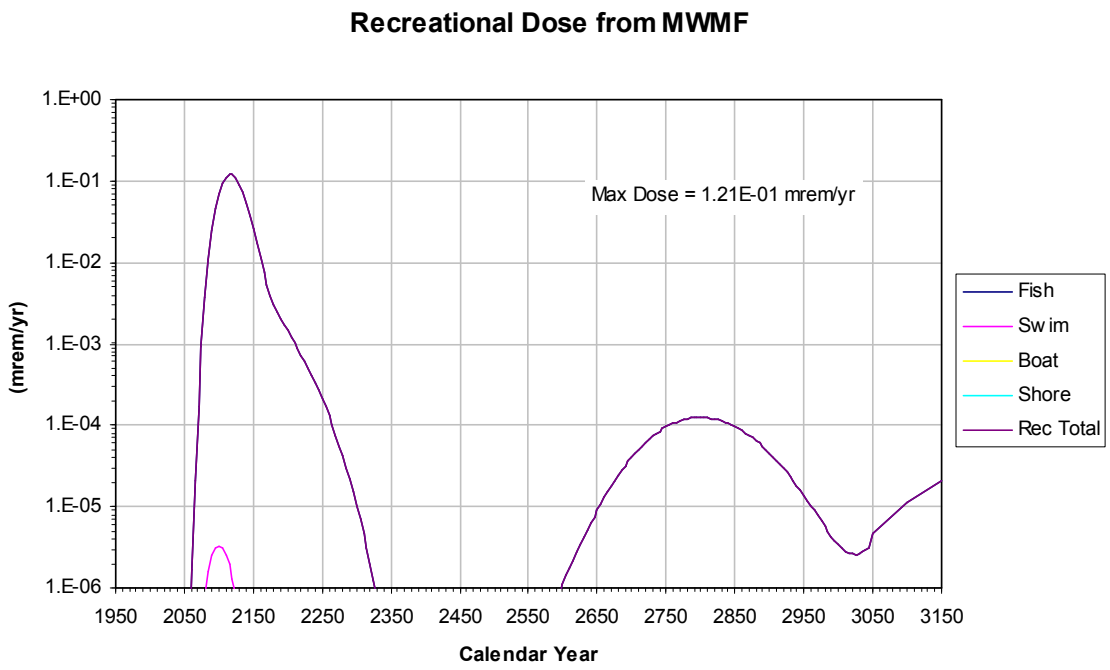


Figure E-38. MWMF Recreational Dose

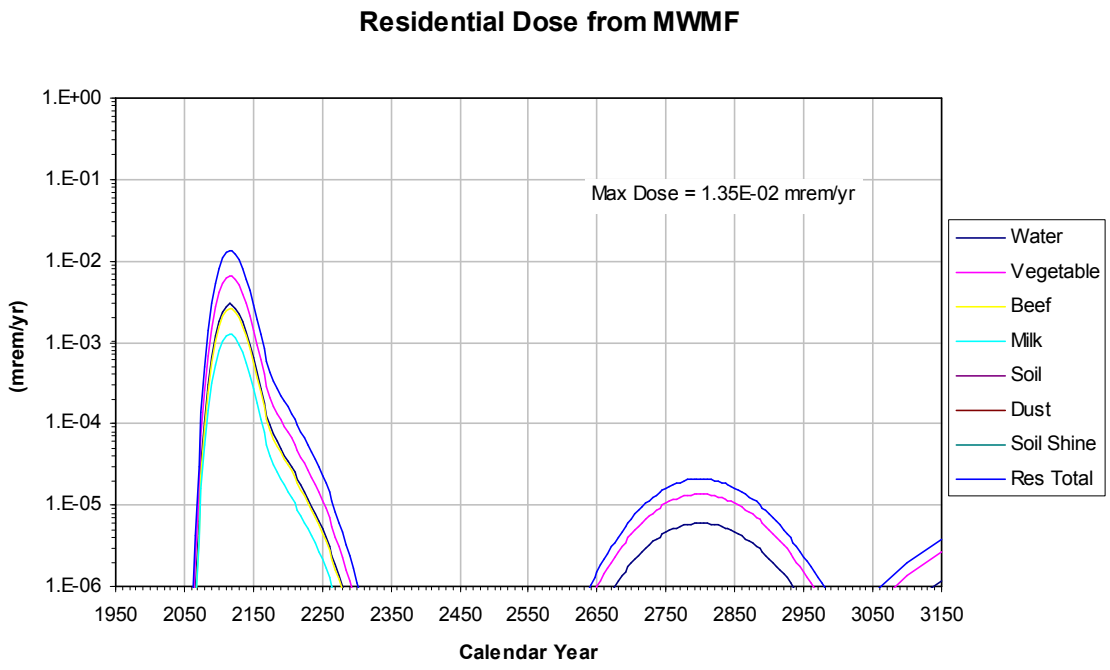


Figure E-39. MWMF Direct Residential Dose

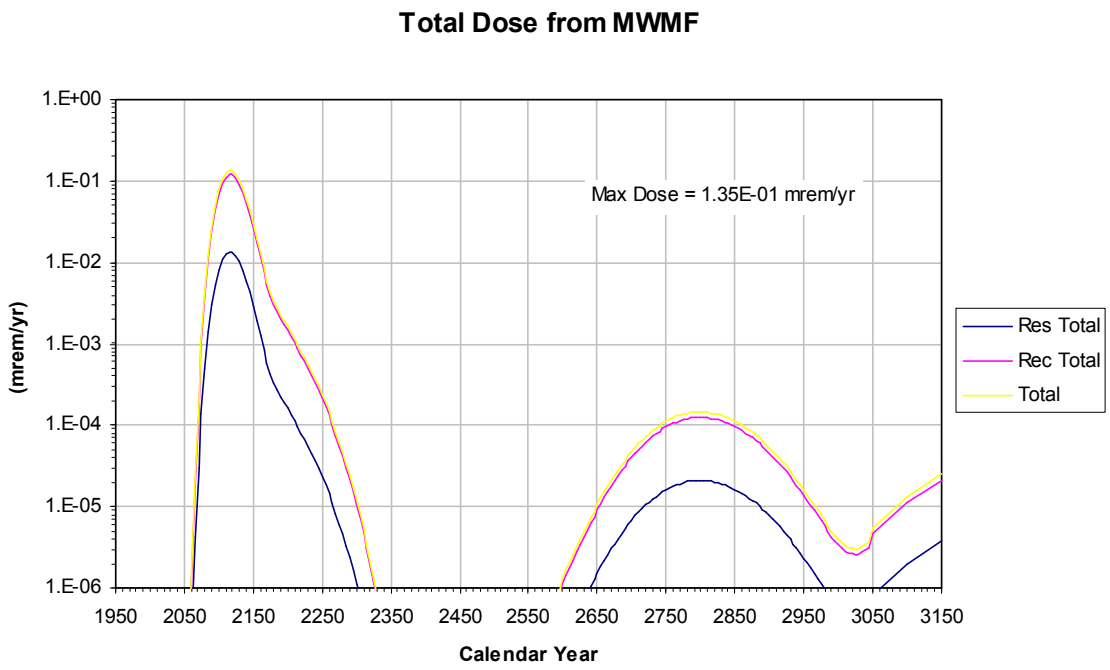


Figure E-40. MWMF Total Direct Dose

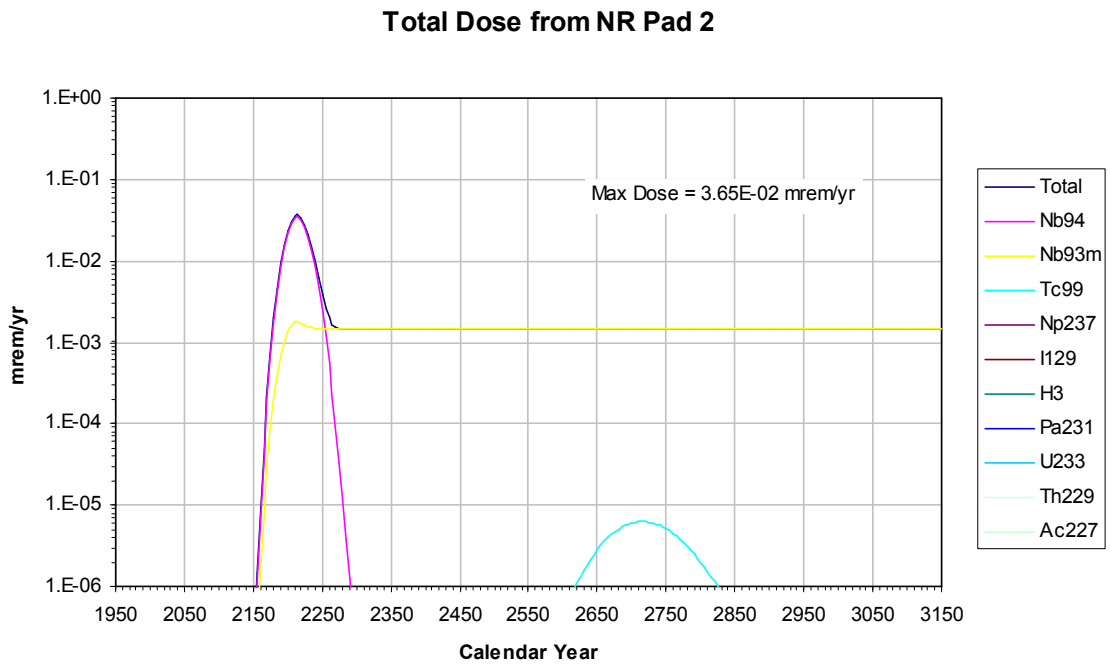


Figure E-41. NR Pad 2 Direct Dose by Radionuclide

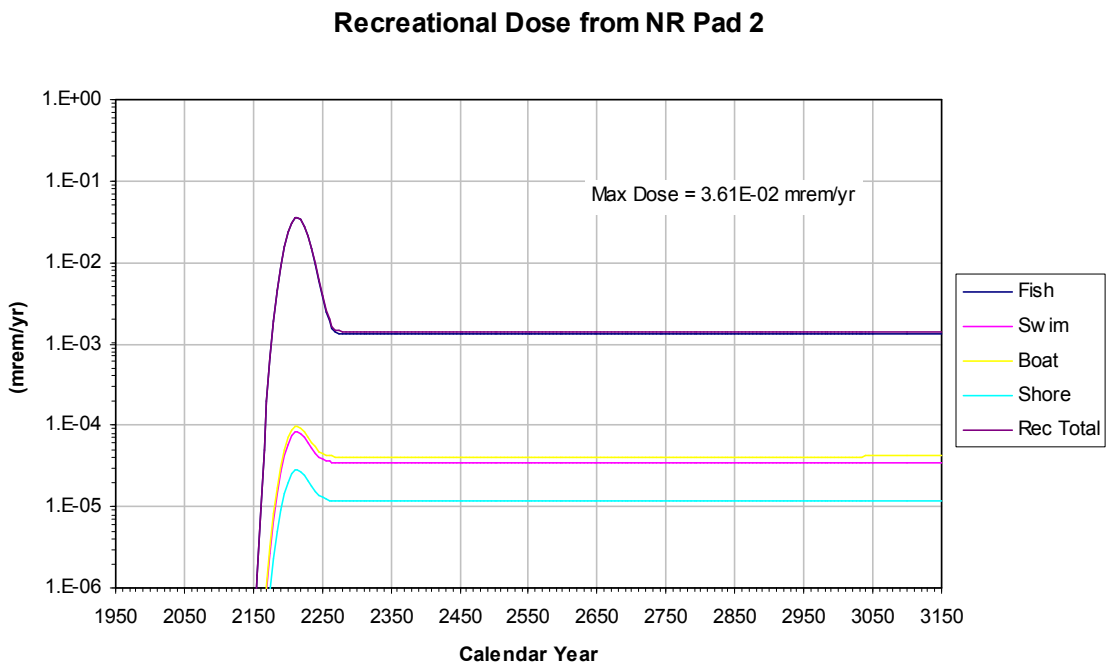


Figure E-42. NR Pad 2 Recreational Dose

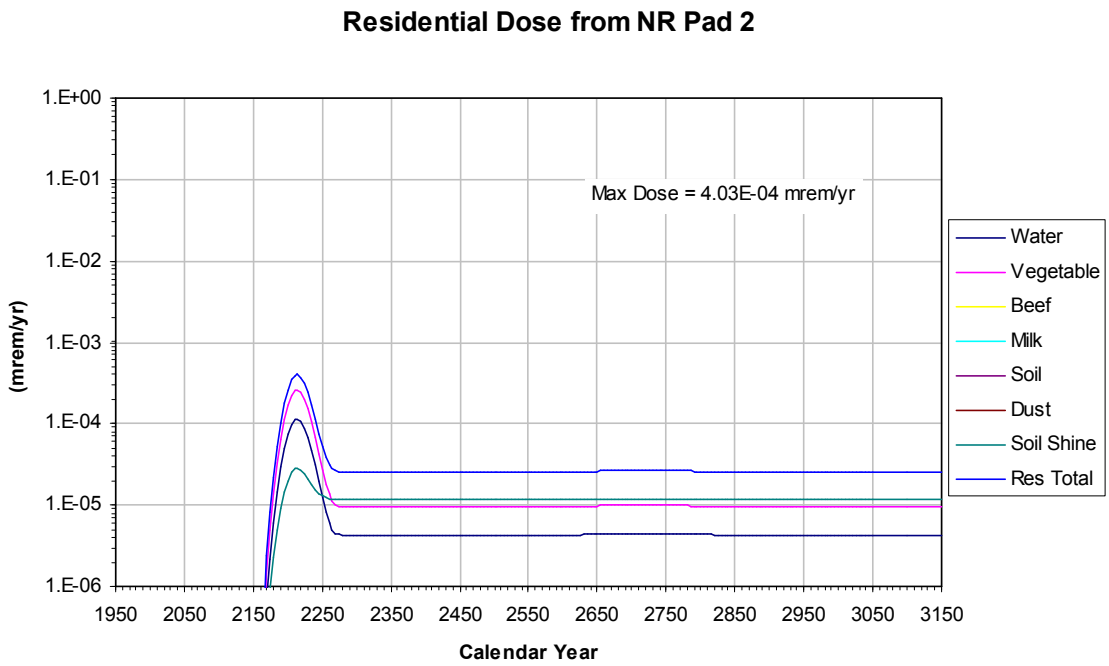


Figure E-43. NR Pad 2 Direct Residential Dose

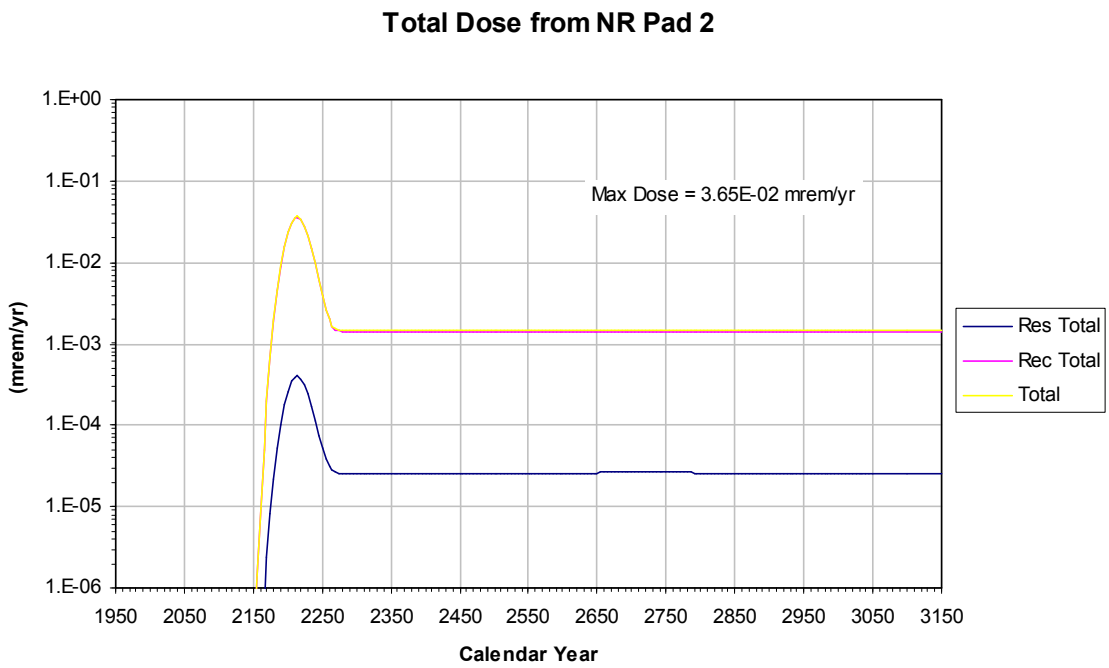


Figure E-44. NR Pad 2 Total Direct Dose

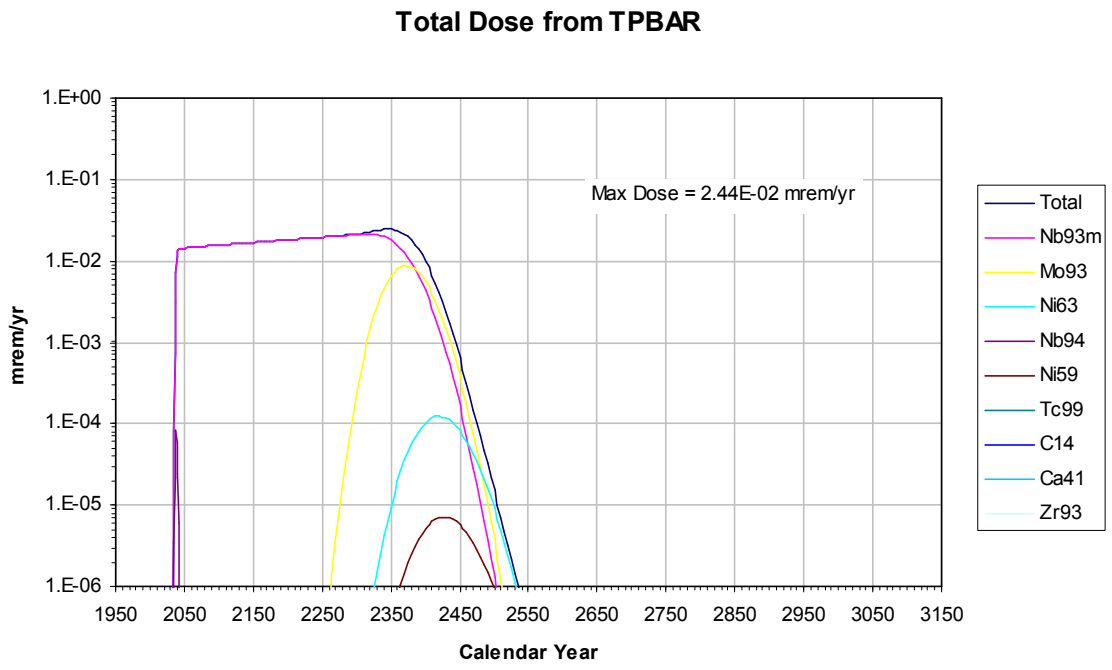


Figure E-45. TPBAR Direct Dose by Radionuclide

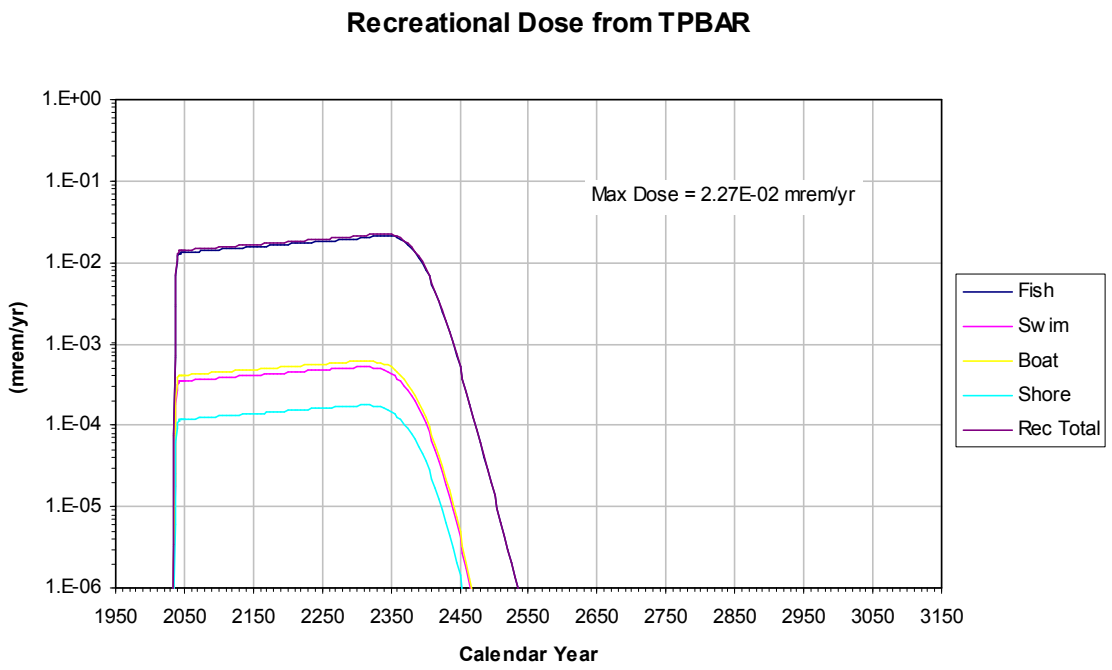


Figure E-46. TPBAR Recreational Dose

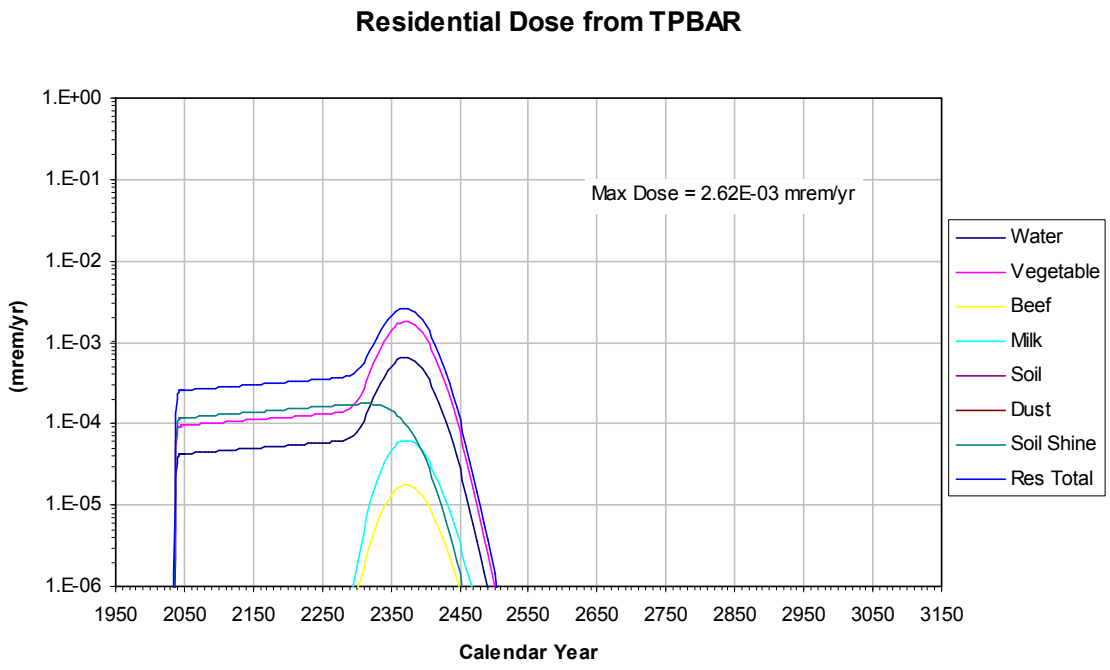


Figure E-47. TPBAR Direct Residential Dose

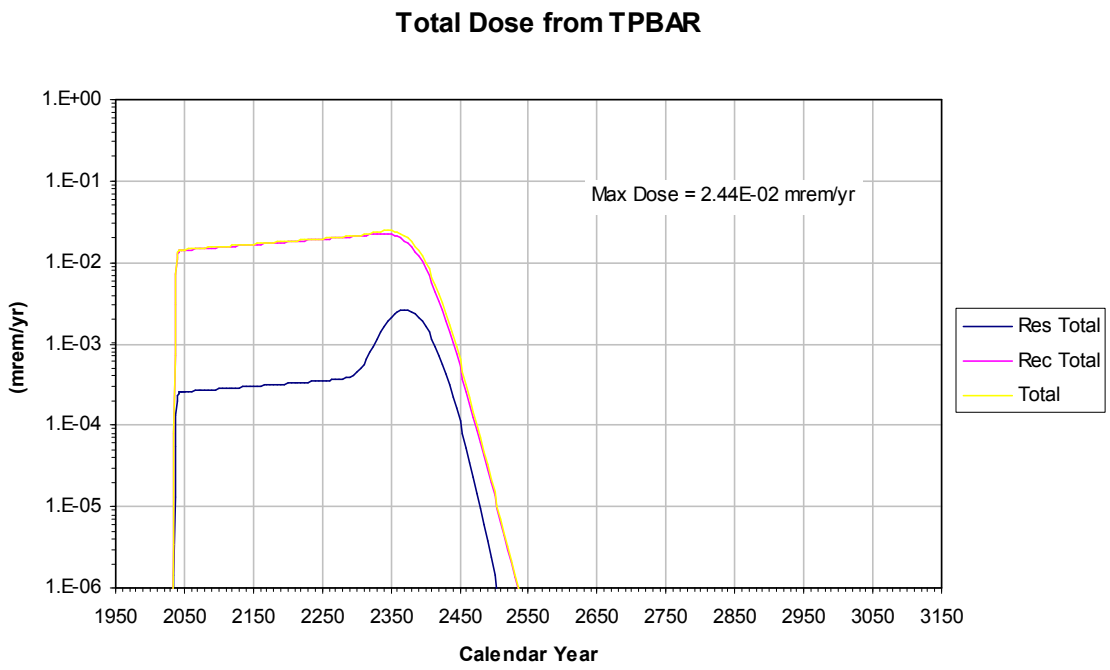


Figure E-48. TPBAR Total Direct Dose

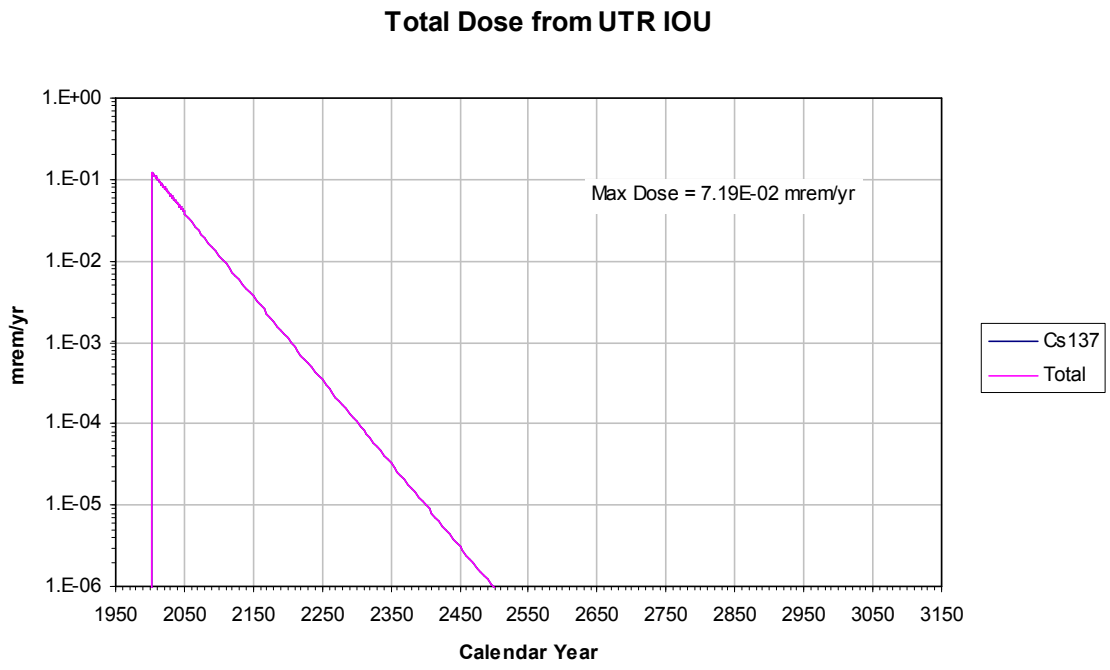


Figure E-49. UTR IOU Direct Dose by Radionuclide

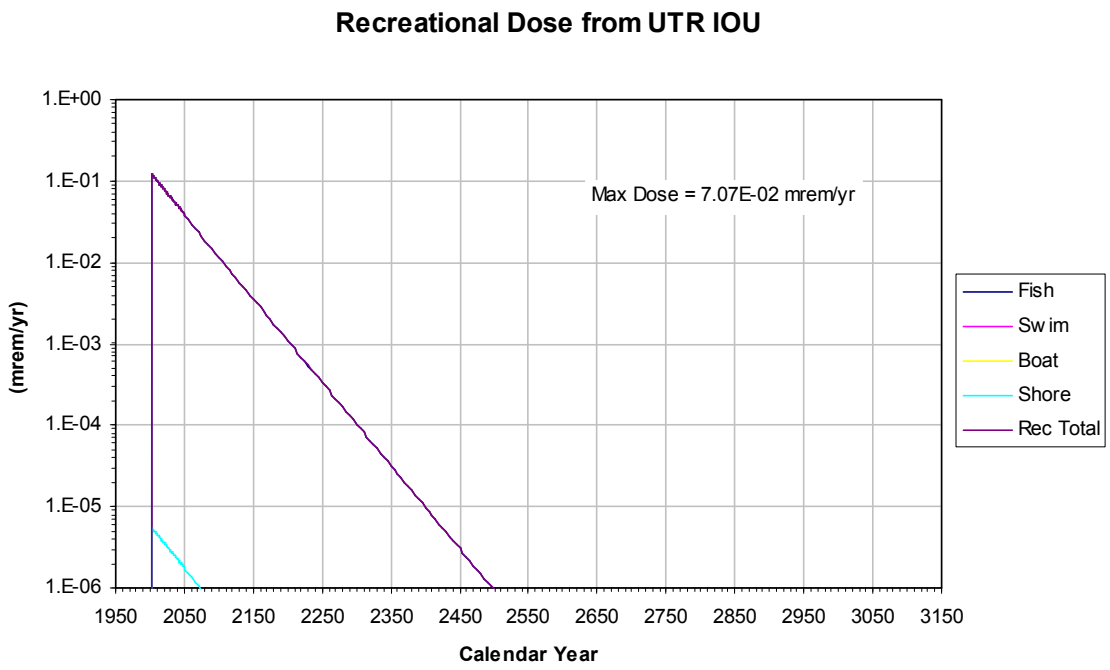


Figure E-50. UTR IOU Recreational Dose

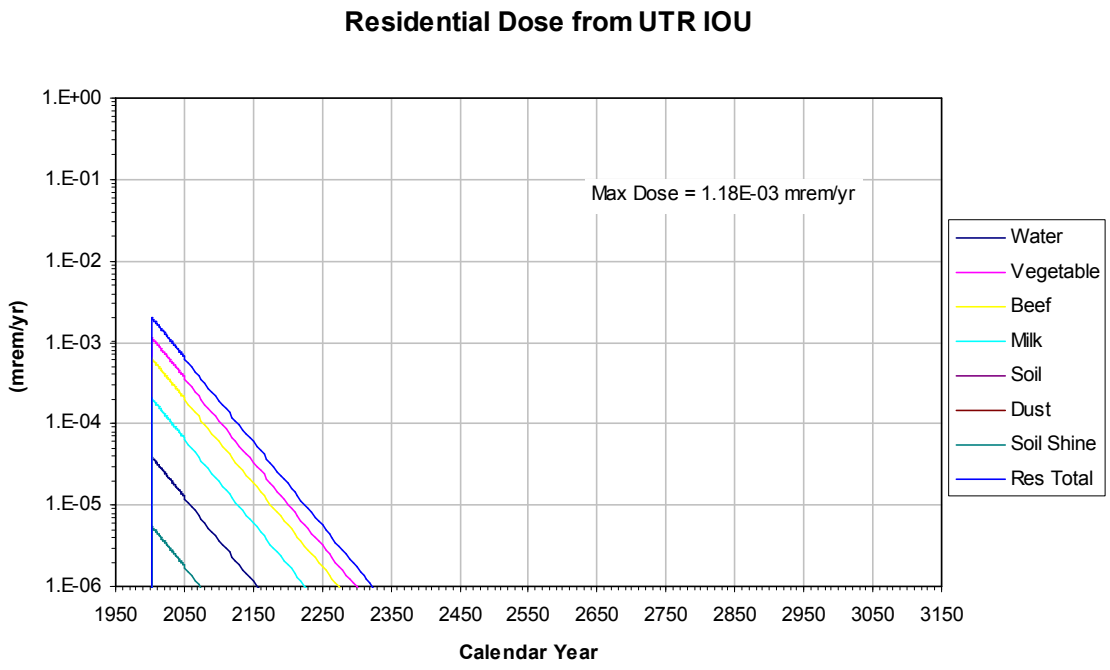


Figure E-51. UTR IOU Direct Residential Dose

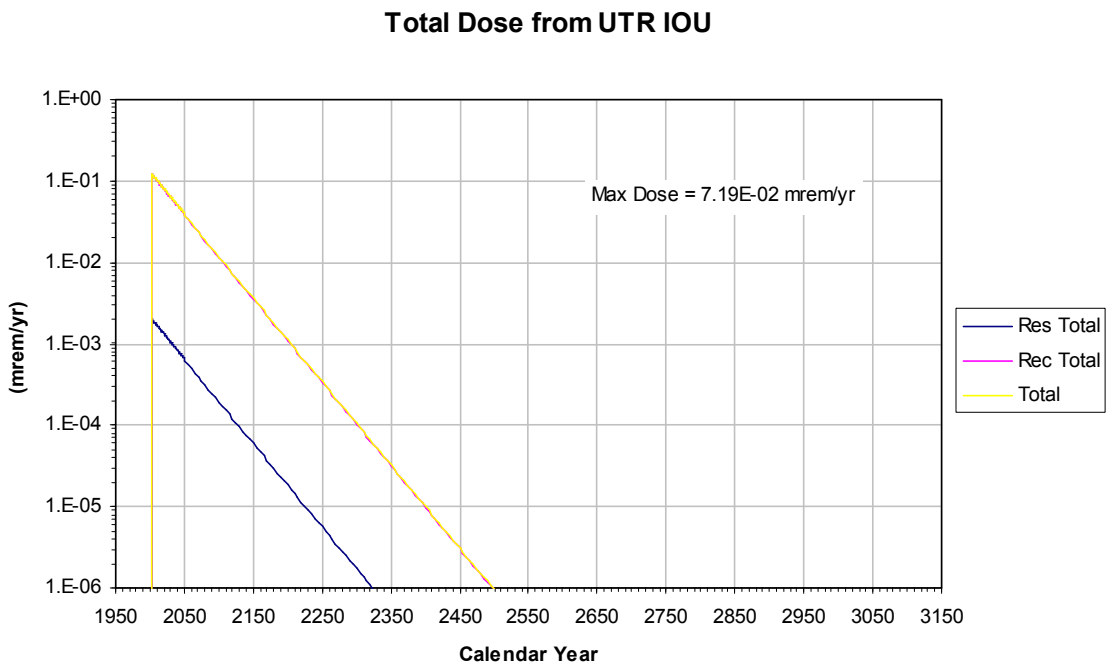


Figure E-52. UTR IOU Total Direct Dose

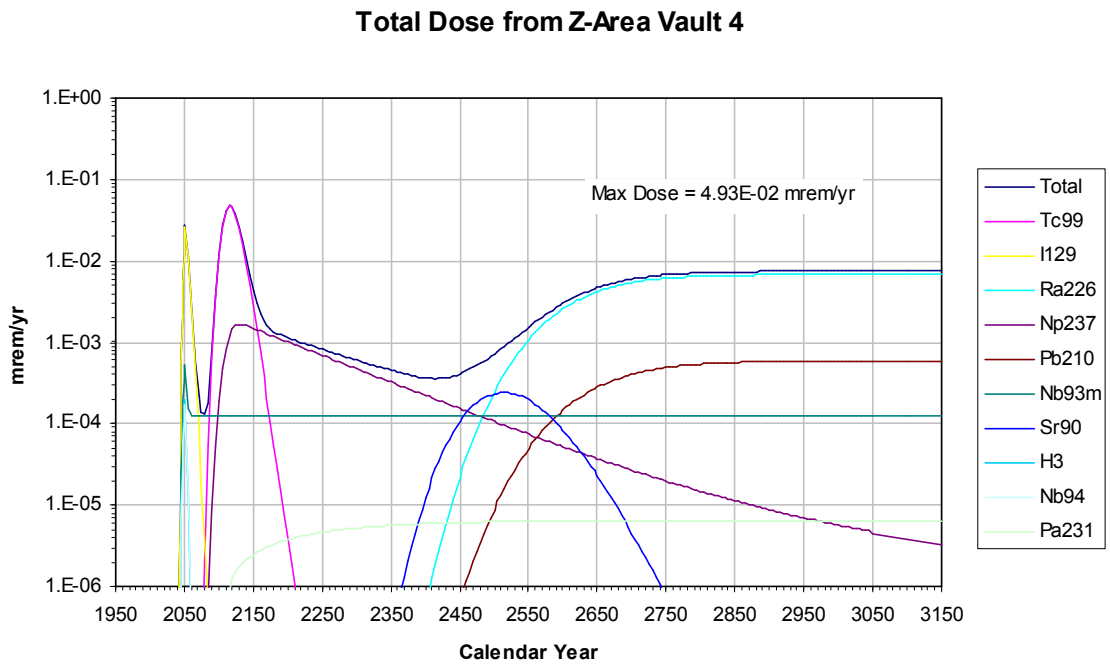


Figure E-53. Z-Area Vault 4 Direct Dose by Radionuclide

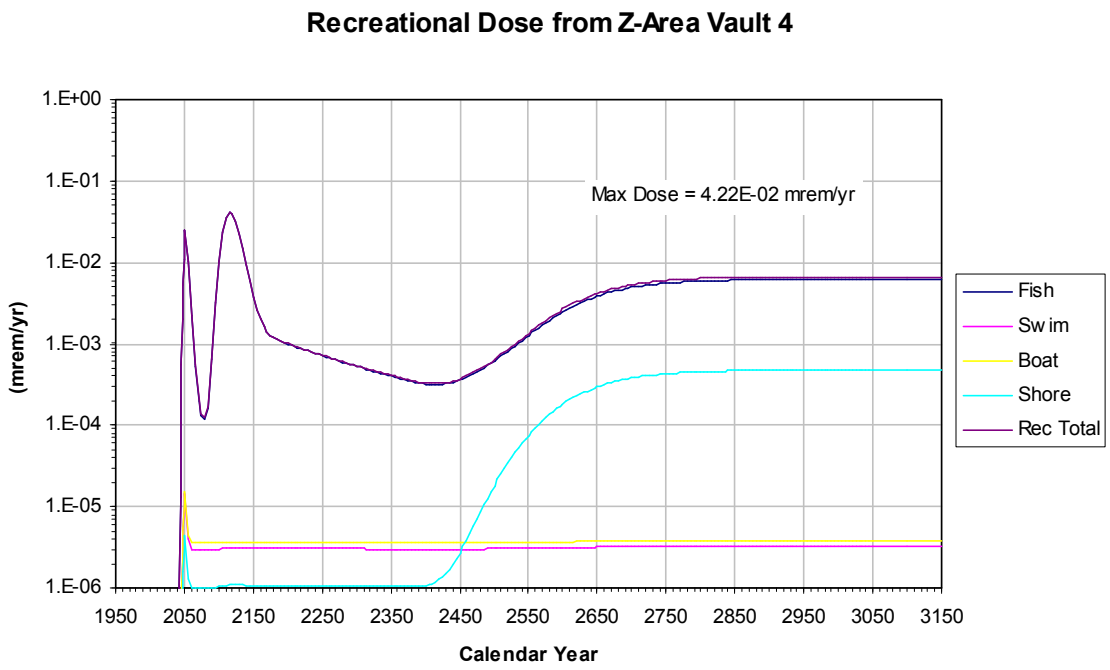


Figure E-54. Z-Area Vault 4 Recreational Dose

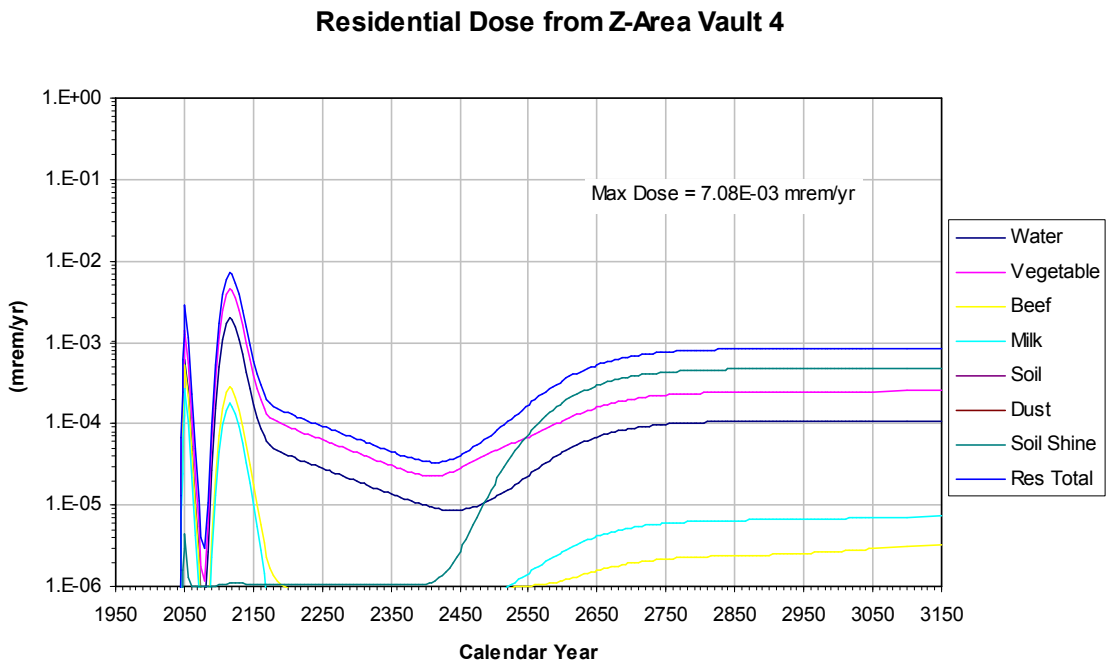


Figure E-55. Z-Area Vault 4 Direct Residential Dose

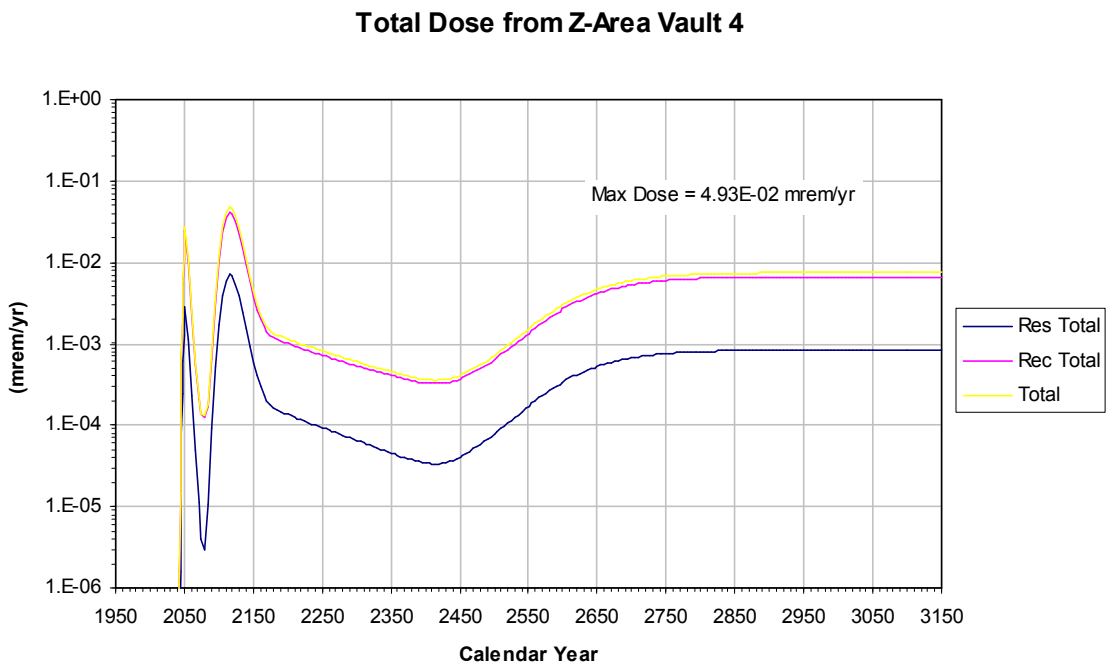


Figure E-56. Z-Area Vault 4 Total Direct Dose

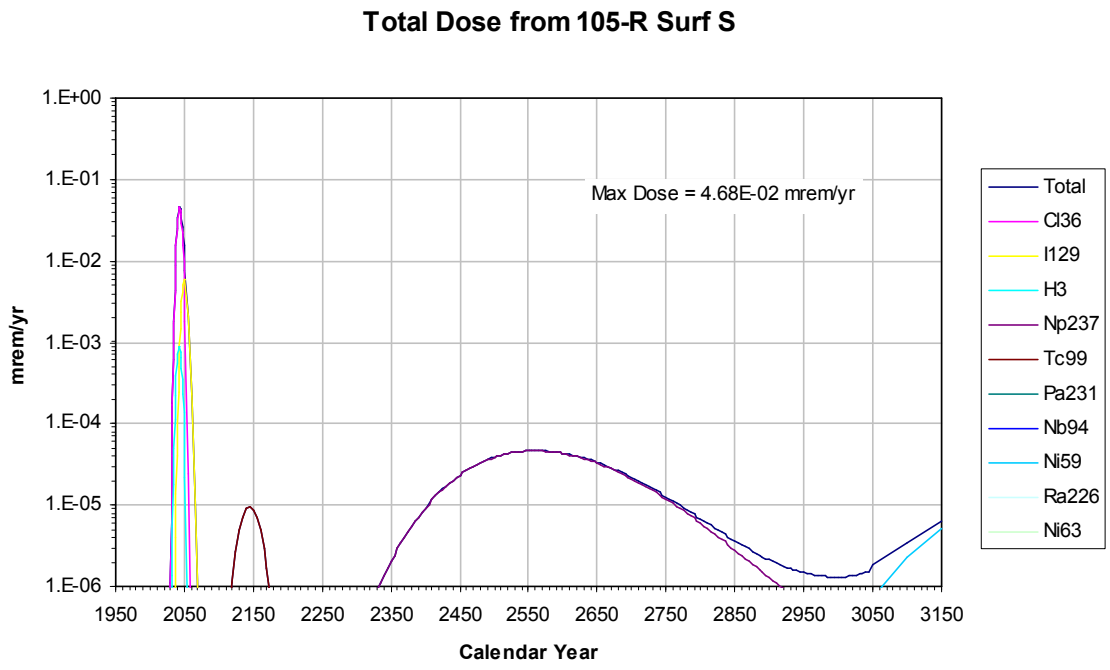


Figure E-57. 105-R Surf S Direct Dose by Radionuclide

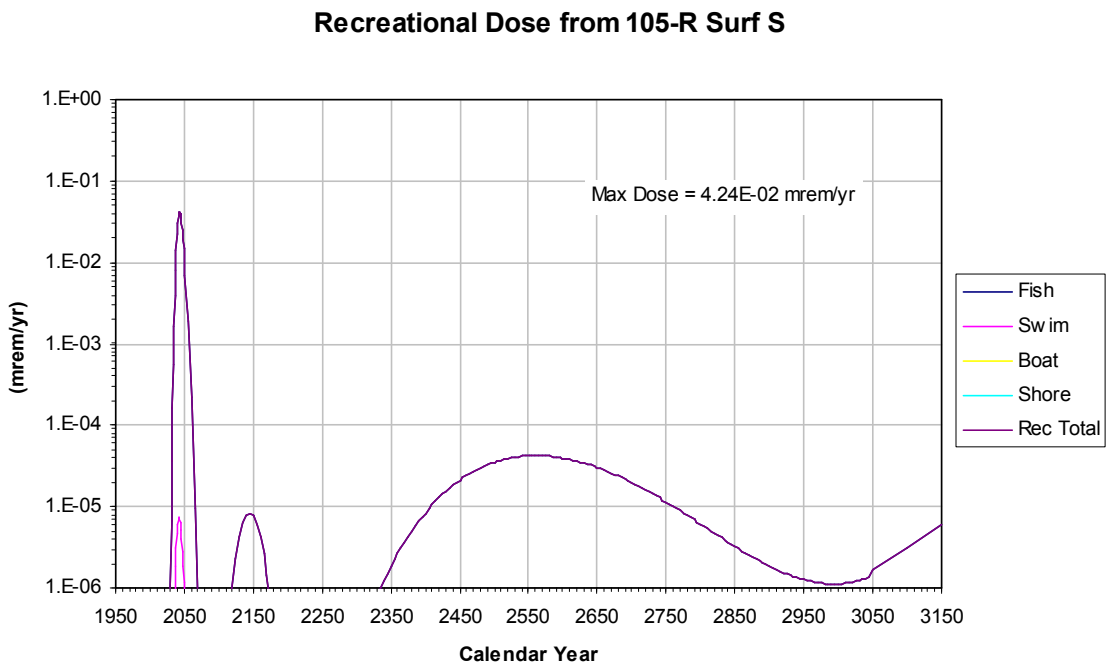


Figure E-58. 105-R Surf S Recreational Dose

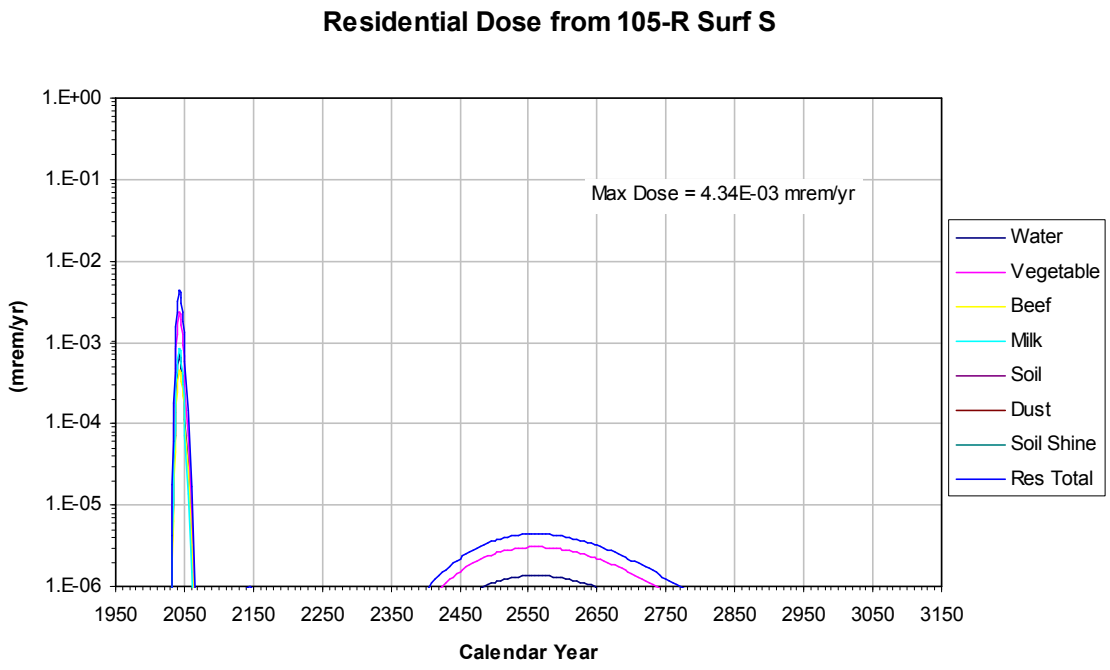


Figure E-59. 105-R Surf S Direct Residential Dose

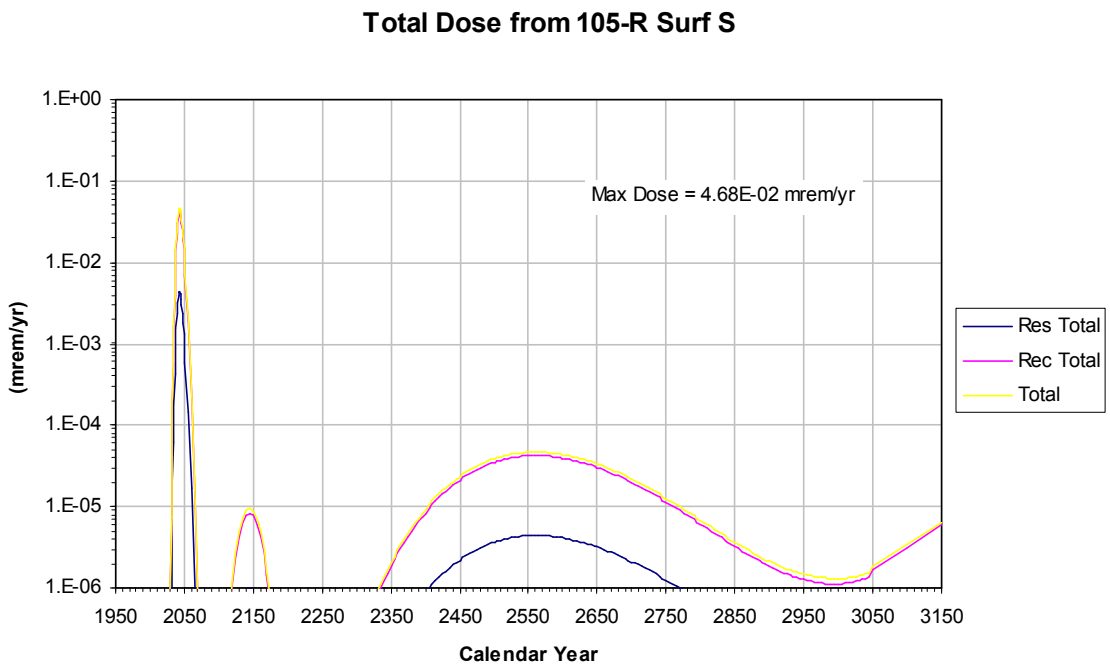


Figure E-60. 105-R Surf S Total Direct Dose

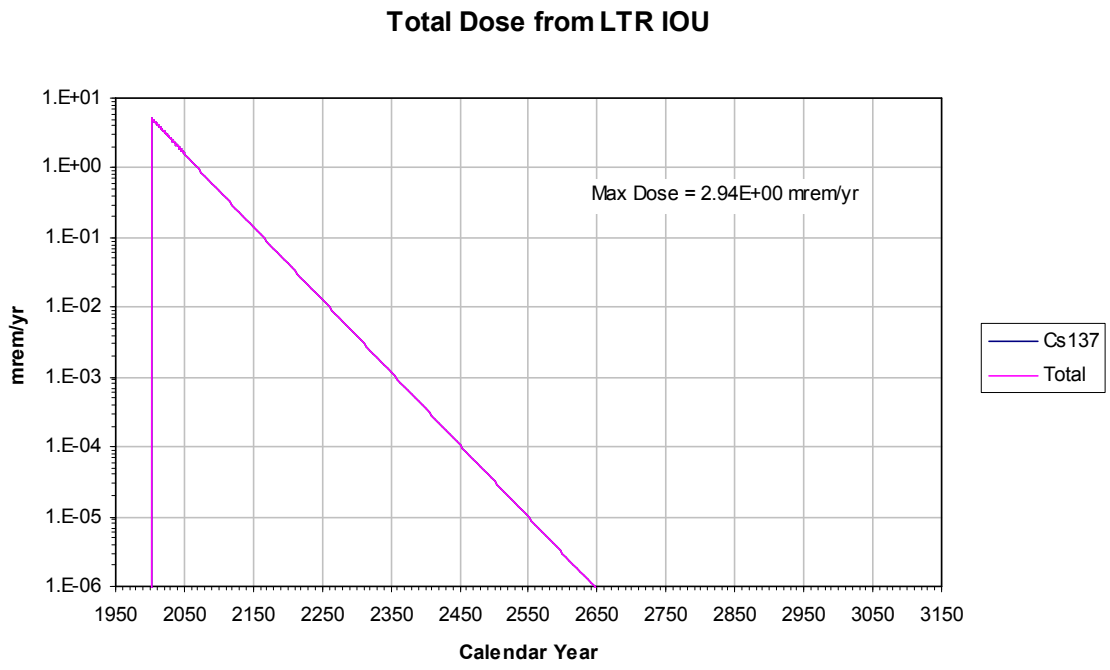


Figure E-61. LTR IOU Direct Dose by Radionuclide

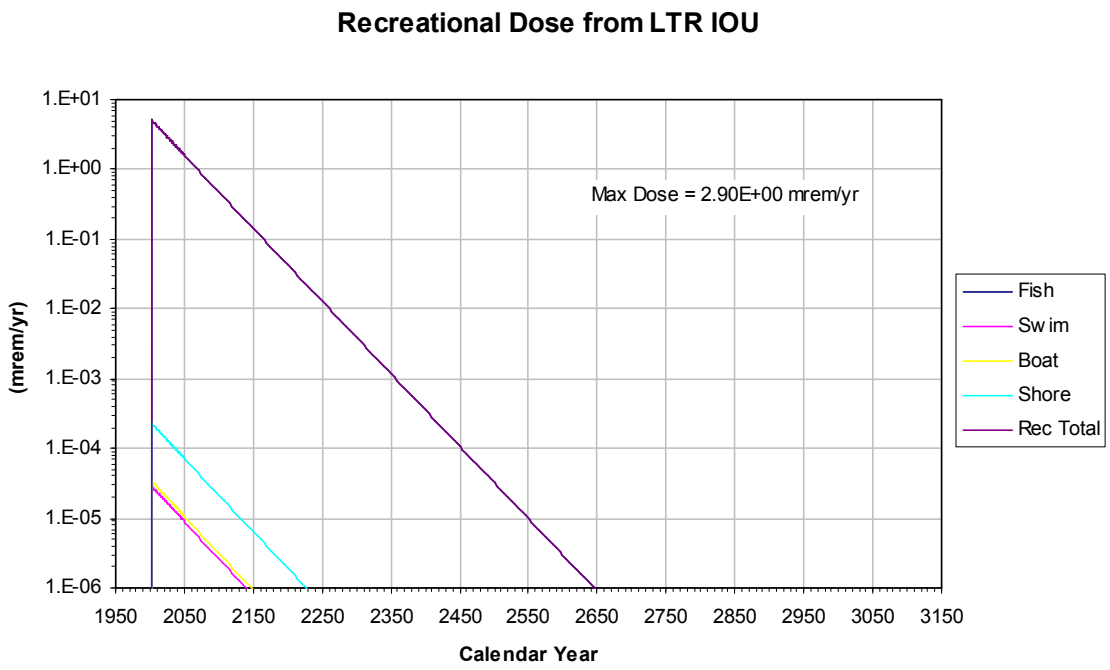


Figure E-62. LTR IOU Recreational Dose

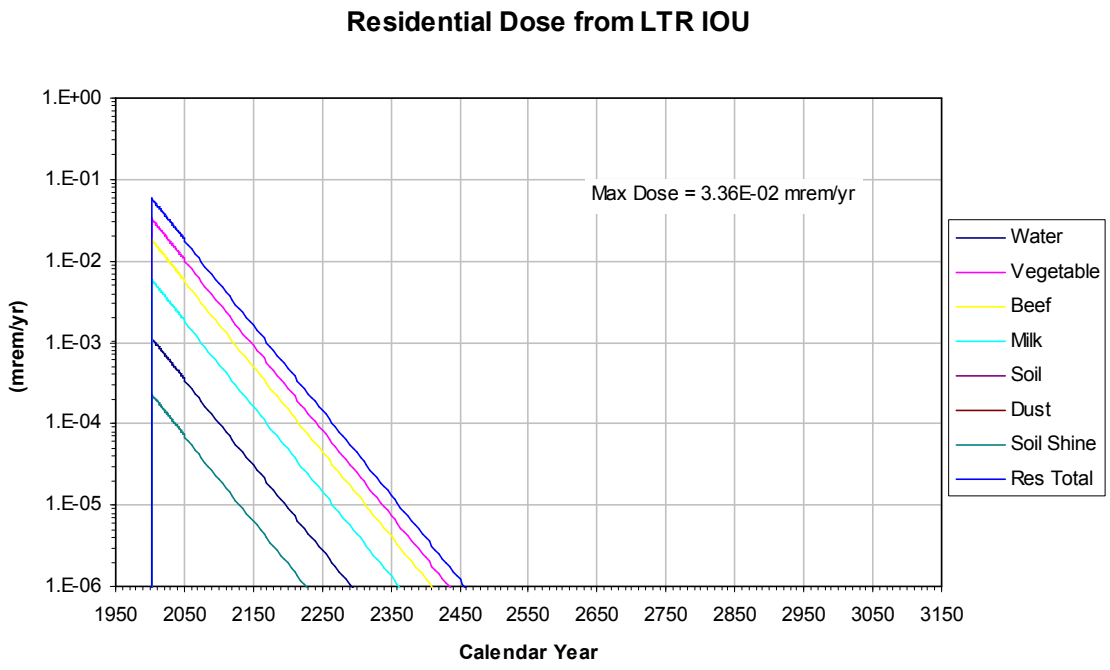


Figure E-63. LTR IOU Direct Residential Dose

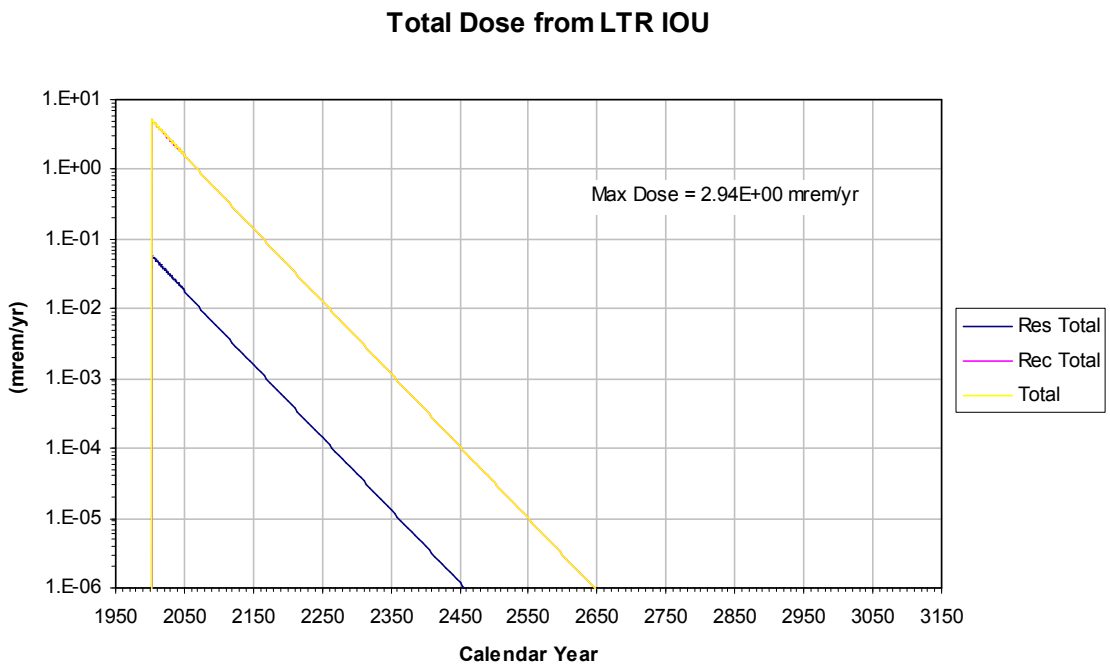


Figure E-64. LTR IOU Total Direct Dose

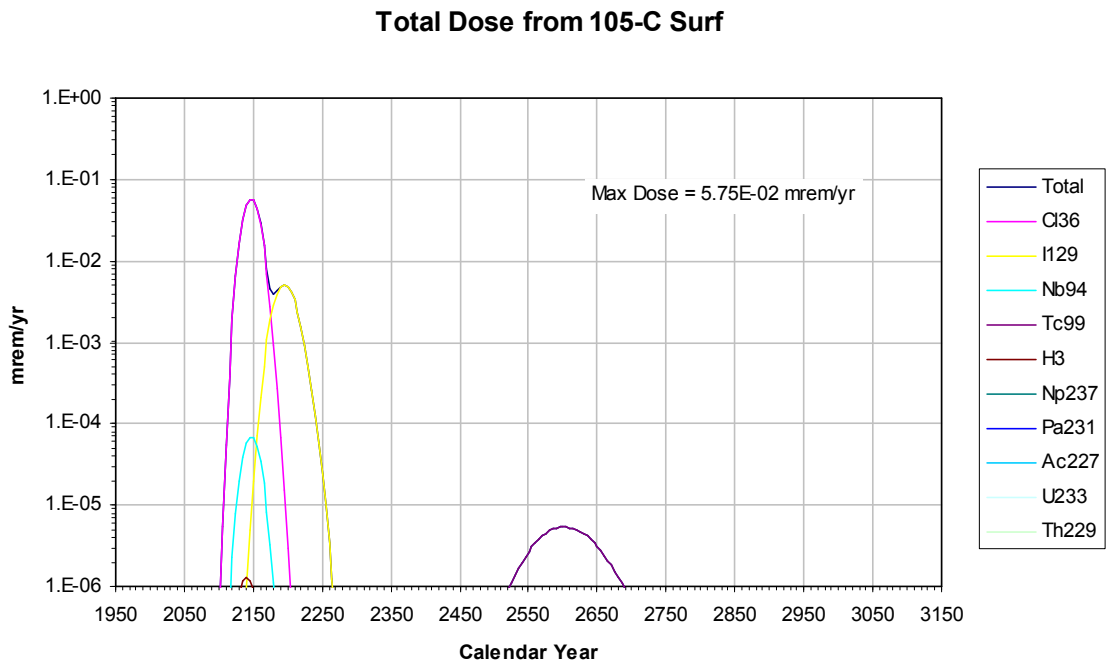


Figure E-65. 105-C Surf Direct Dose by Radionuclide

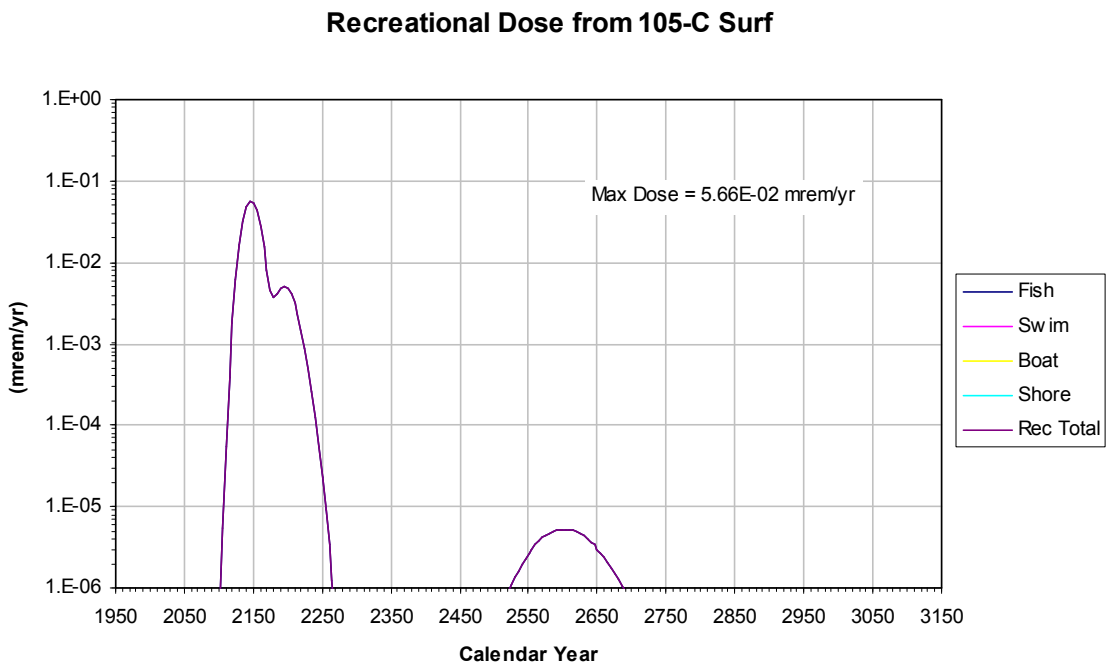


Figure E-66. 105-C Surf Recreational Dose

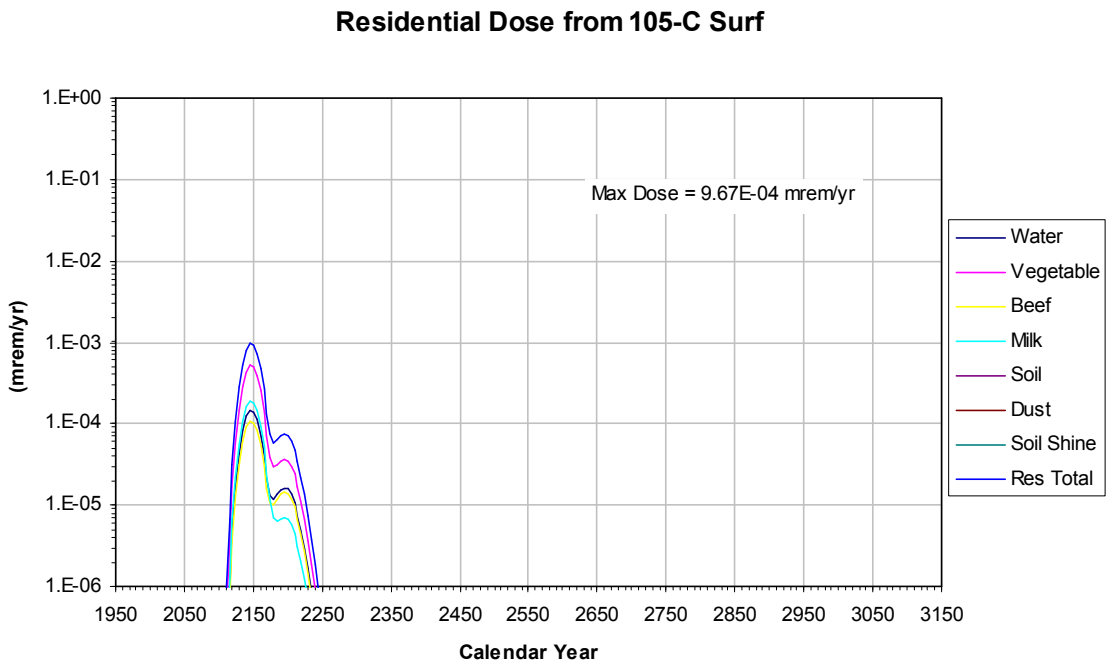


Figure E-67. 105-C Surf Direct Residential Dose

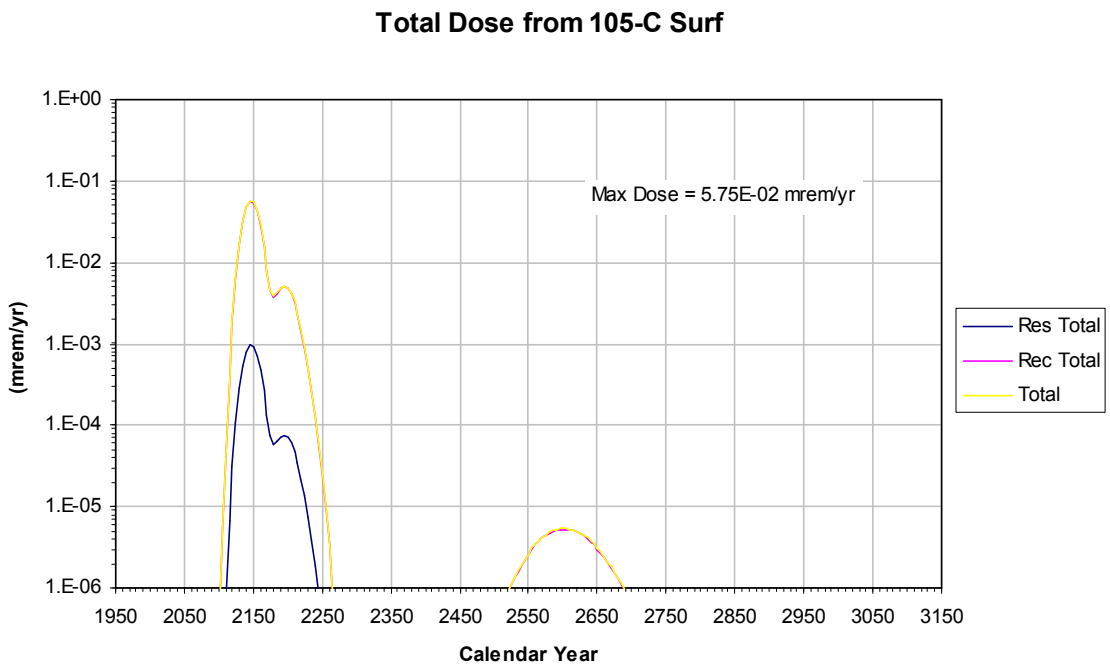


Figure E-68. 105-C Surf Total Direct Dose

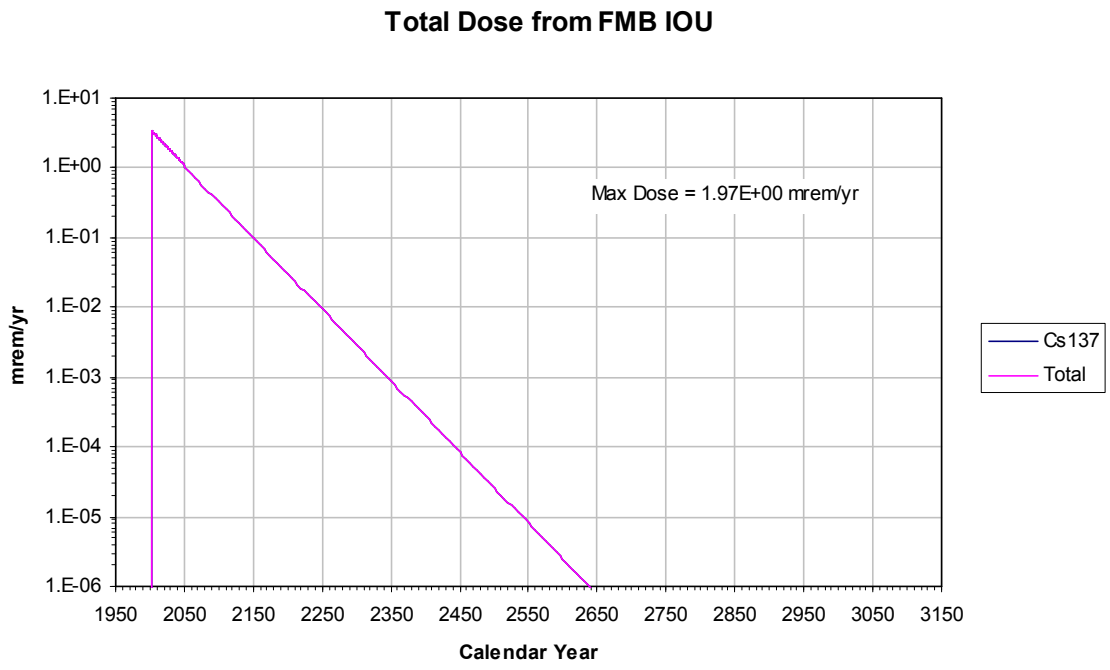


Figure E-69. FMB IOU Direct Dose by Radionuclide

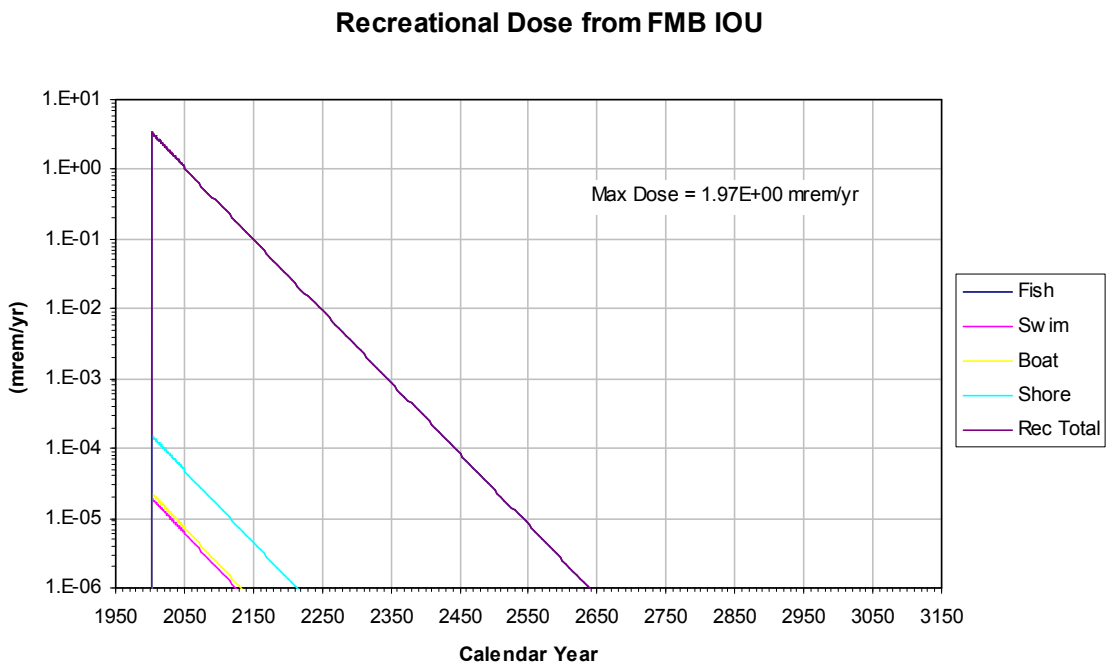


Figure E-70. FMB IOU Recreational Dose

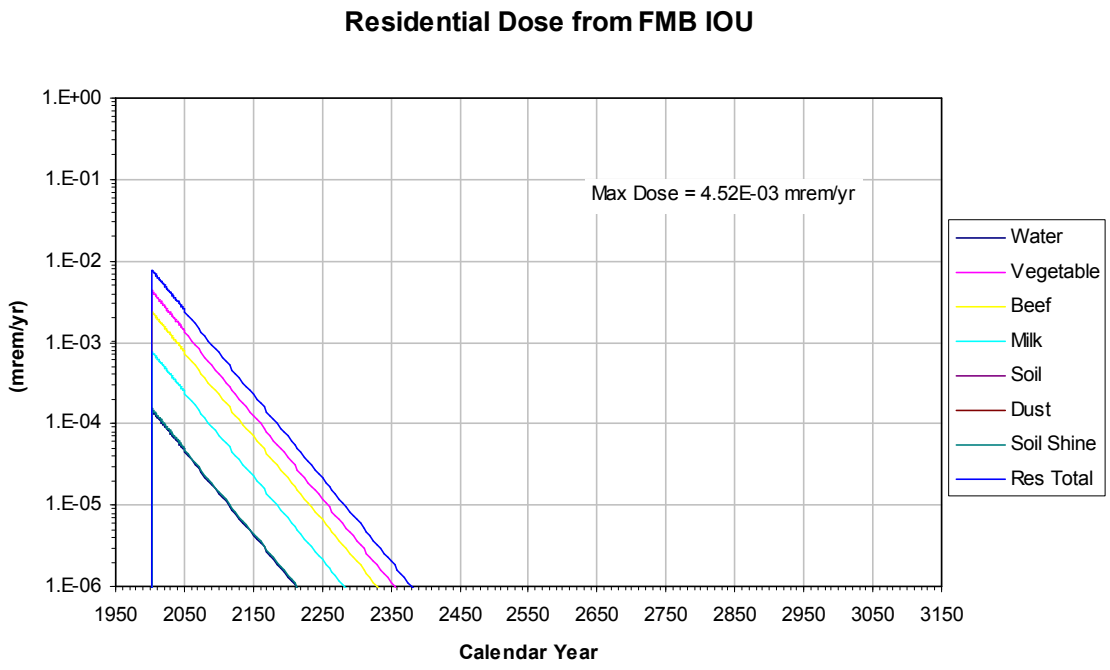


Figure E-71. FMB IOU Direct Residential Dose

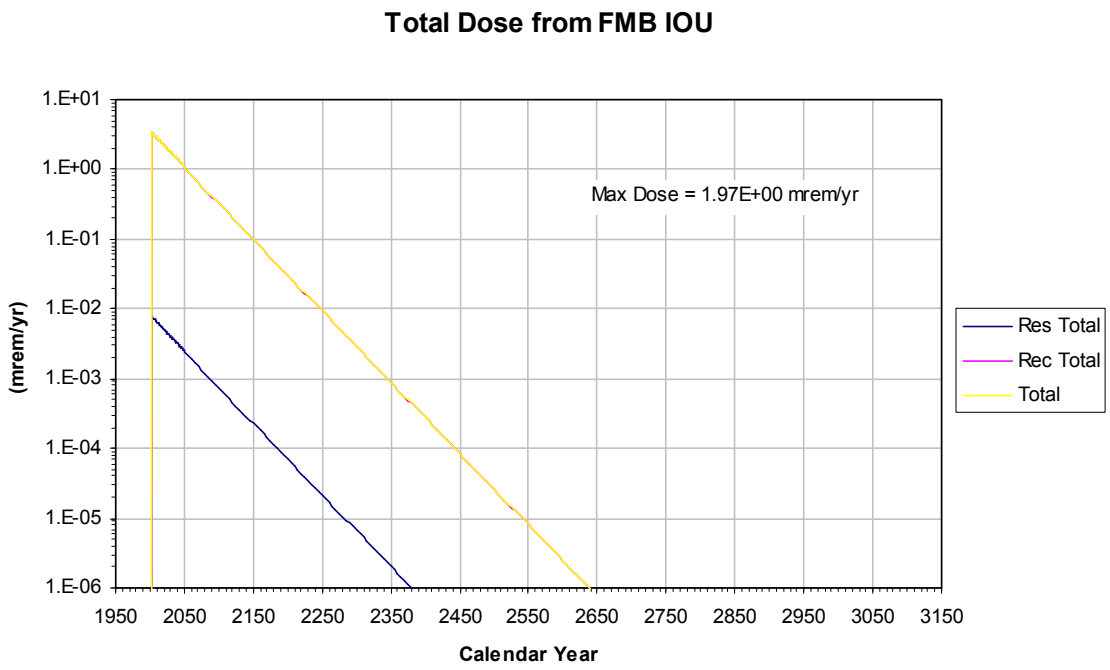


Figure E-72. FMB IOU Total Direct Dose

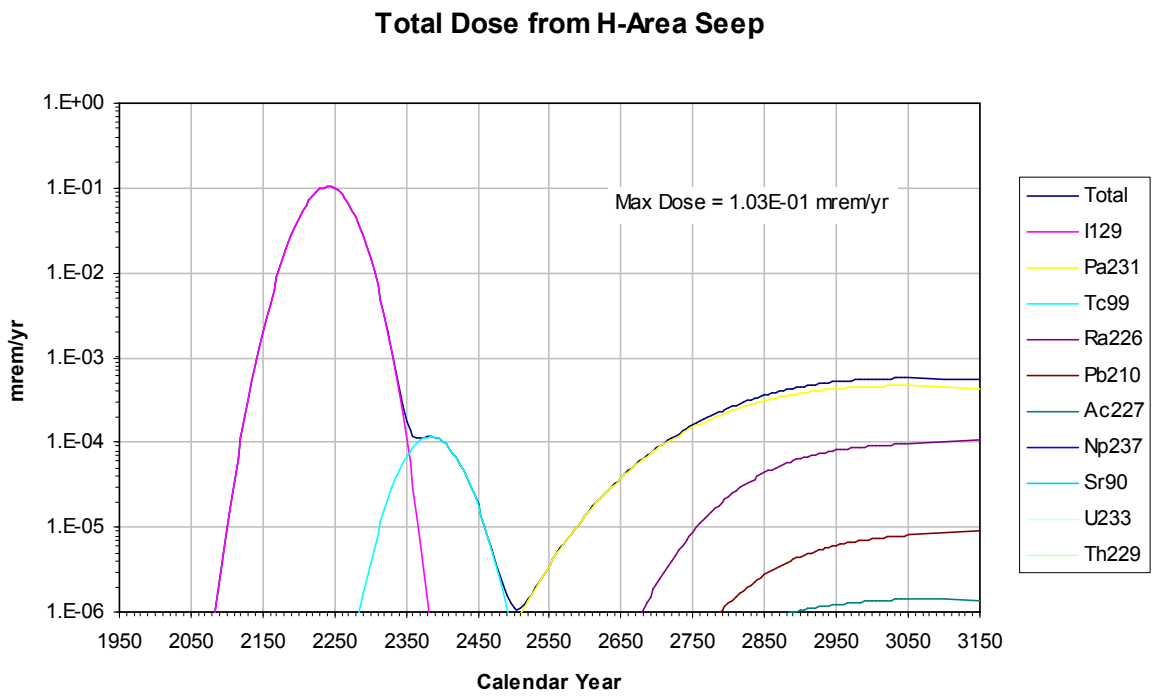


Figure E-73. H-Area Seep Direct Dose by Radionuclide

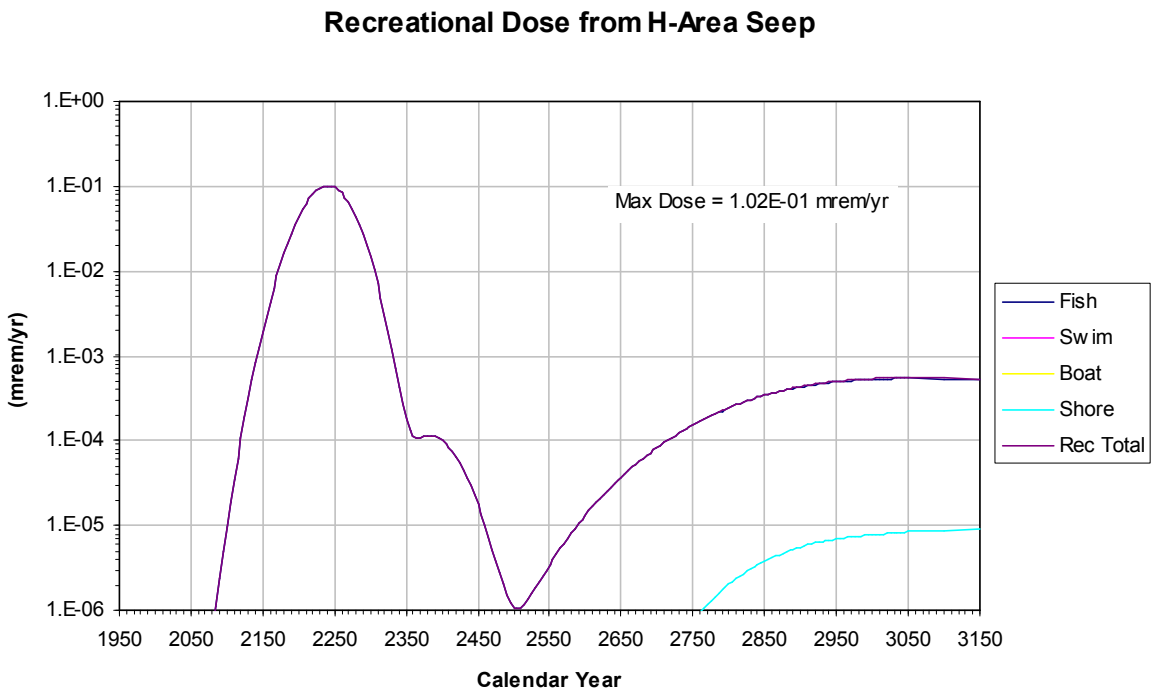


Figure E-74. H-Area Seep Recreational Dose

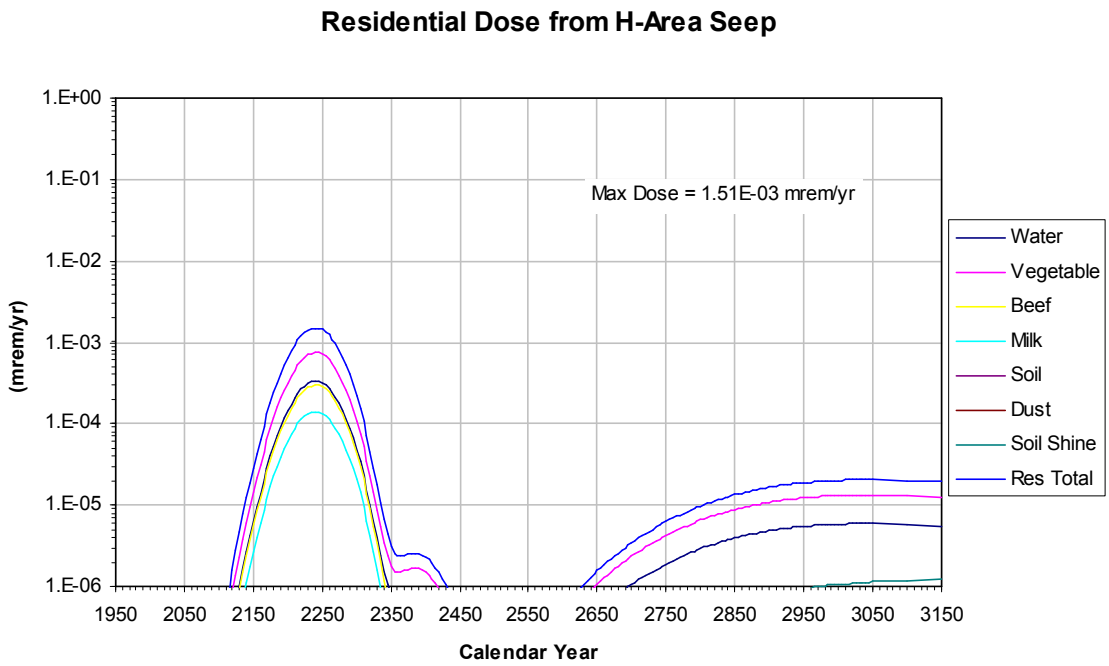


Figure E-75. H-Area Seep Direct Residential Dose

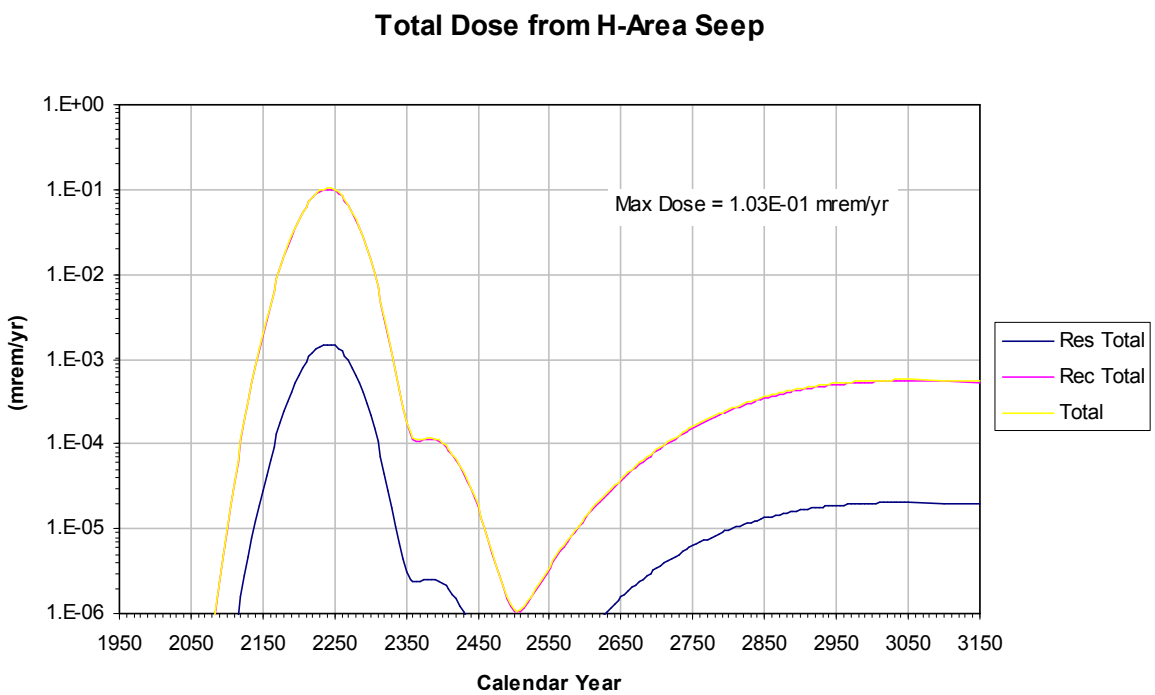


Figure E-76. H-Area Seep Total Direct Dose

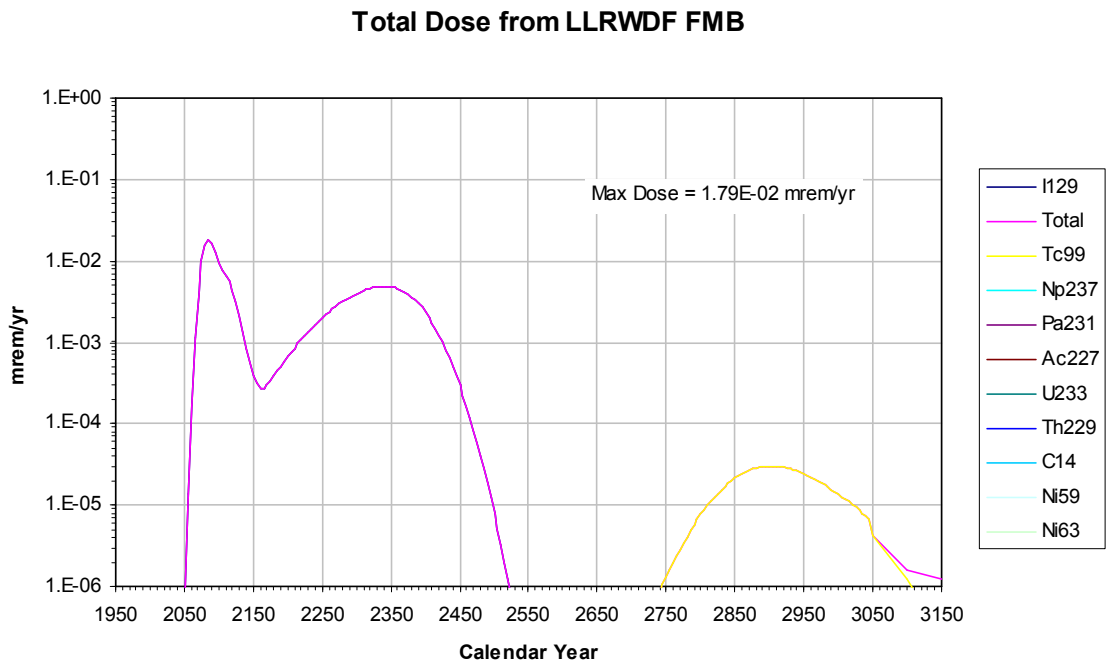


Figure E-77. LLRWDF FMB Direct Dose by Radionuclide

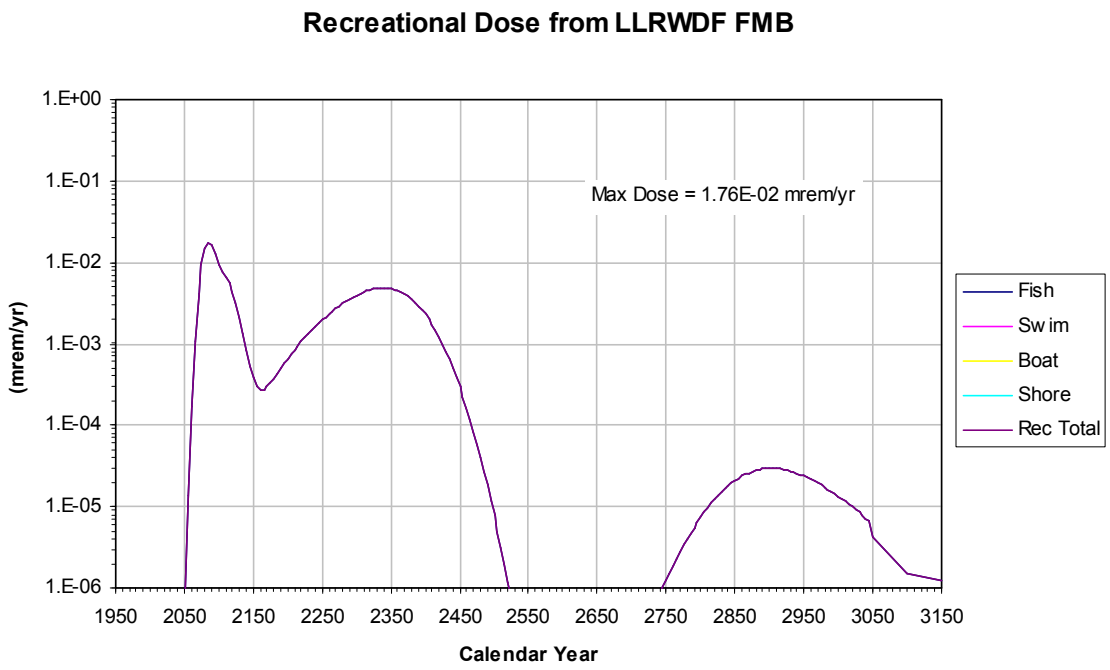


Figure E-78. LLRWDF FMB Recreational Dose

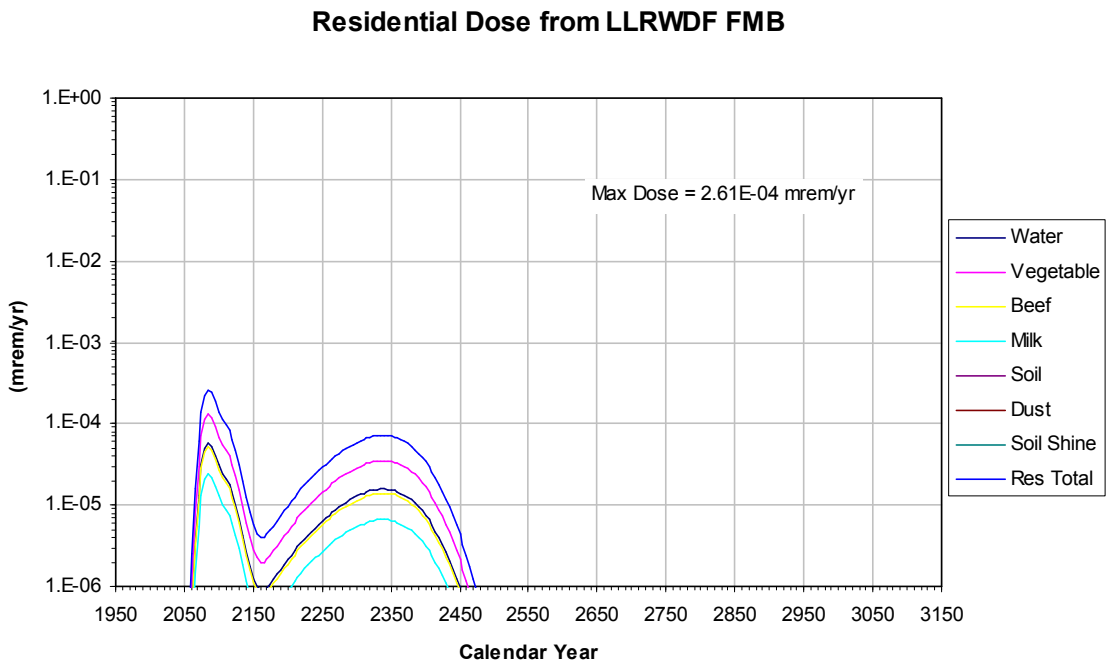


Figure E-79. LLRWDF FMB Direct Residential Dose

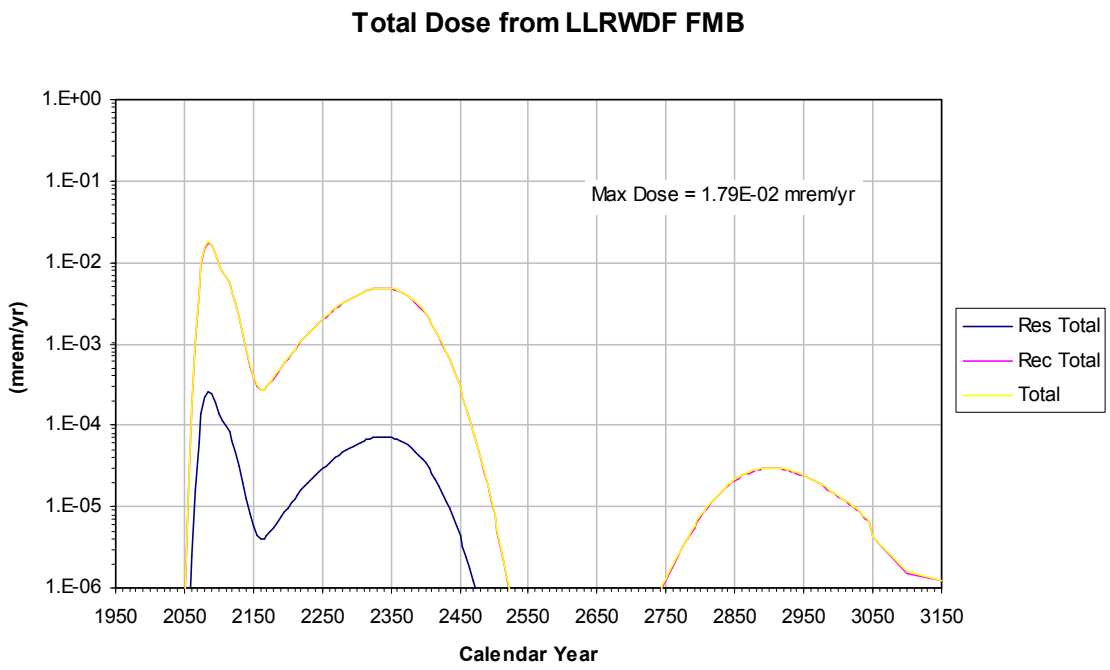


Figure E-80. LLRWDF FMB Total Direct Dose

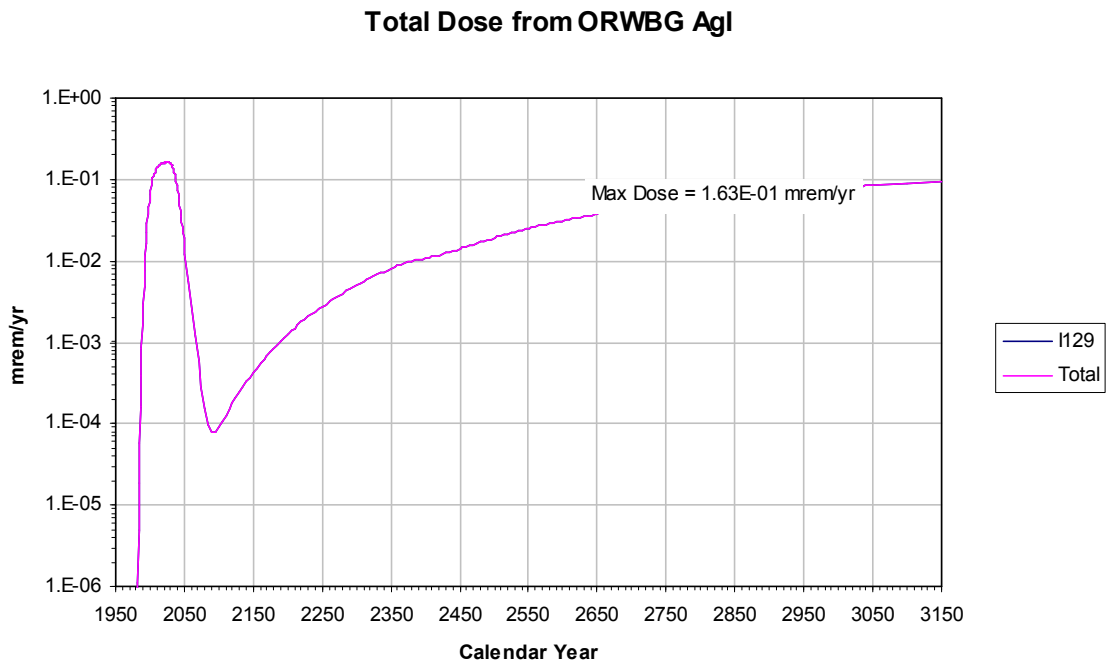


Figure E-81. ORWBG AgI Direct Dose by Radionuclide

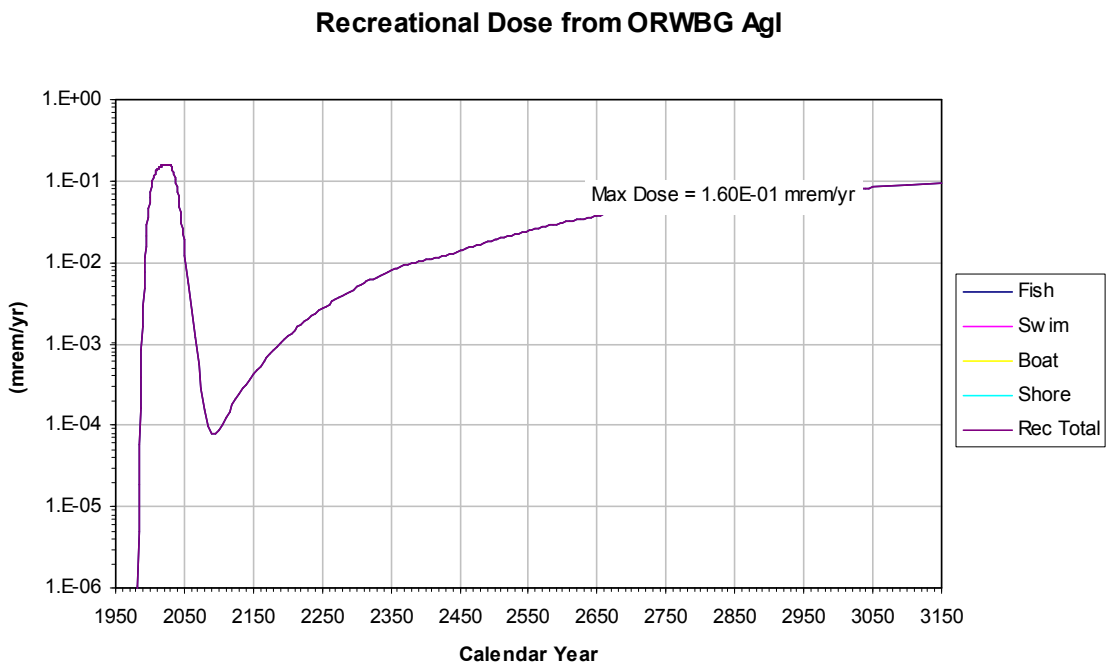


Figure E-82. ORWBG AgI Recreational Dose

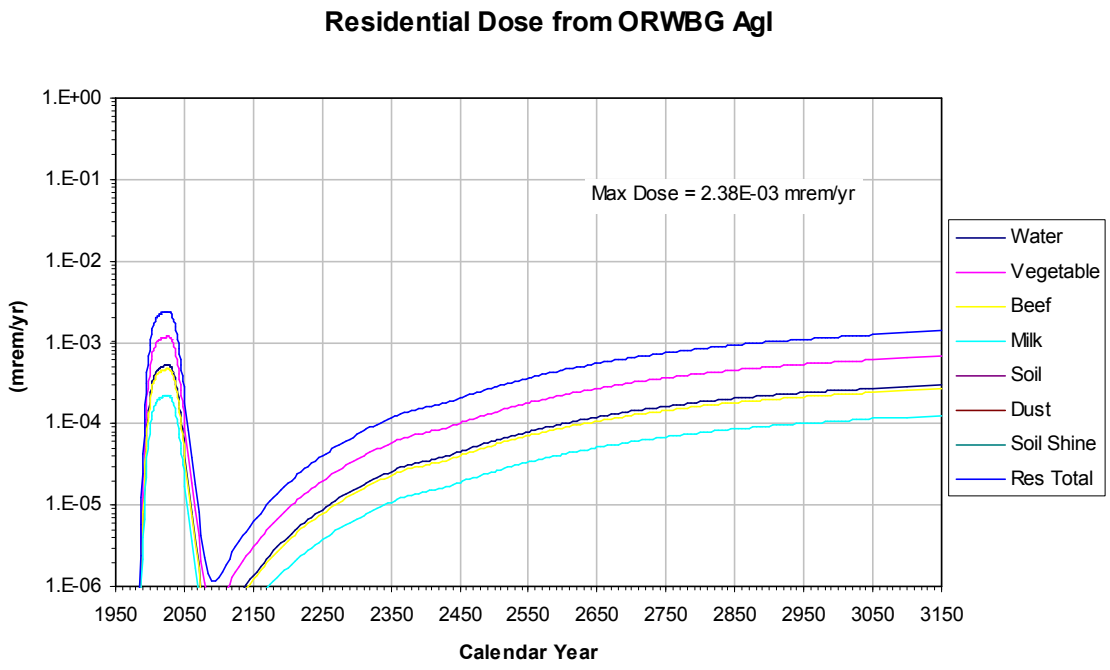


Figure E-83. ORWBG AgI Direct Residential Dose

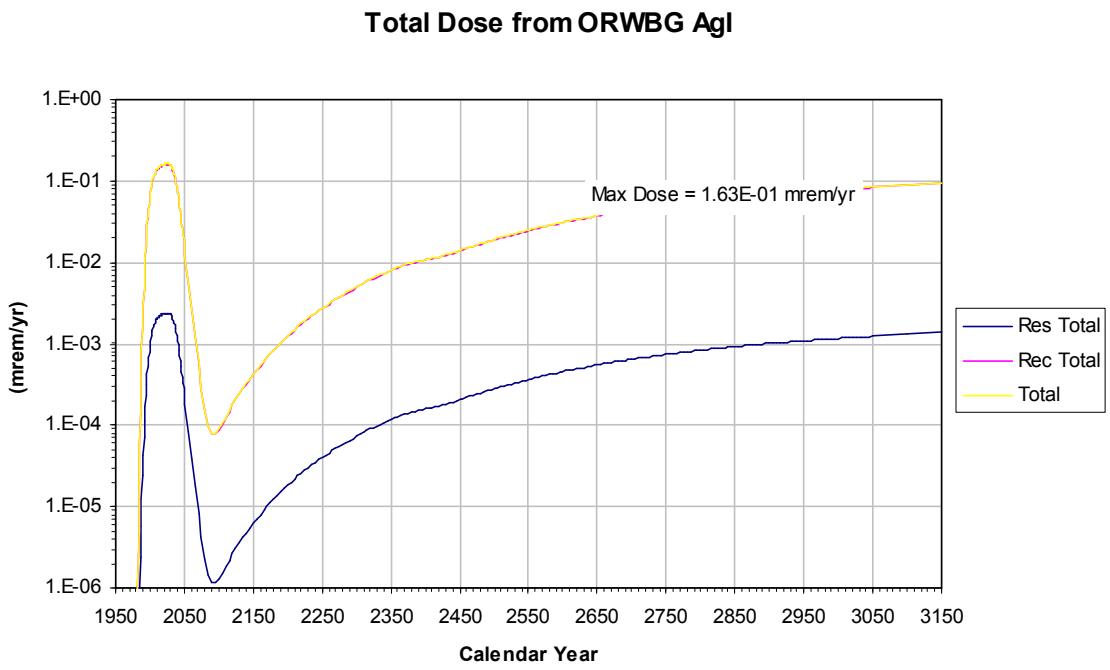


Figure E-84. ORWBG AgI Total Direct Dose

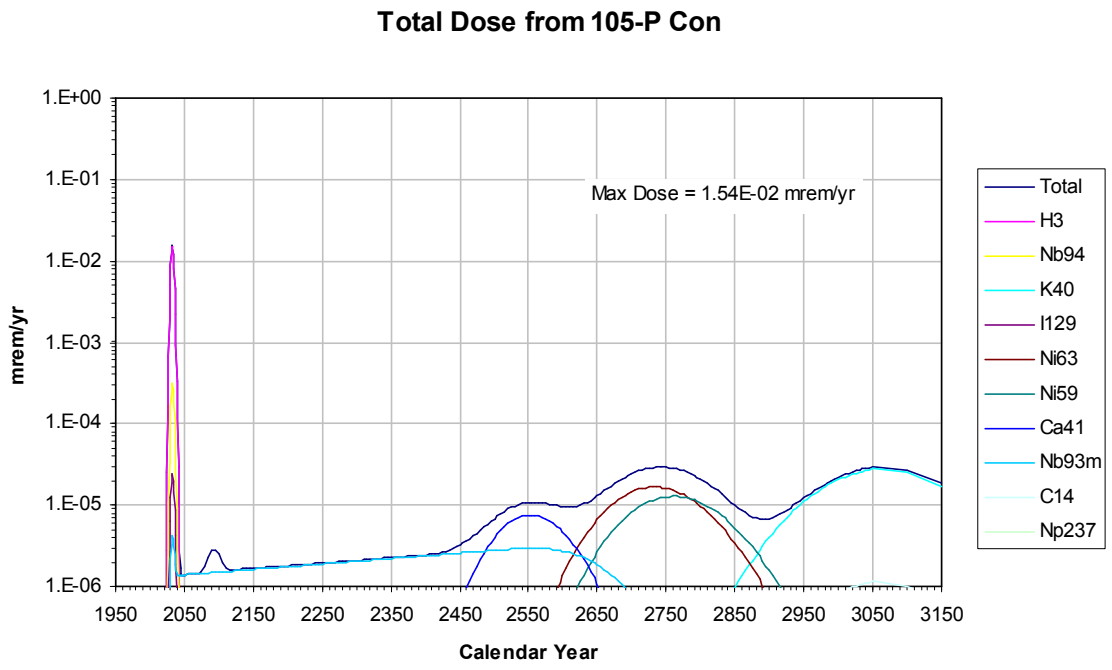


Figure E-85. 105-P Con Direct Dose by Radionuclide

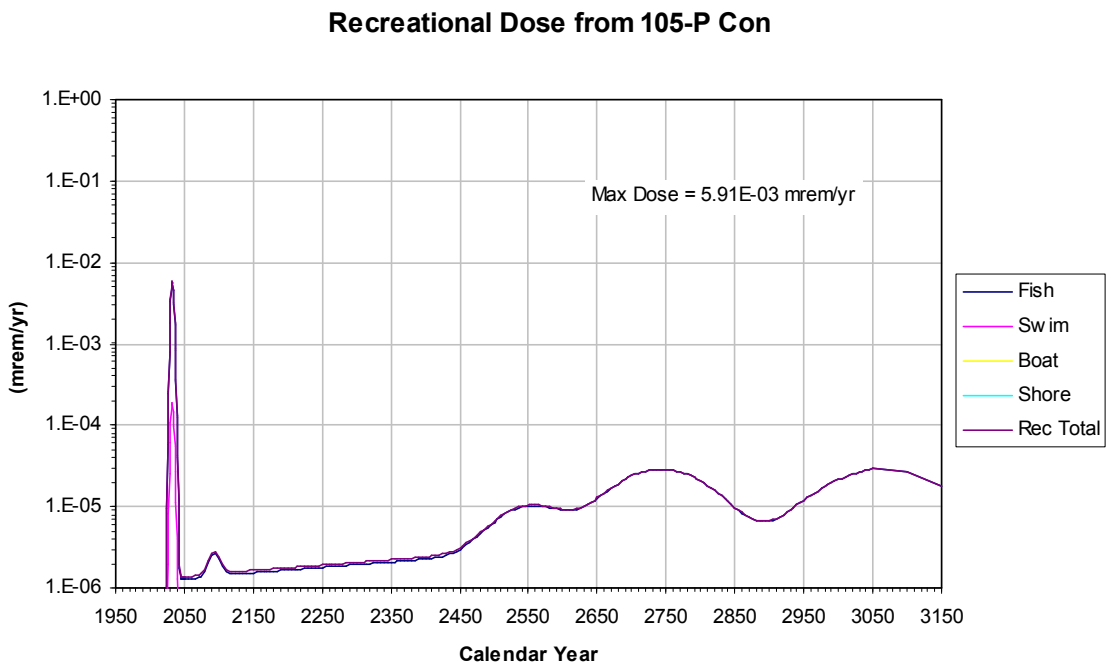


Figure E-86. 105-P Con Recreational Dose

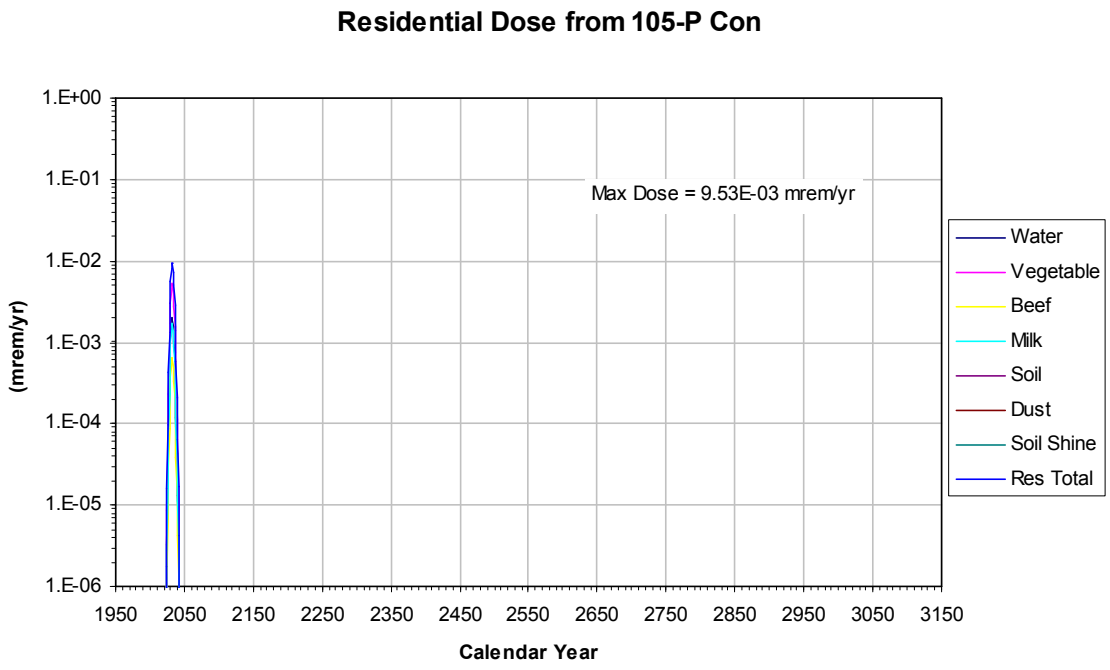


Figure E-87. 105-P Con Direct Residential Dose

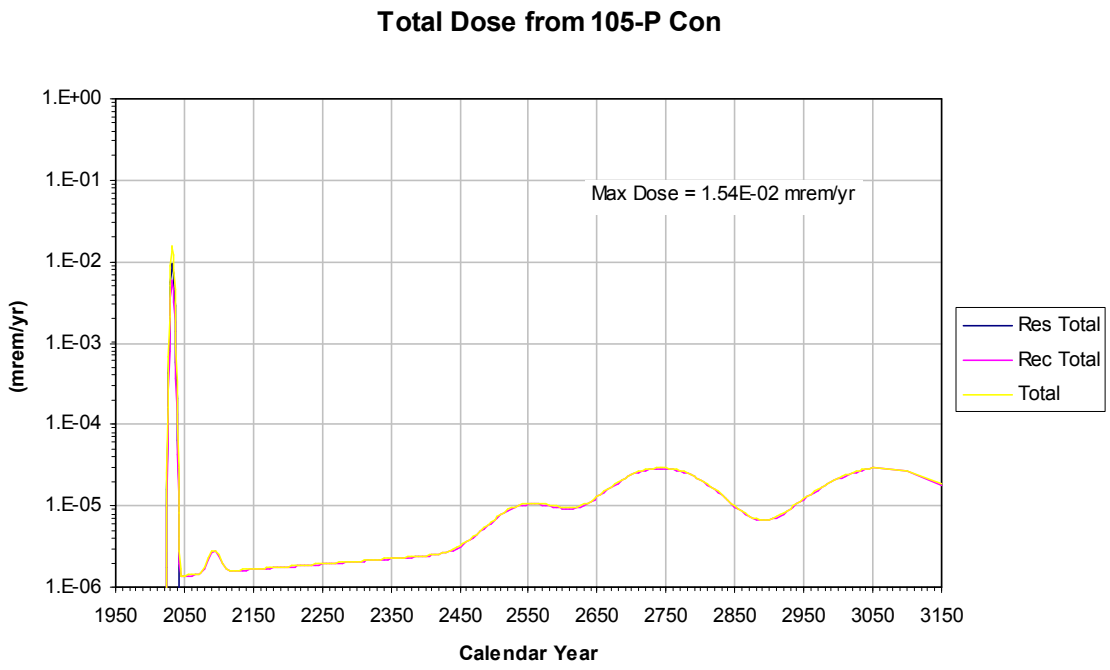


Figure E-88. 105-P Con Total Direct Dose

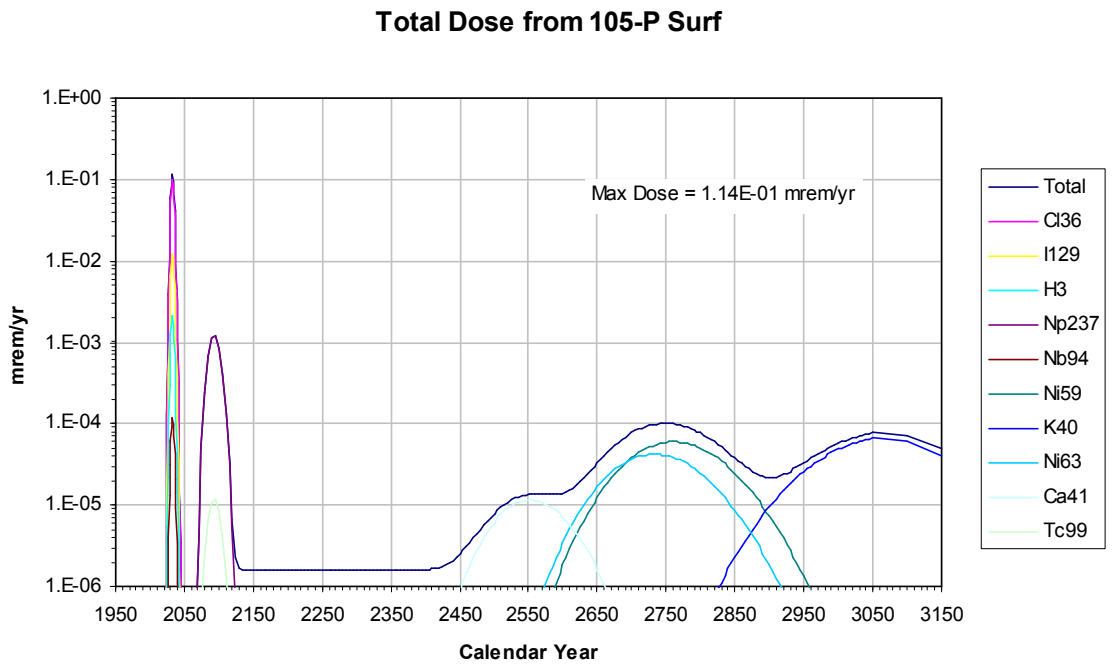


Figure E-89. 105-P Surf Direct Dose by Radionuclide

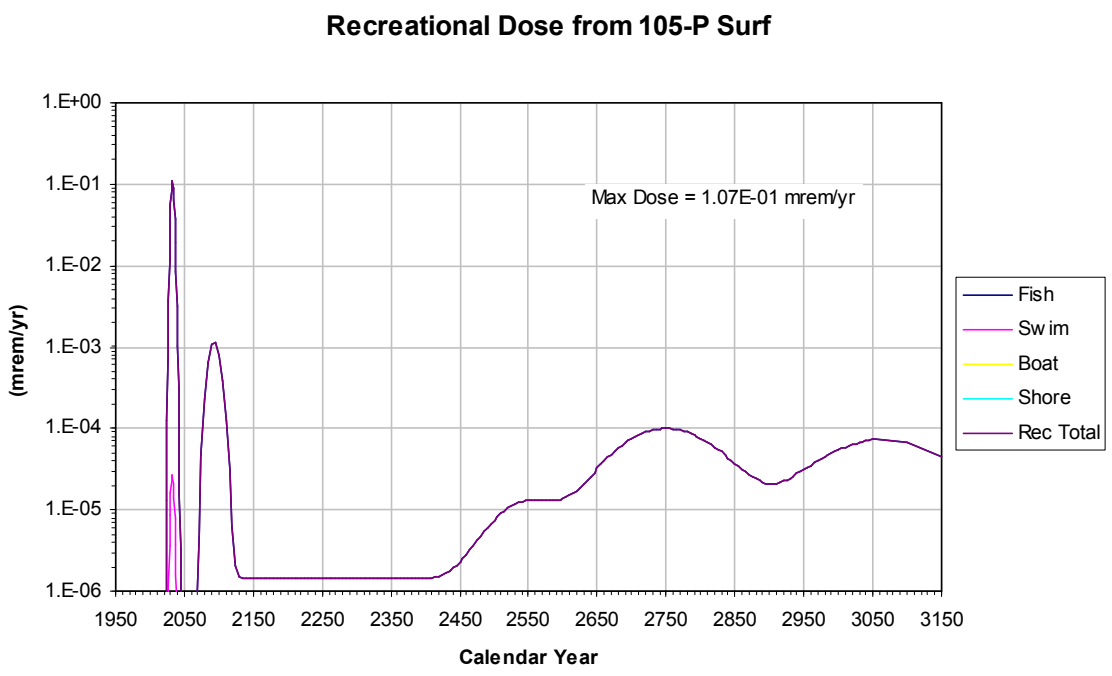


Figure E-90. 105-P Surf Recreational Dose

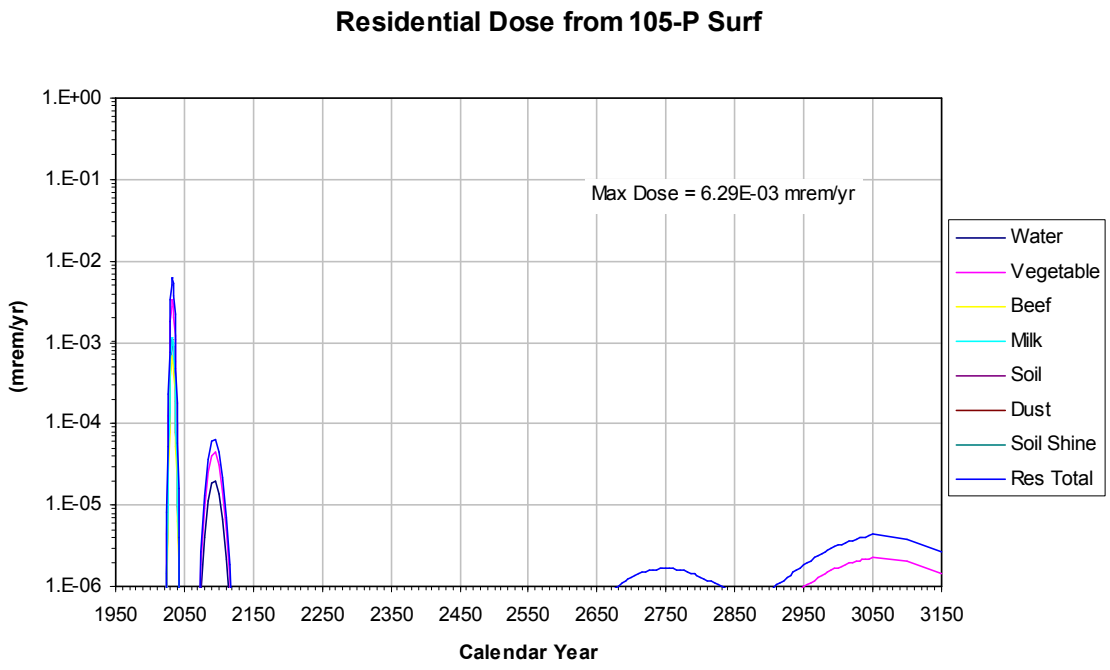


Figure E-91. 105-P Surf Direct Residential Dose

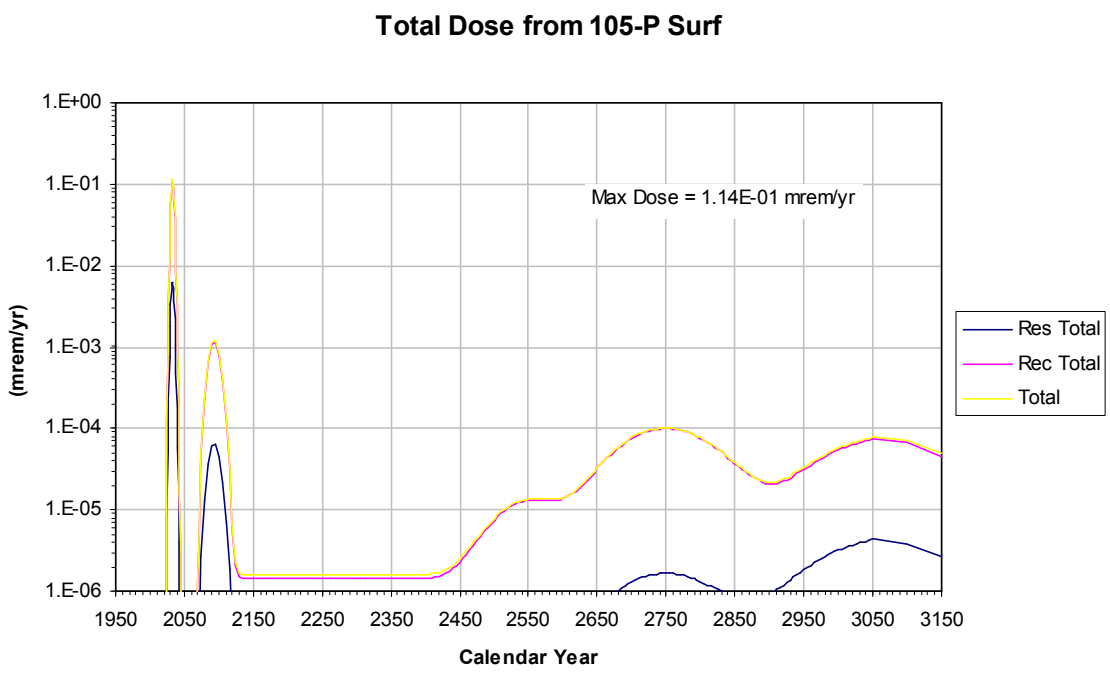


Figure E-92. 105-P Surf Total Direct Dose

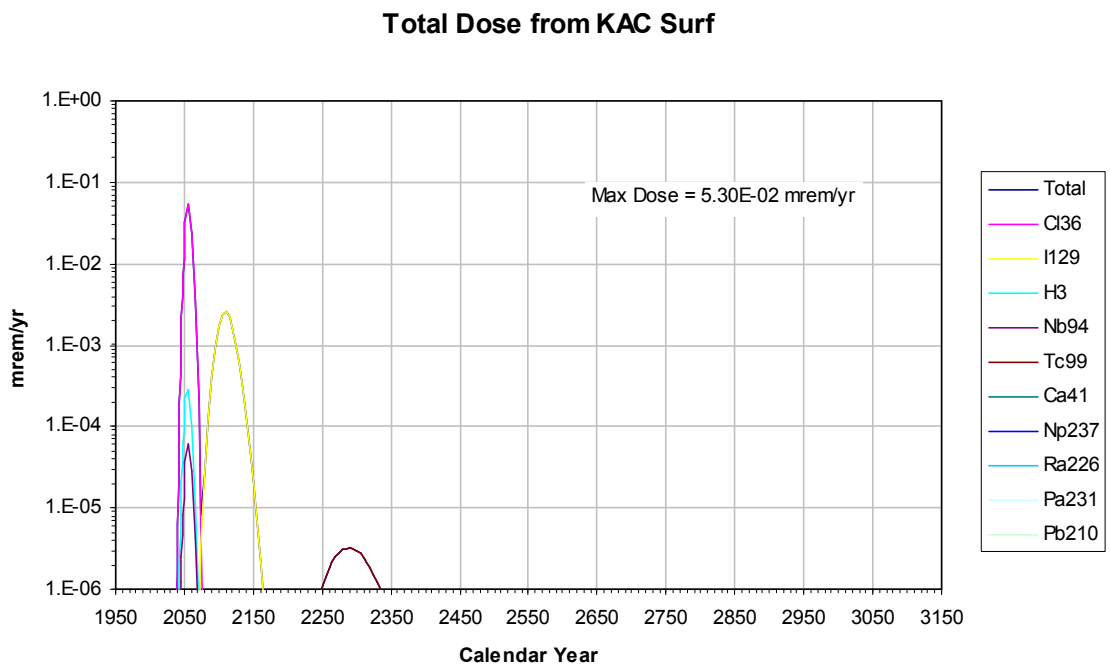


Figure E-93. KAC Surf Direct Dose by Radionuclide

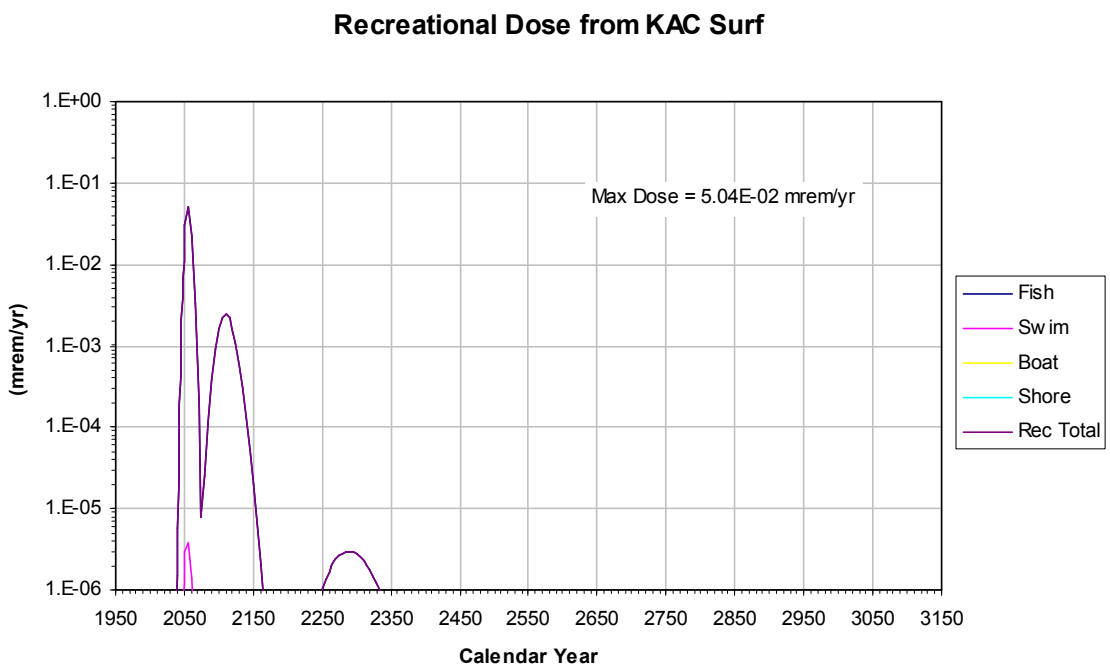


Figure E-94. KAC Surf Recreational Dose

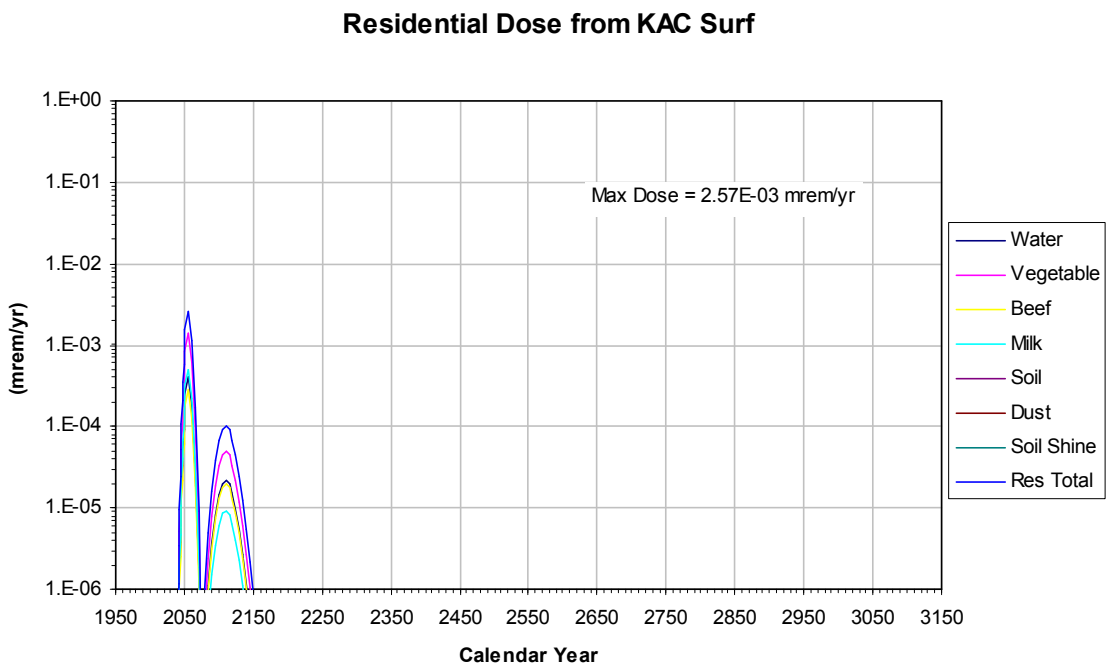


Figure E-95. KAC Surf Direct Residential Dose

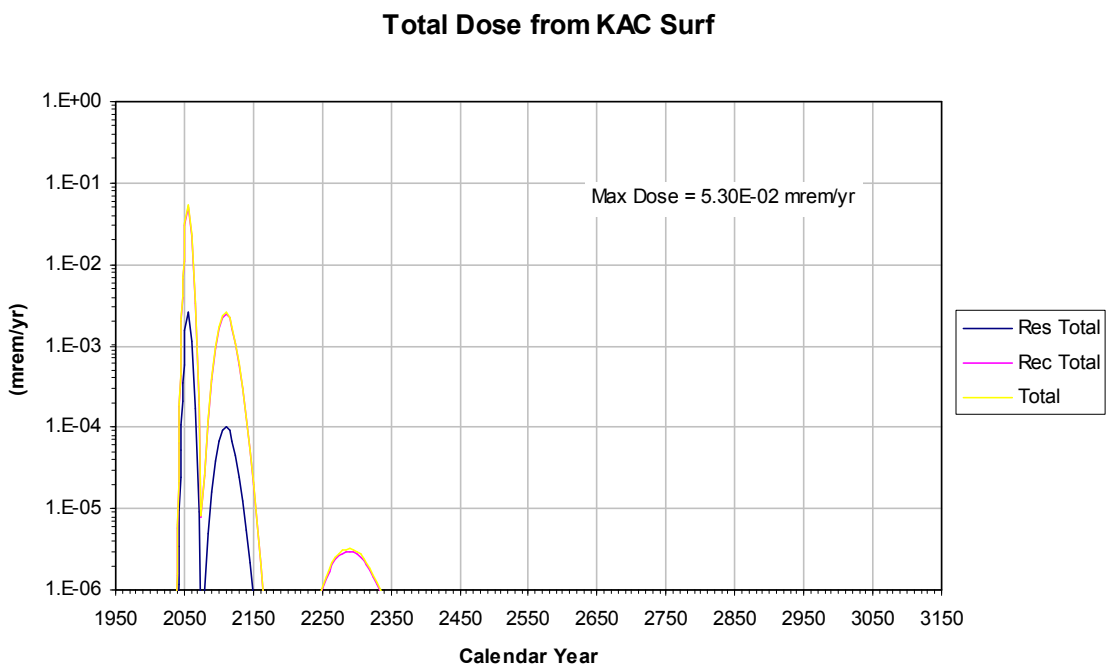


Figure E-96. KAC Surf Total Direct Dose

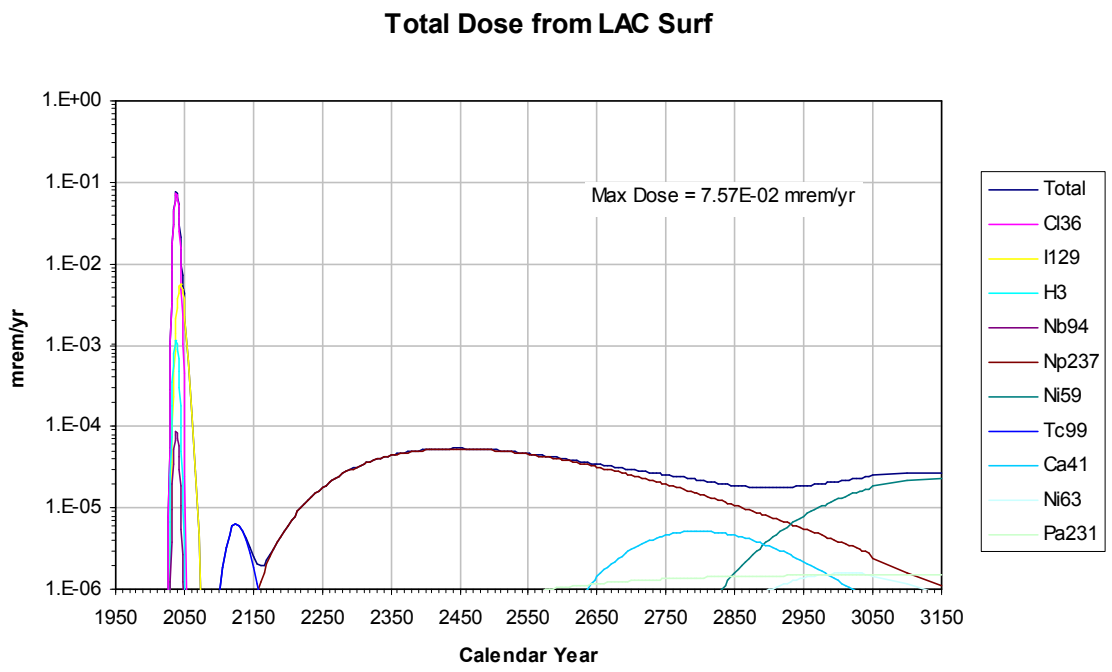


Figure E-97. LAC Surf Direct Dose by Radionuclide

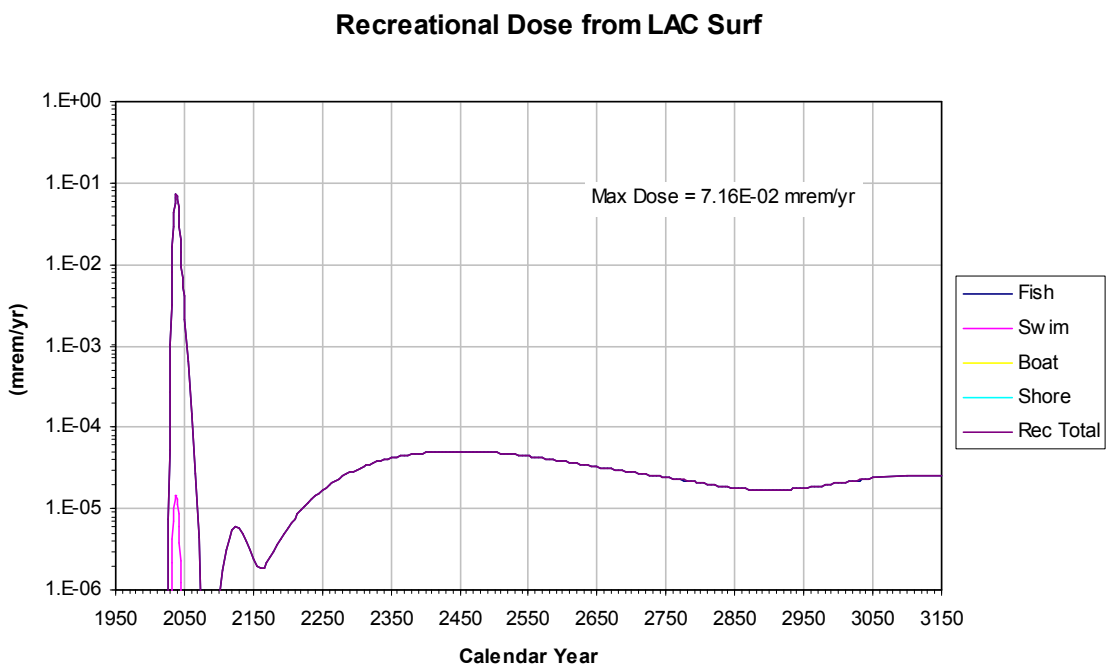


Figure E-98. LAC Surf Recreational Dose

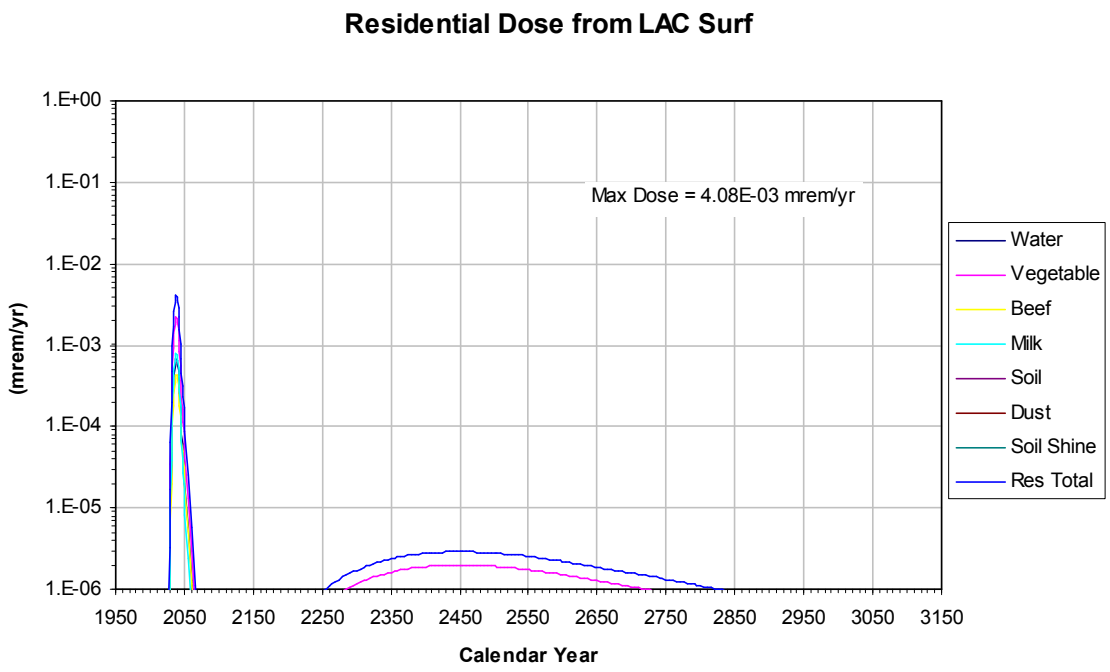


Figure E-99. LAC Surf Direct Residential Dose

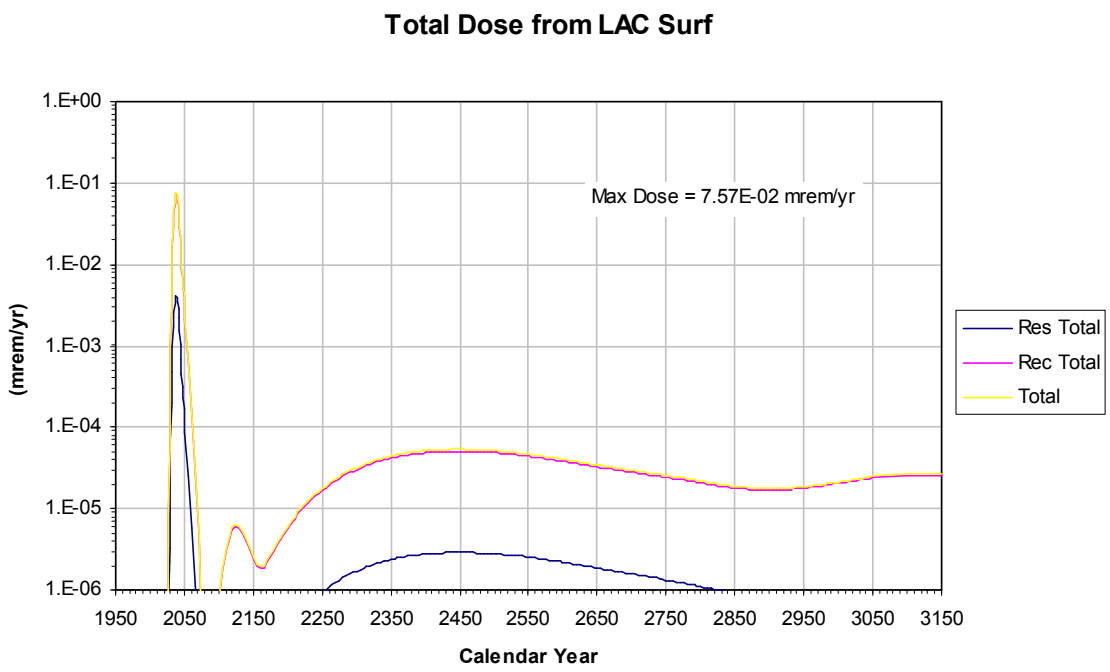


Figure E-100. LAC Surf Total Direct Dose

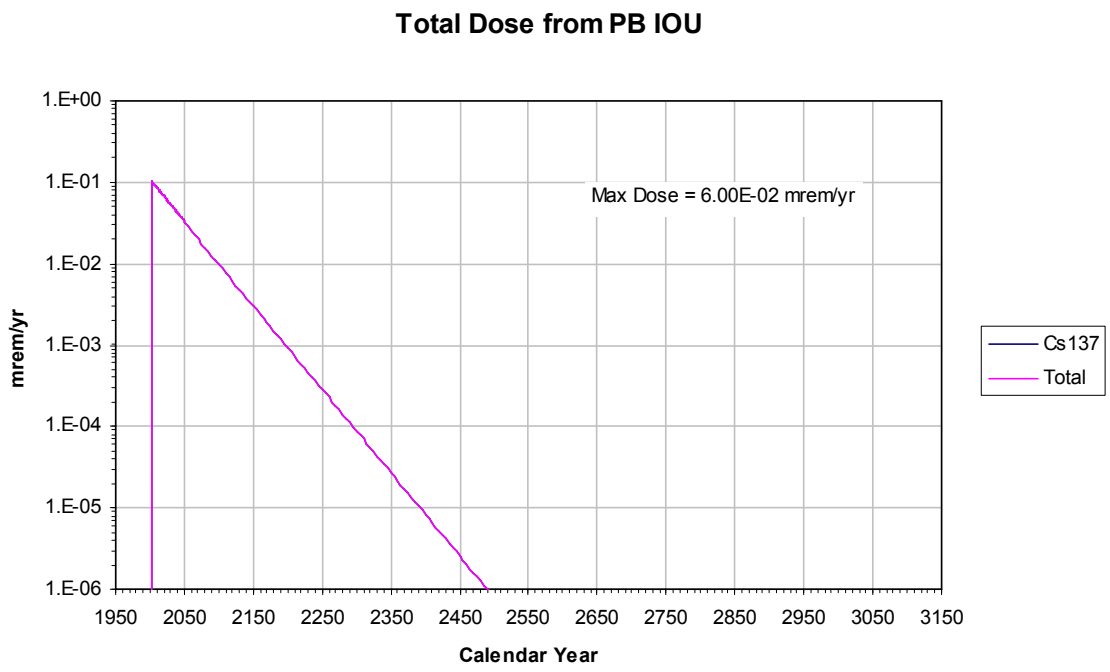


Figure E-101. PB IOU Direct Dose by Radionuclide

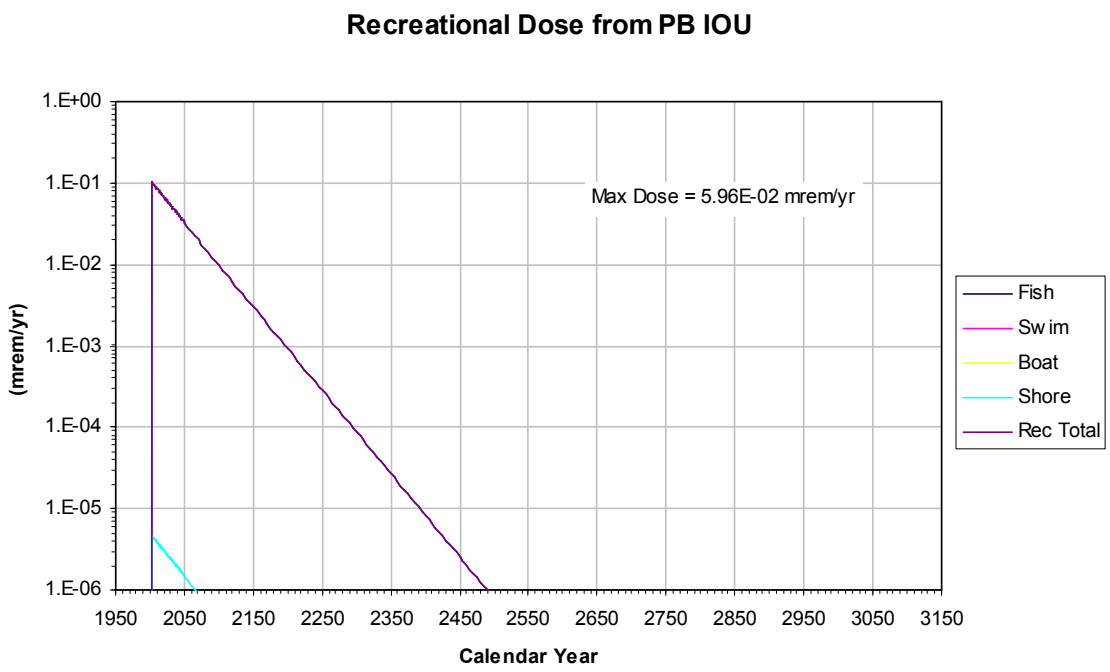


Figure E-102. PB IOU Recreational Dose

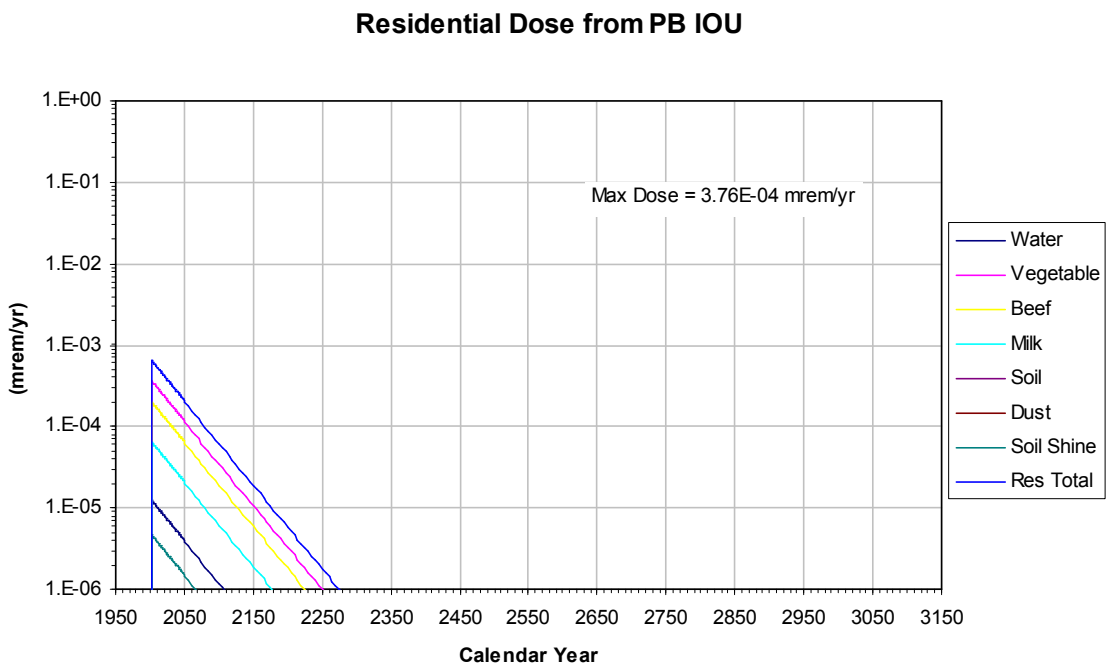


Figure E-103. PB IOU Direct Residential Dose

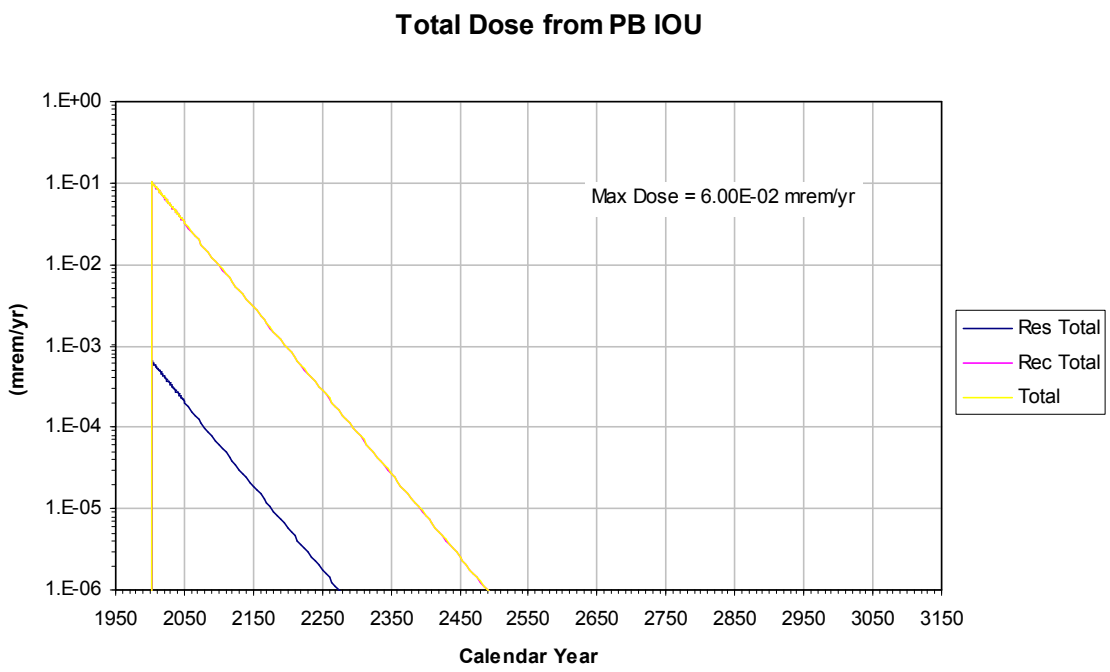


Figure E-104. PB IOU Total Direct Dose

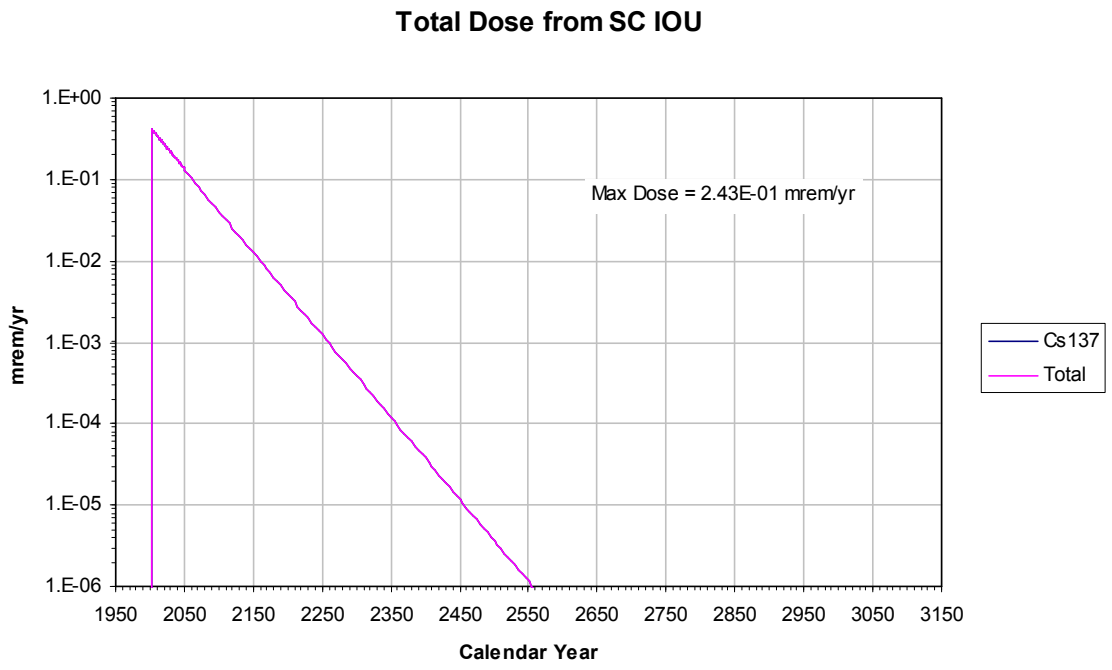


Figure E-105. SC IOU Direct Dose by Radionuclide

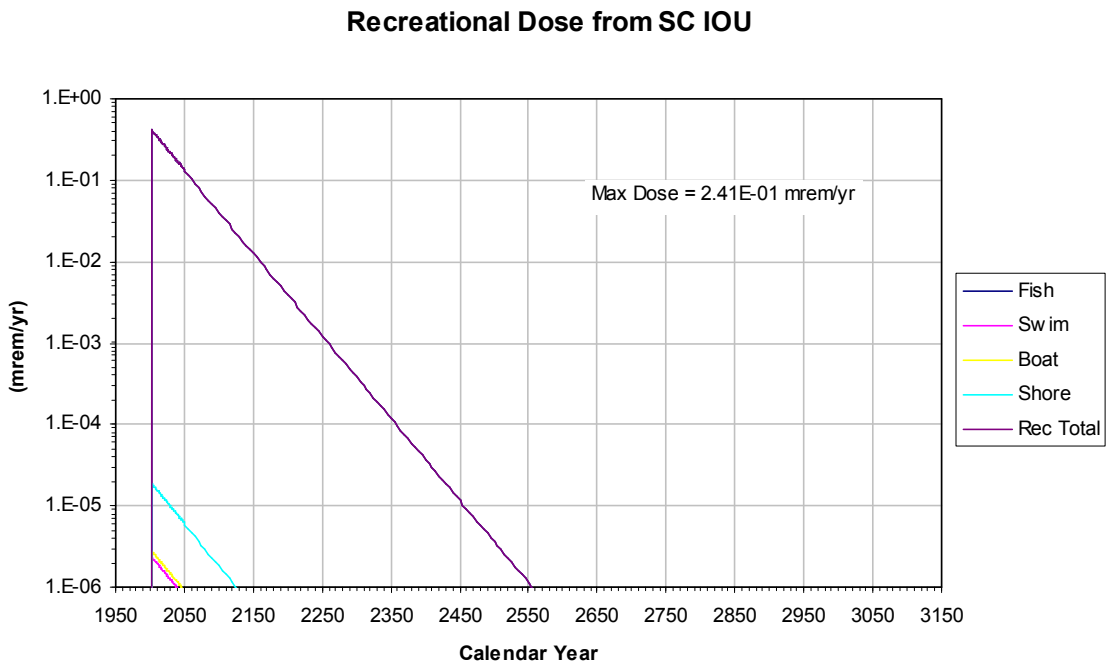


Figure E-106. SC IOU Recreational Dose

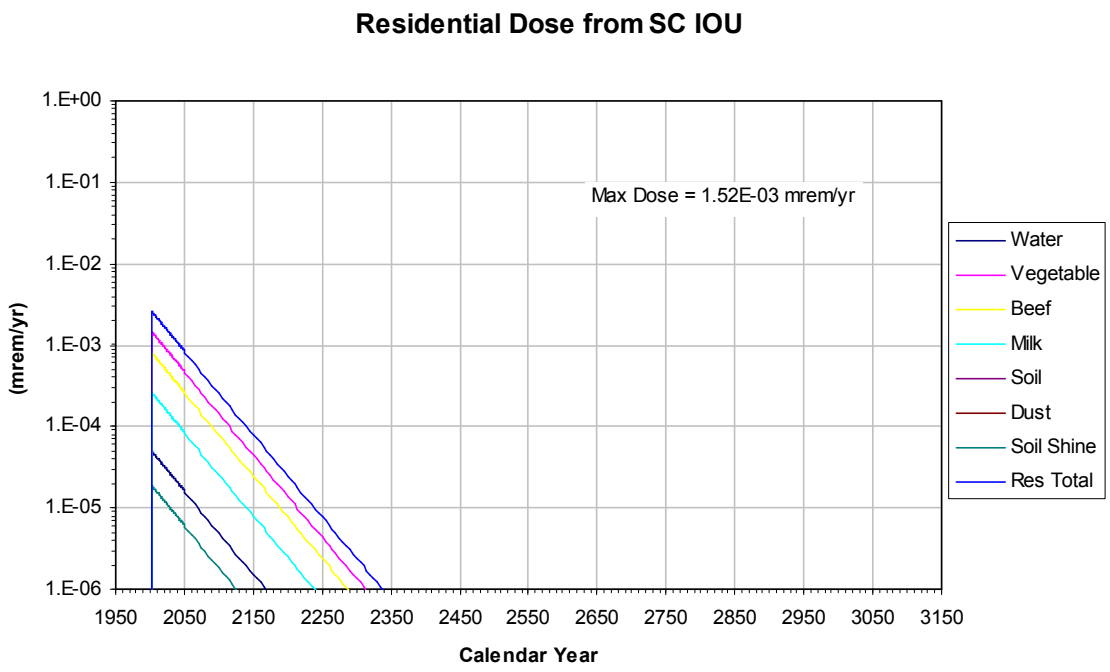


Figure E-107. SC IOU Direct Residential Dose

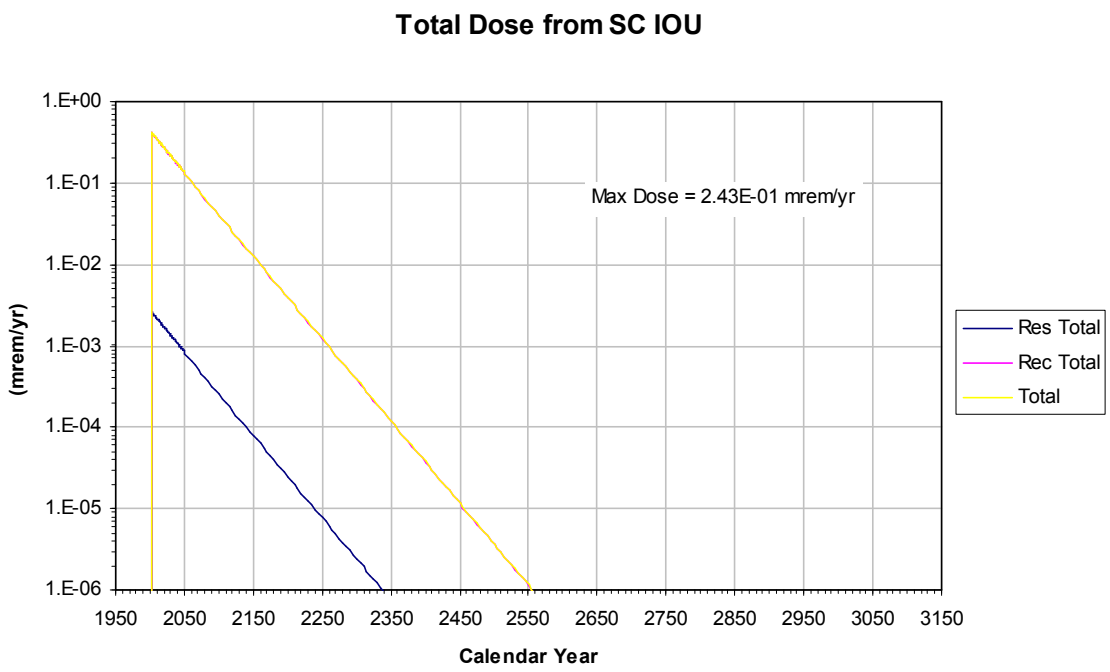


Figure E-108. SC IOU Total Direct Dose

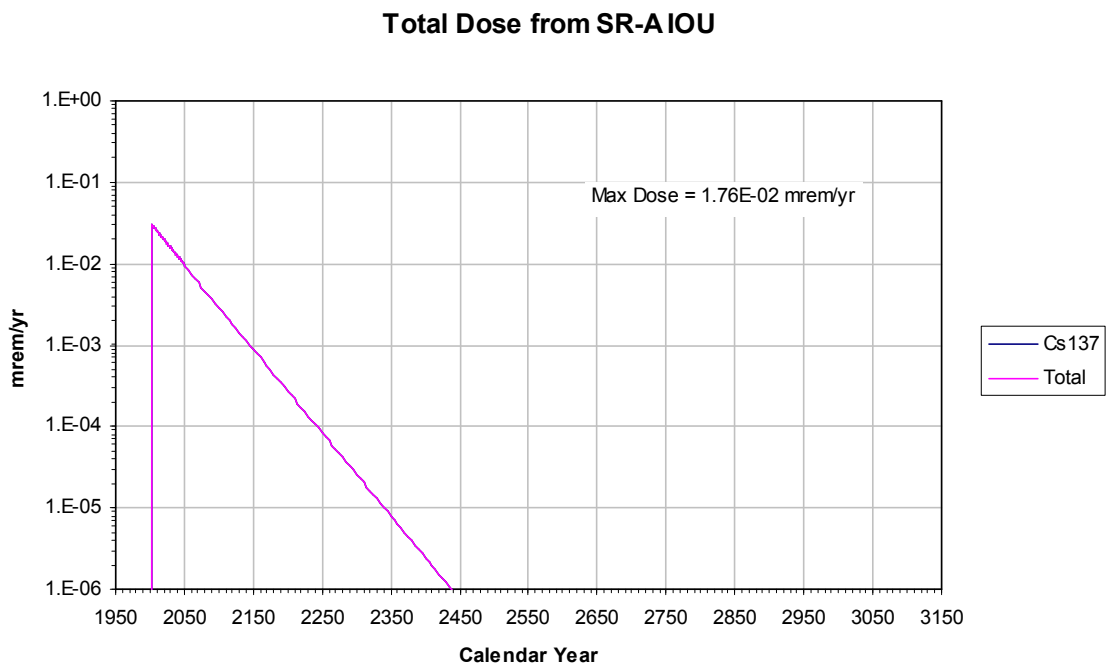


Figure E-109. SR-A IOU Direct Dose by Radionuclide

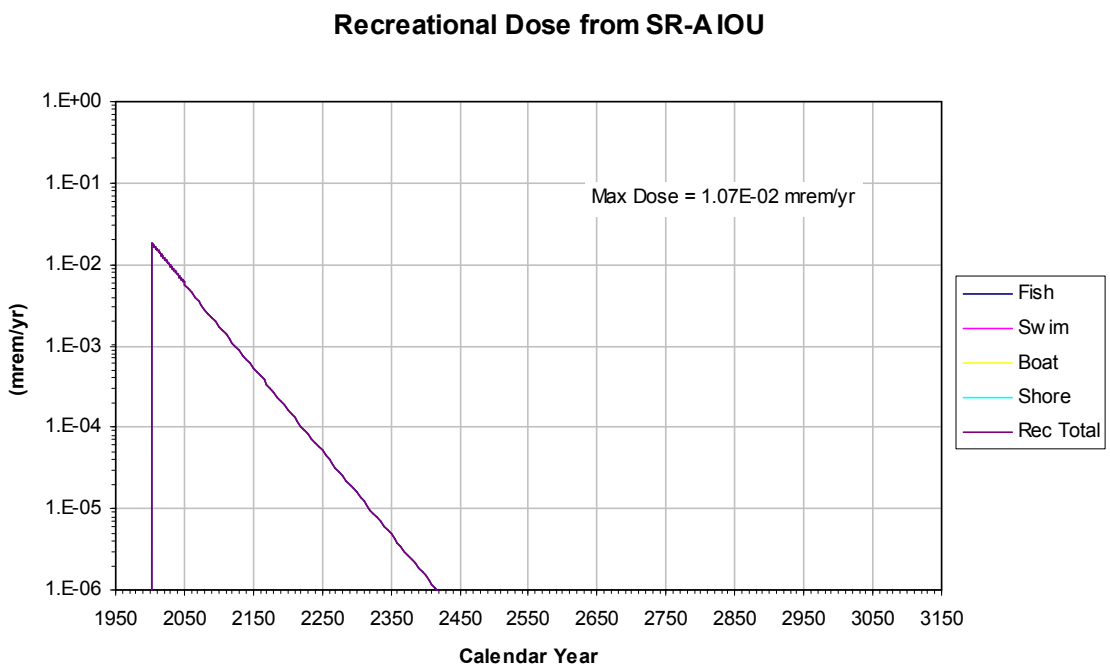


Figure E-110. SR-A IOU Recreational Dose

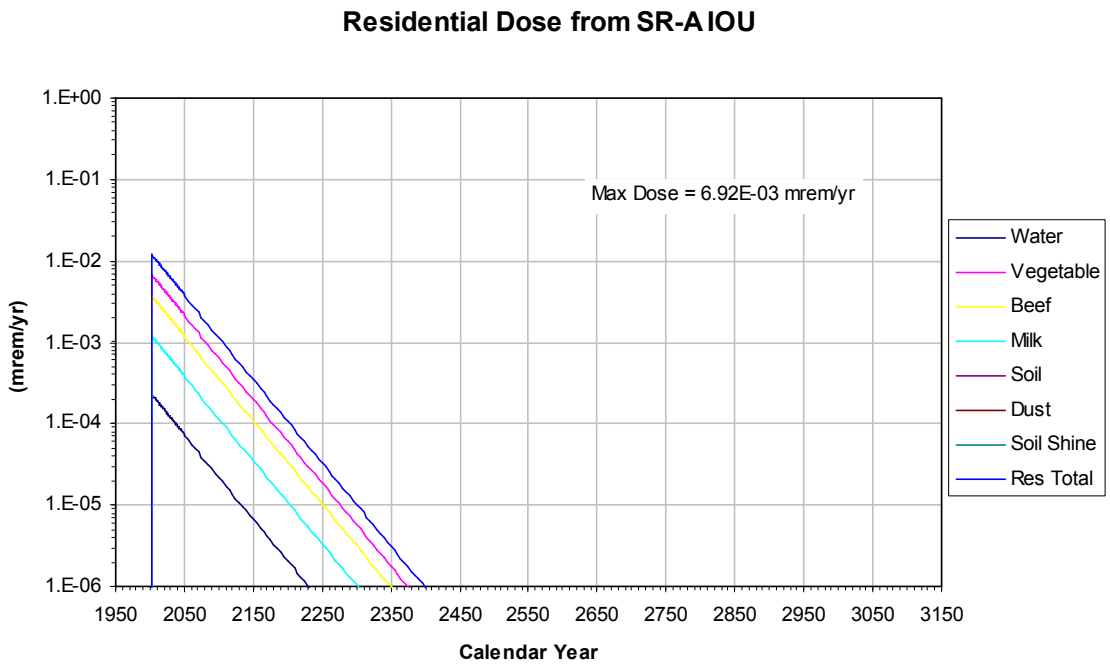


Figure E-111. SR-A IOU Direct Residential Dose

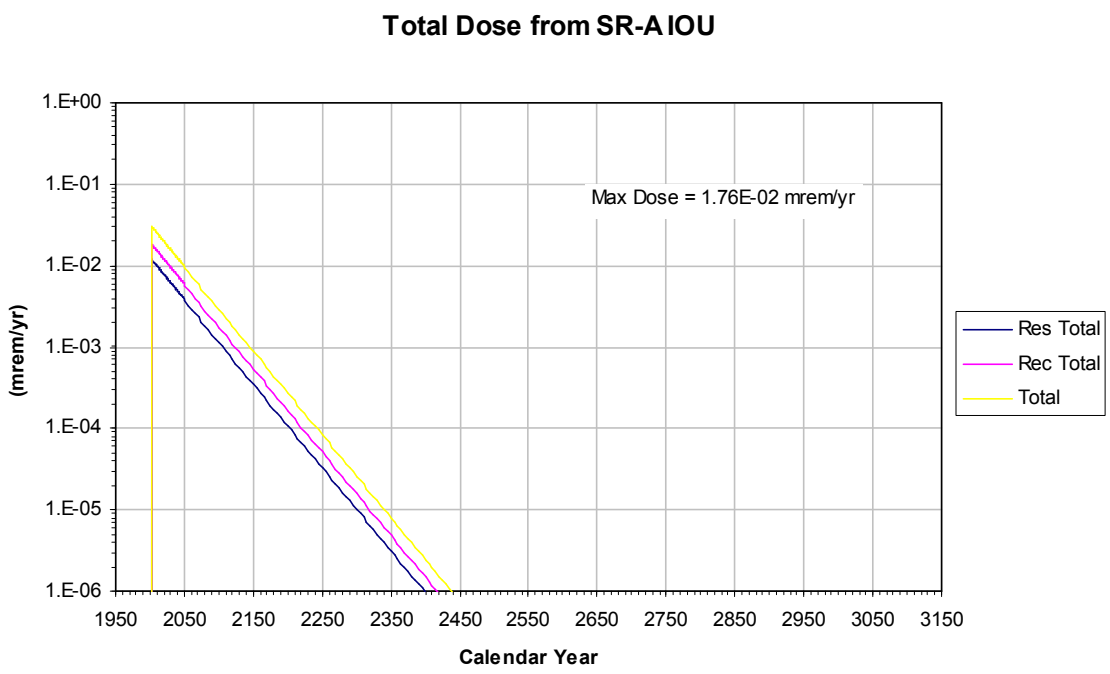


Figure E-112. SR-A IOU Total Direct Dose

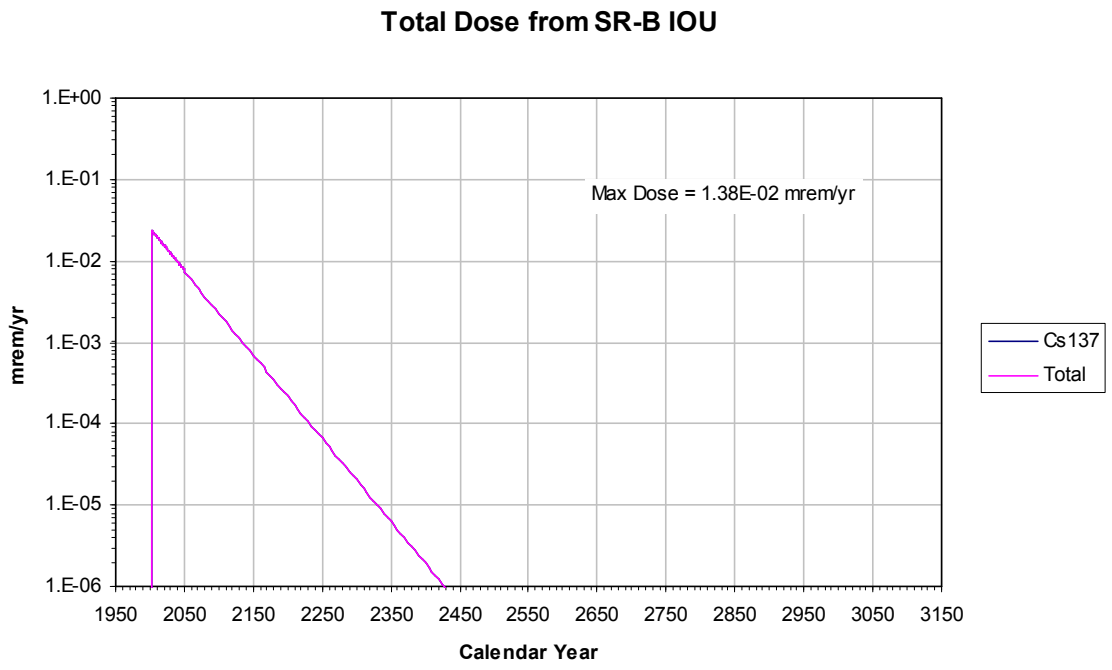


Figure E-113. SR-B IOU Direct Dose by Radionuclide

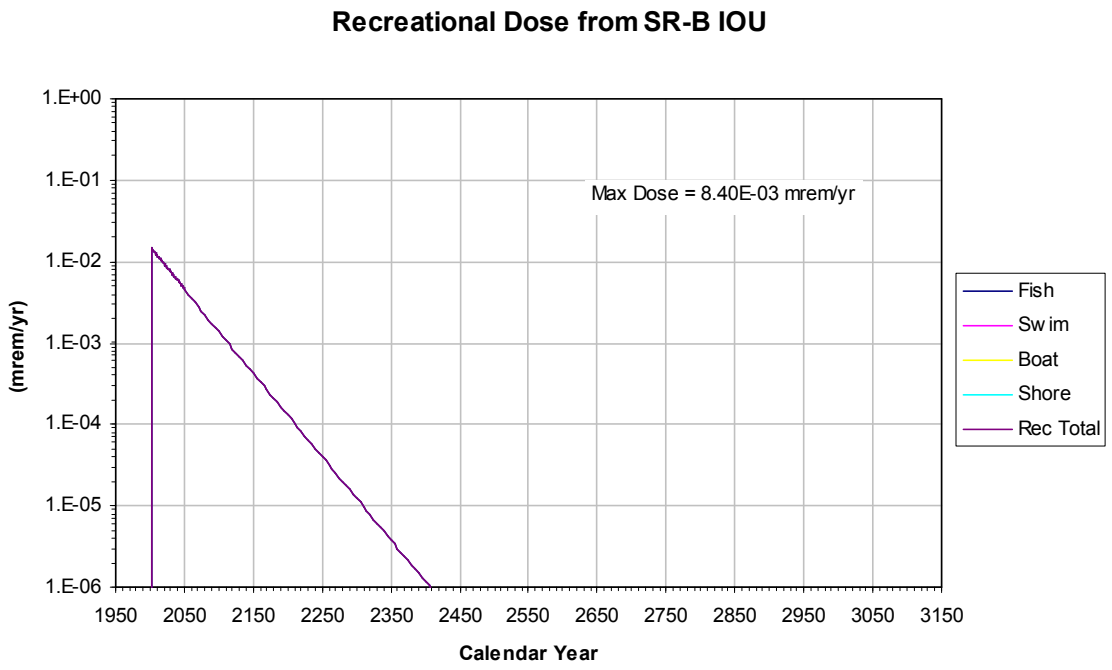


Figure E-114. SR-B IOU Recreational Dose

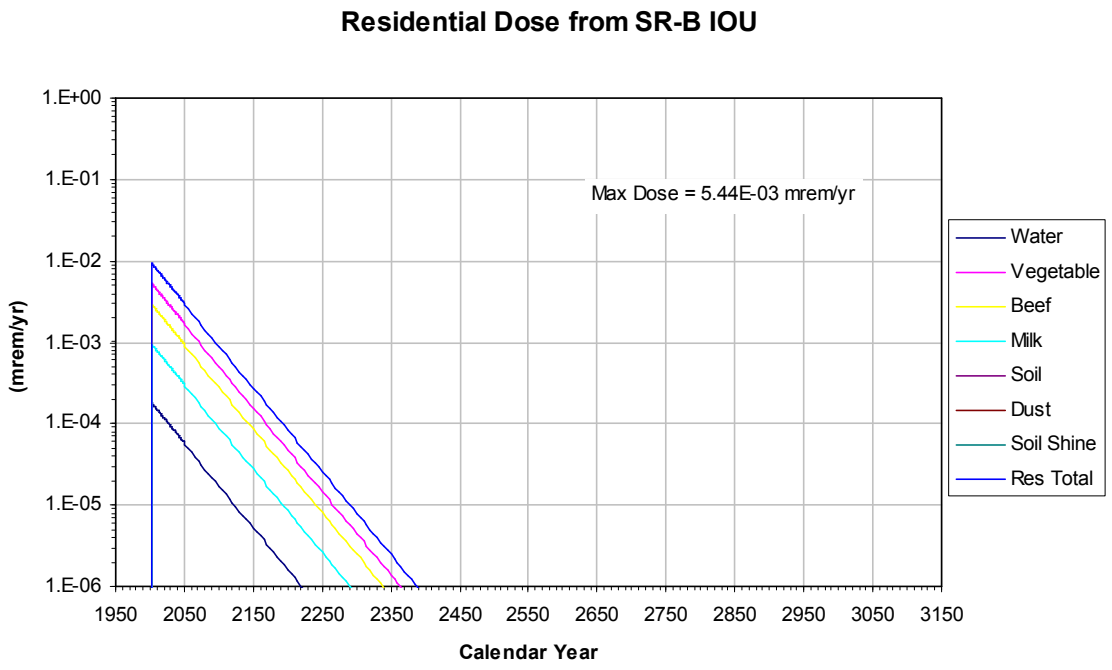


Figure E-115. SR-B IOU Direct Residential Dose

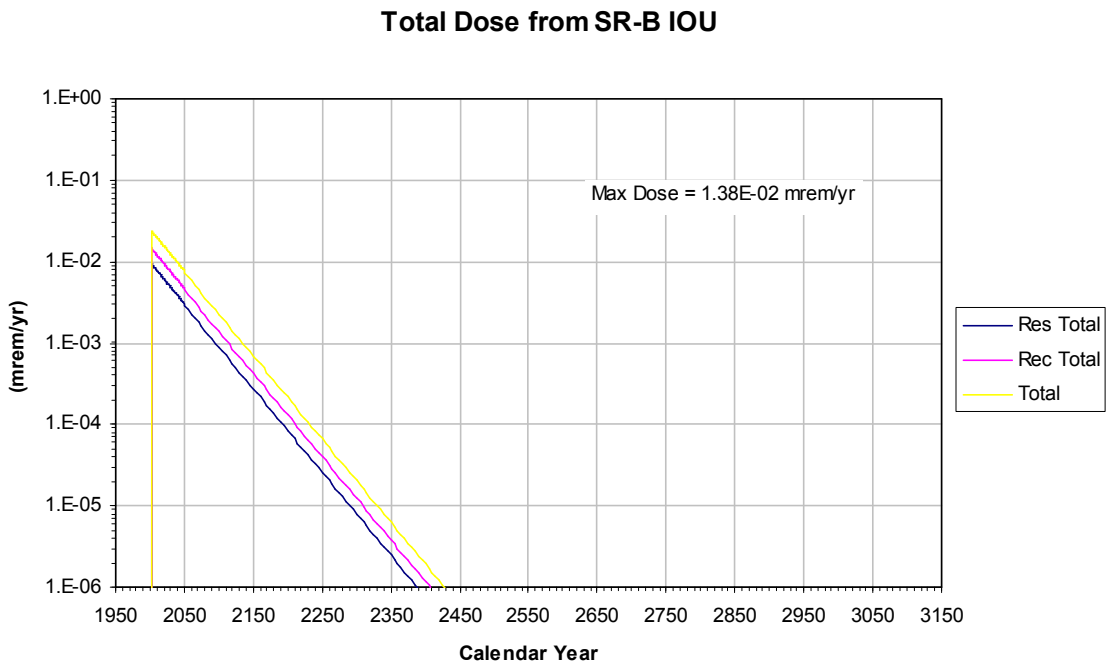


Figure E-116. SR-B IOU Total Direct Dose

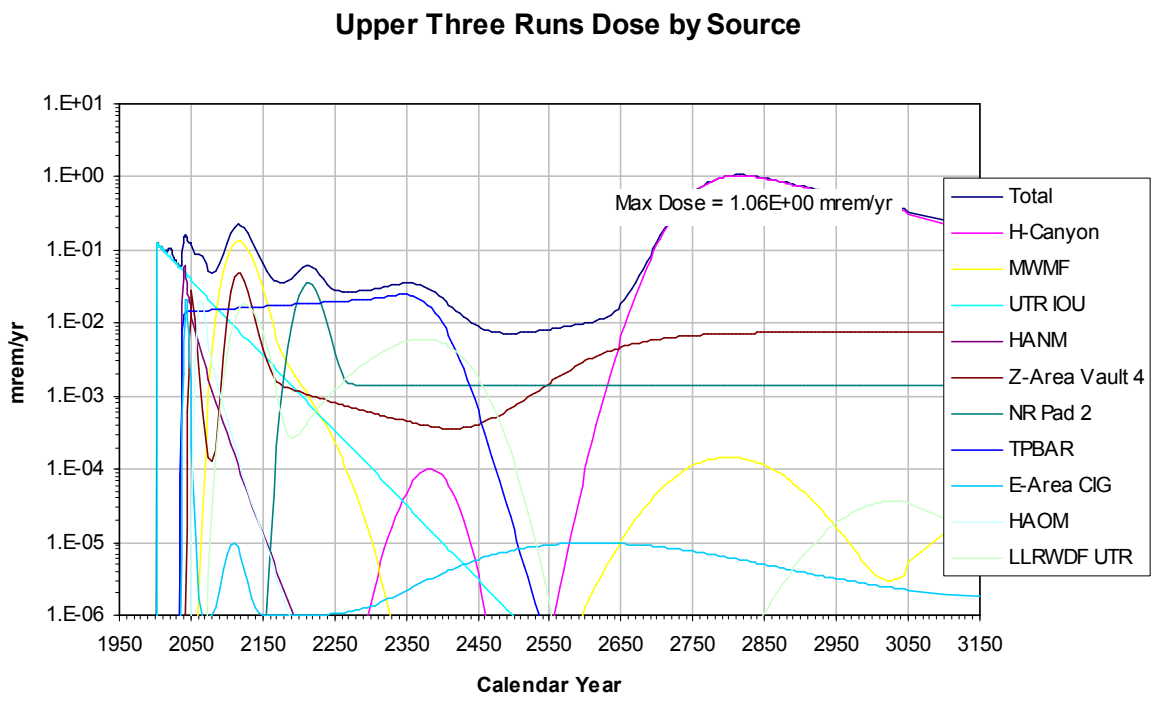


Figure E-117. Total Direct Dose to Upper Three Runs POA and Contribution from Major Sources

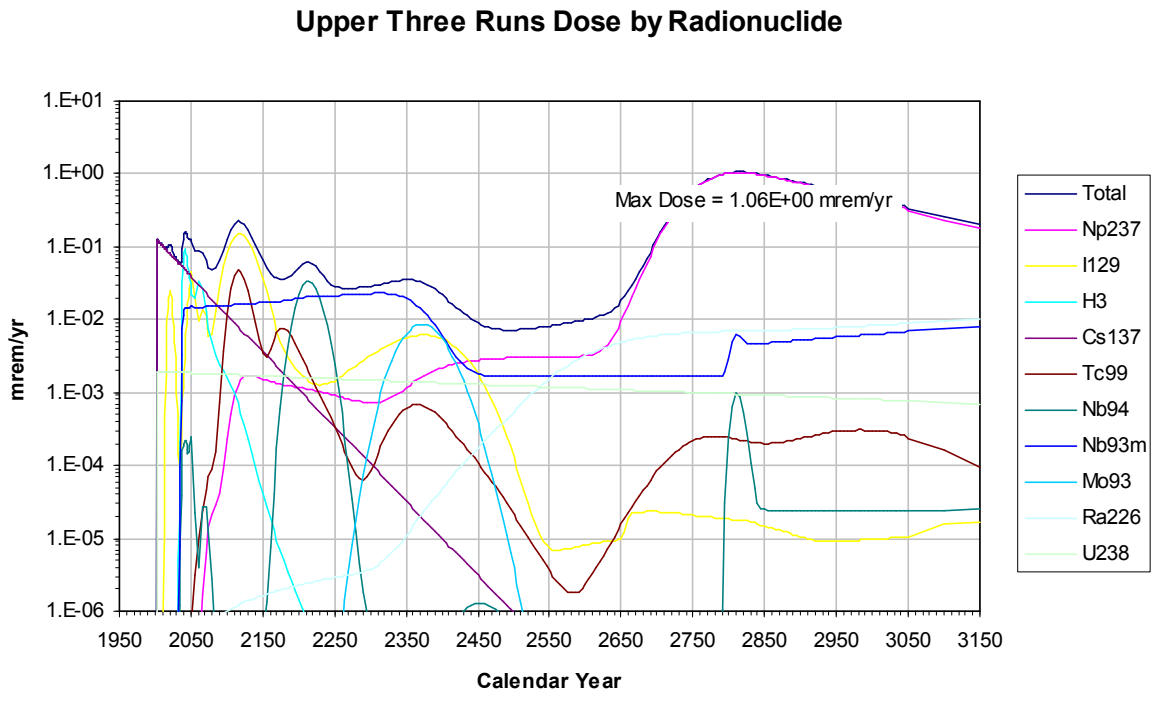


Figure E-118. Total Direct Dose to Upper Three Runs POA and Contribution from Major Radionuclides

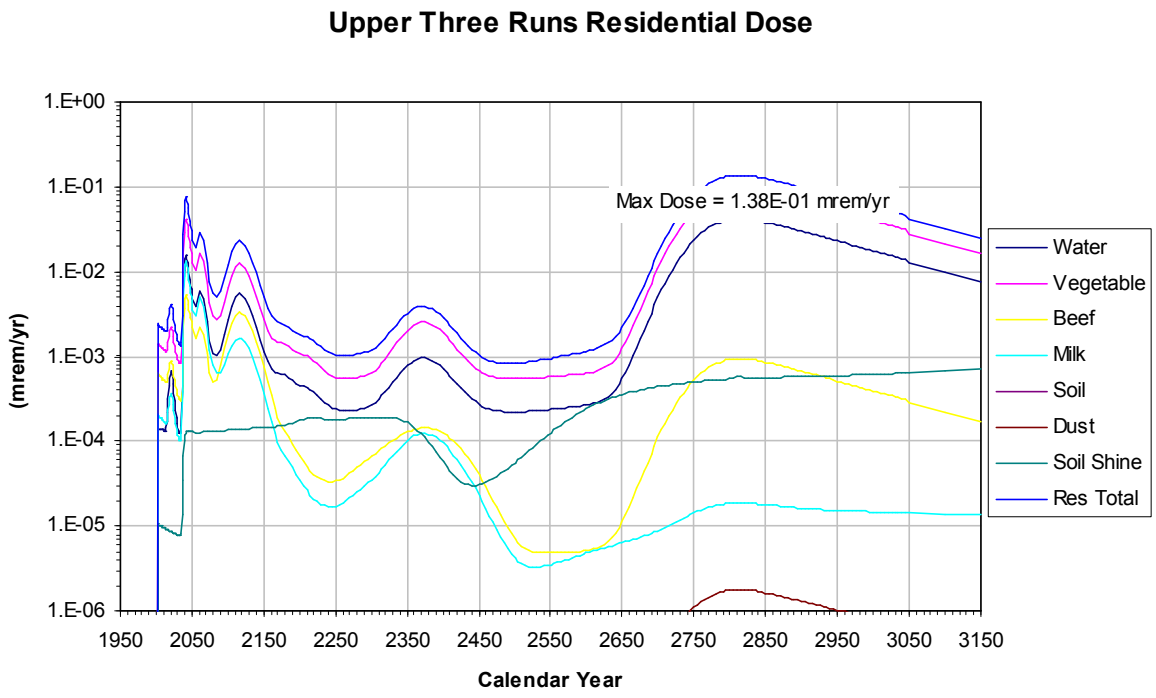


Figure E-119. **Residential Direct Dose to Upper Three Runs POA by Pathways**

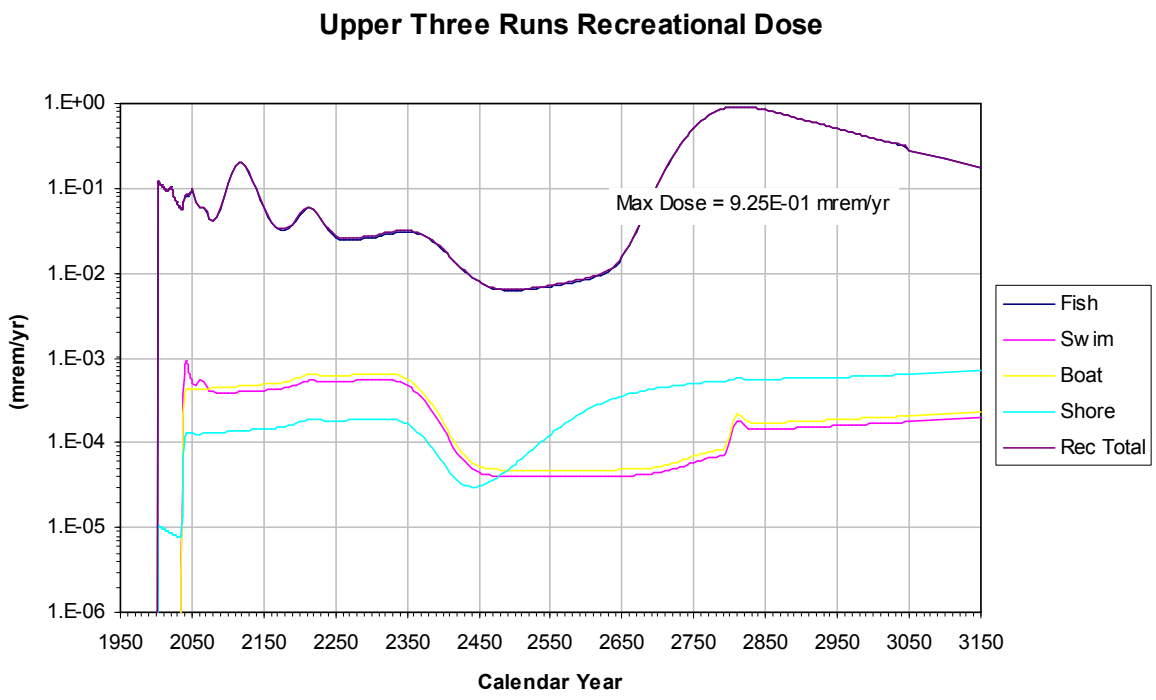


Figure E-120. **Recreational Dose to Upper Three Runs POA by Pathways**

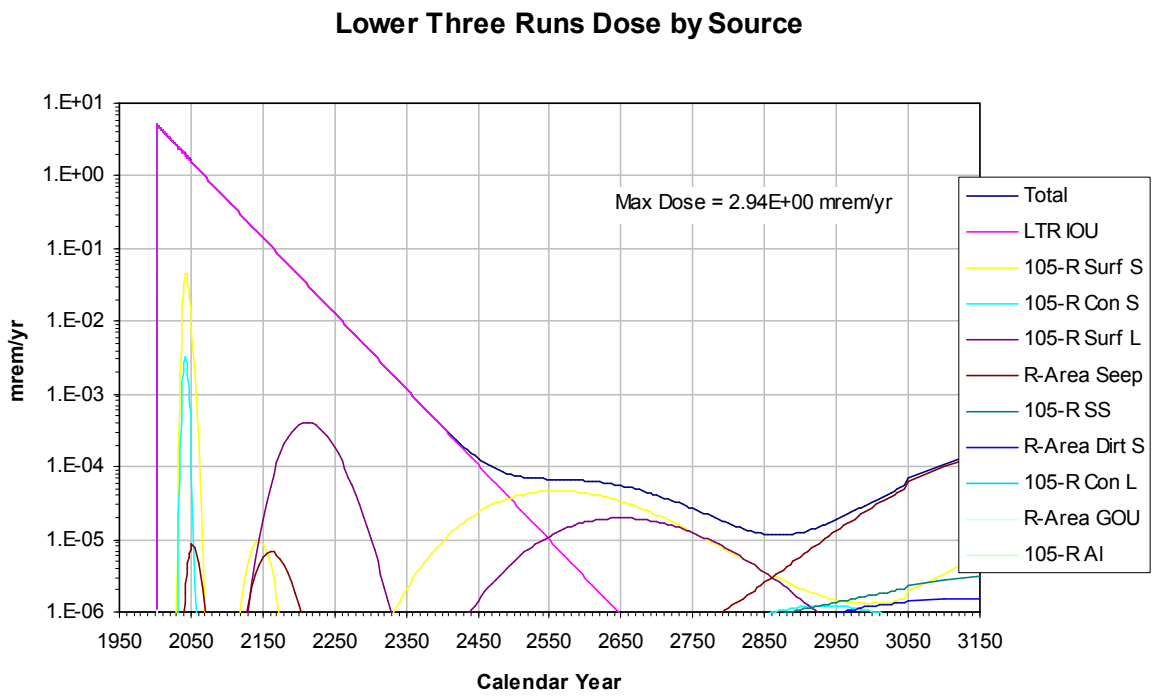


Figure E-121. Total Direct Dose to Lower Three Runs POA and Contribution from Major Sources

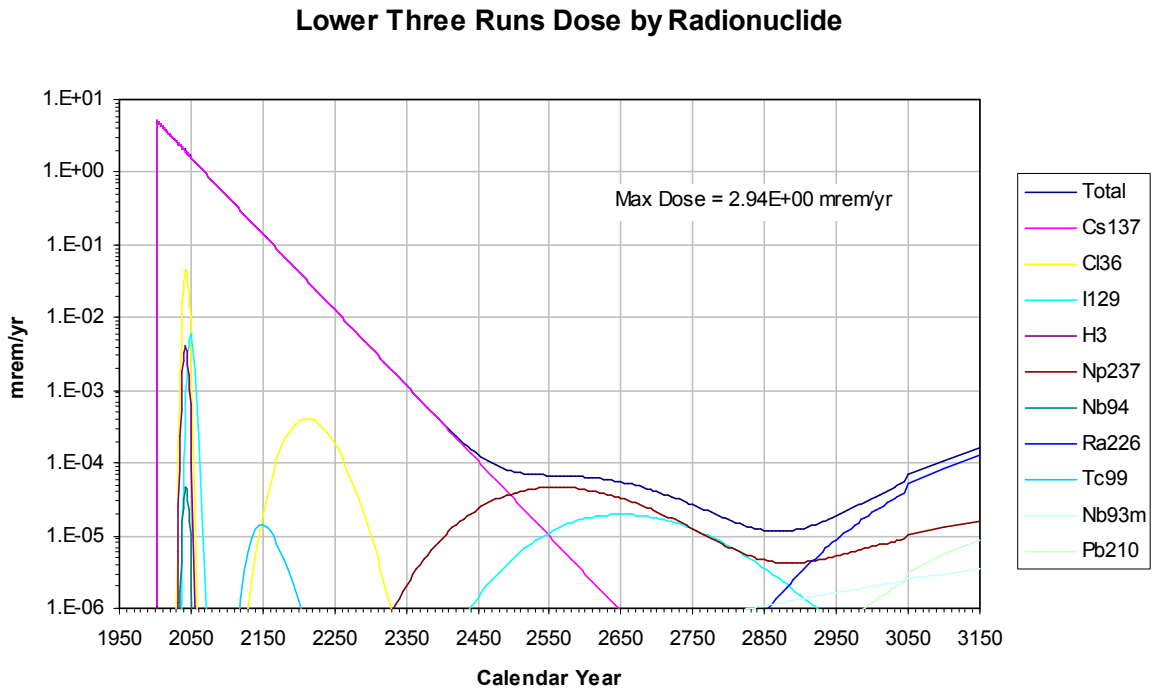


Figure E-122. Total Direct Dose to Lower Three Runs POA and Contribution from Major Radionuclides

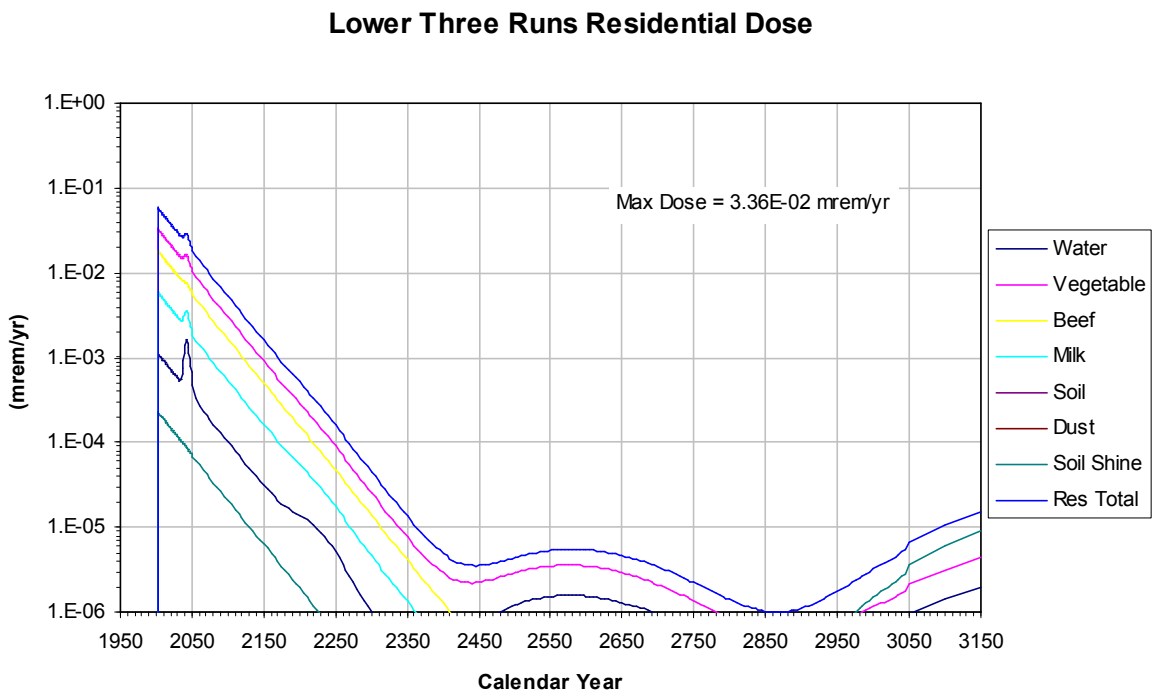


Figure E-123. **Residential Direct Dose to Lower Three Runs POA by Pathways**

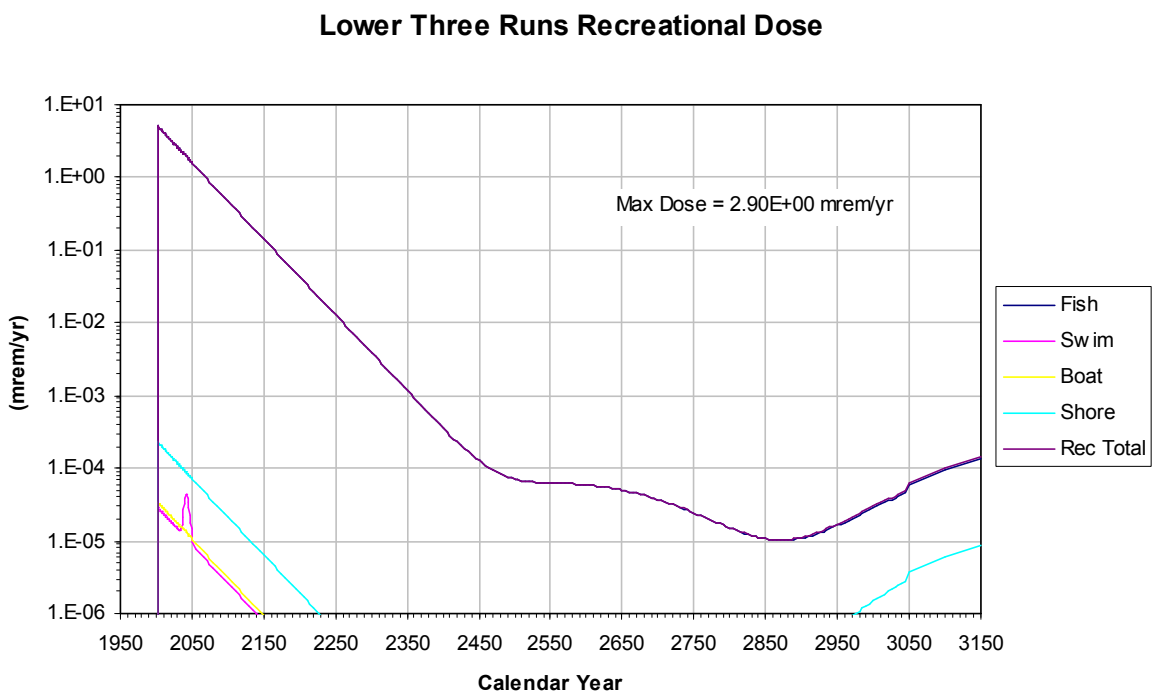


Figure E-124. Recreational Dose to Lower Three Runs POA by Pathways

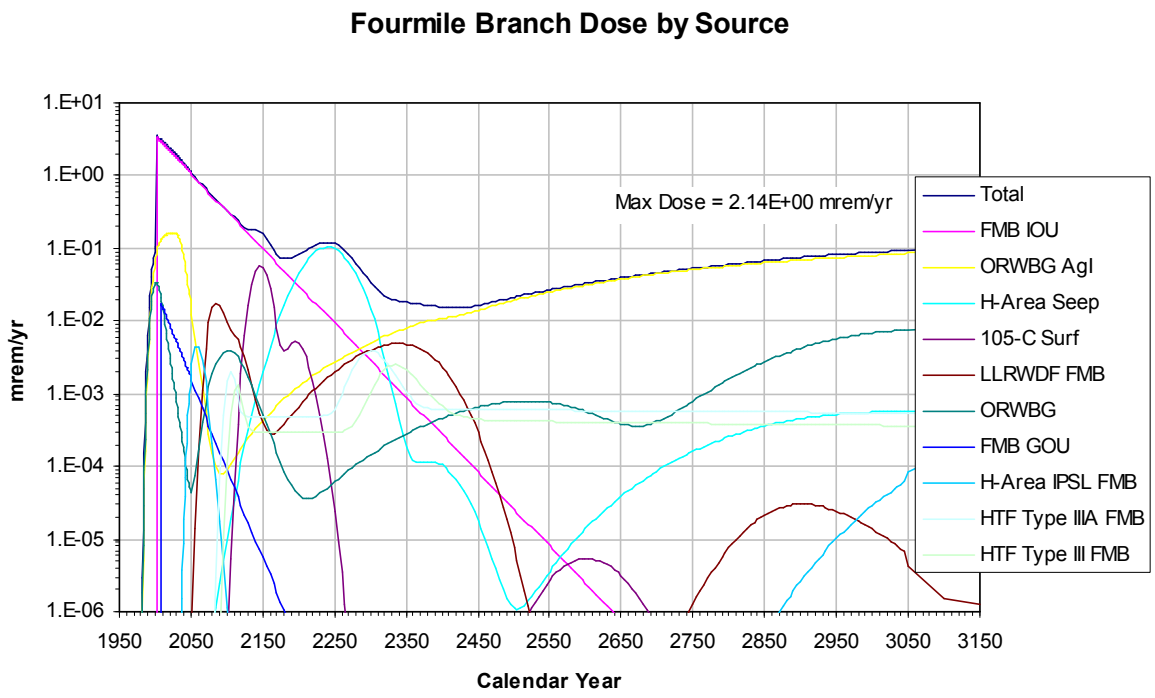


Figure E-125. Total Direct Dose to Fourmile Branch POA and Contribution from Major Sources

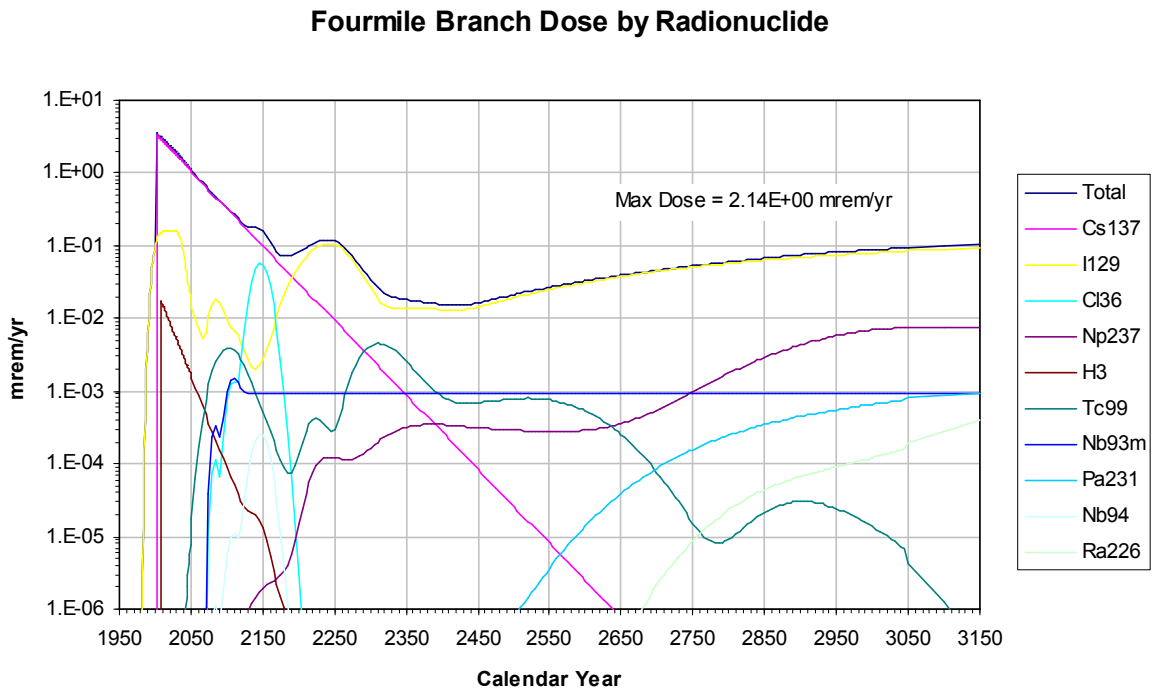


Figure E-126. Total Direct Dose to Fourmile Branch POA and Contribution from Major Radionuclides

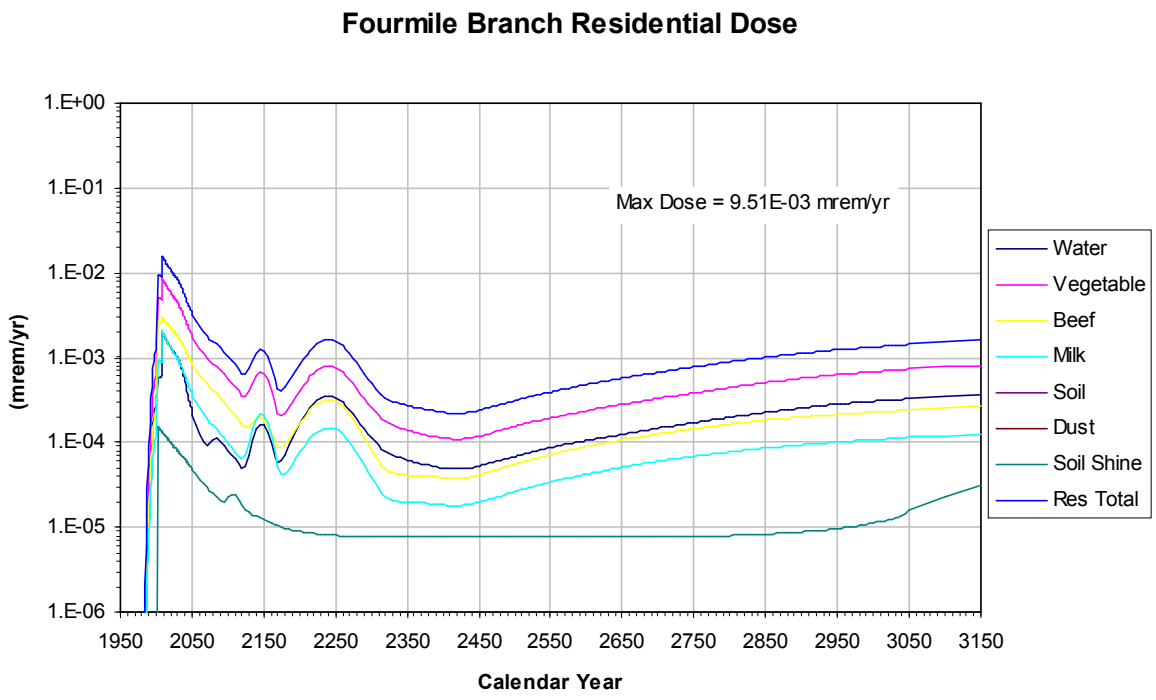


Figure E-127. **Residential Direct Dose to Fourmile Branch POA by Pathways**

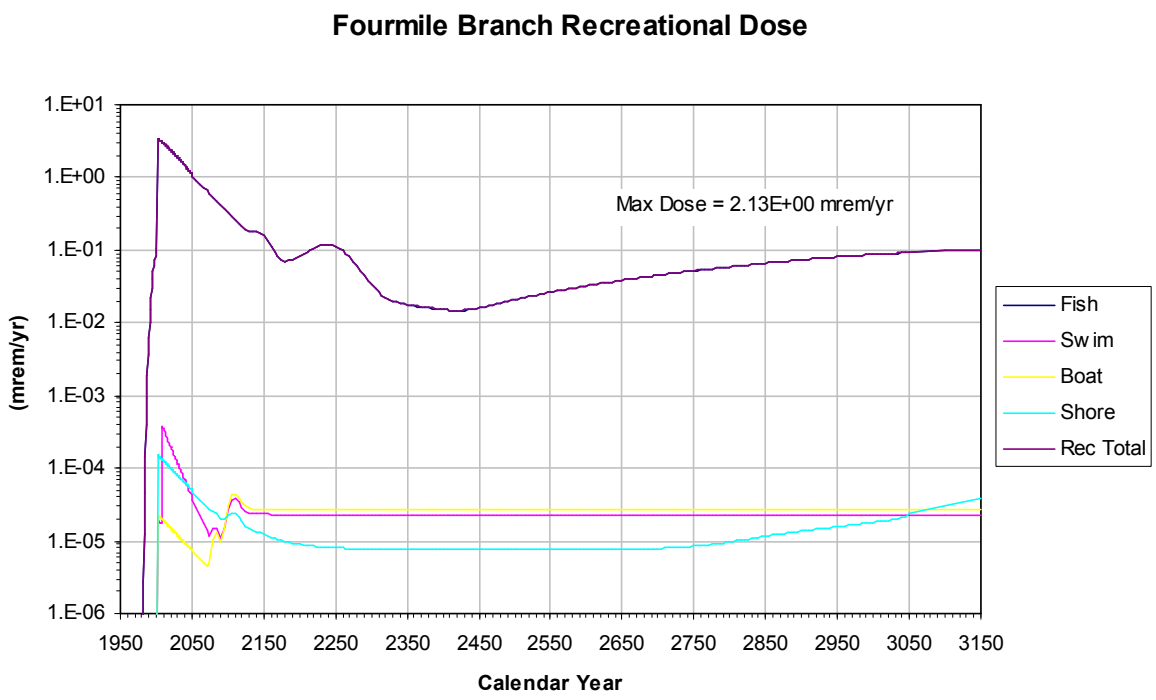


Figure E-128. **Recreational Dose to Fourmile Branch POA by Pathways**

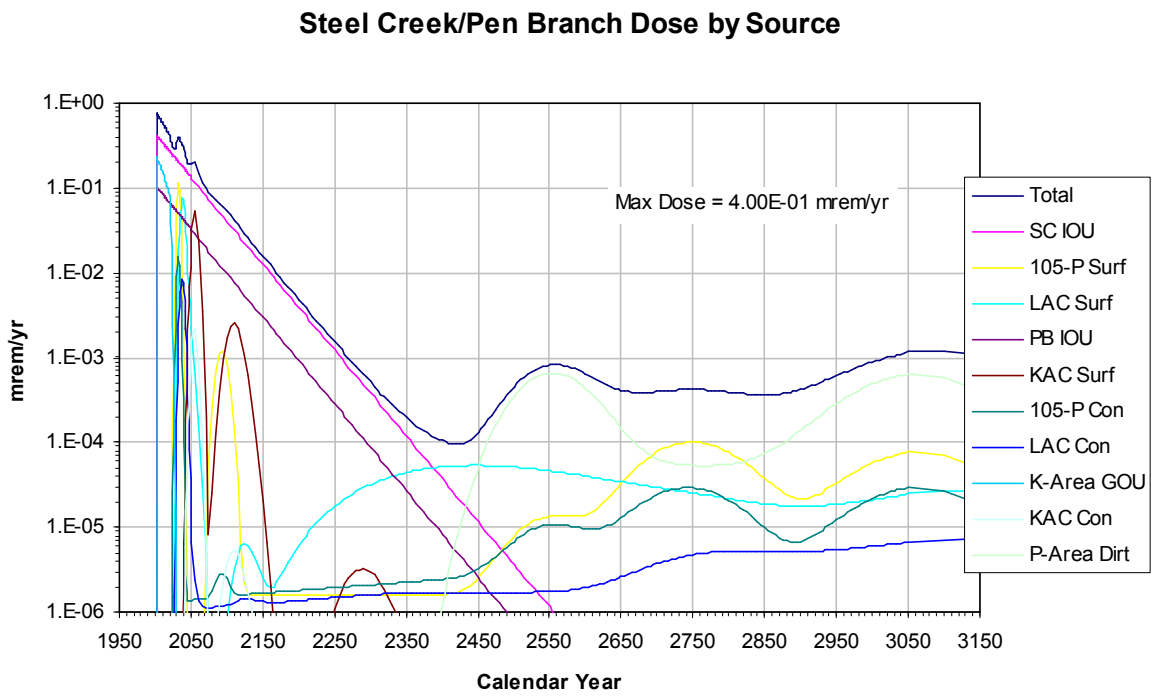


Figure E-129. **Total Direct Dose to Steel Creek/Pen Branch POA and Contribution from Major Sources**

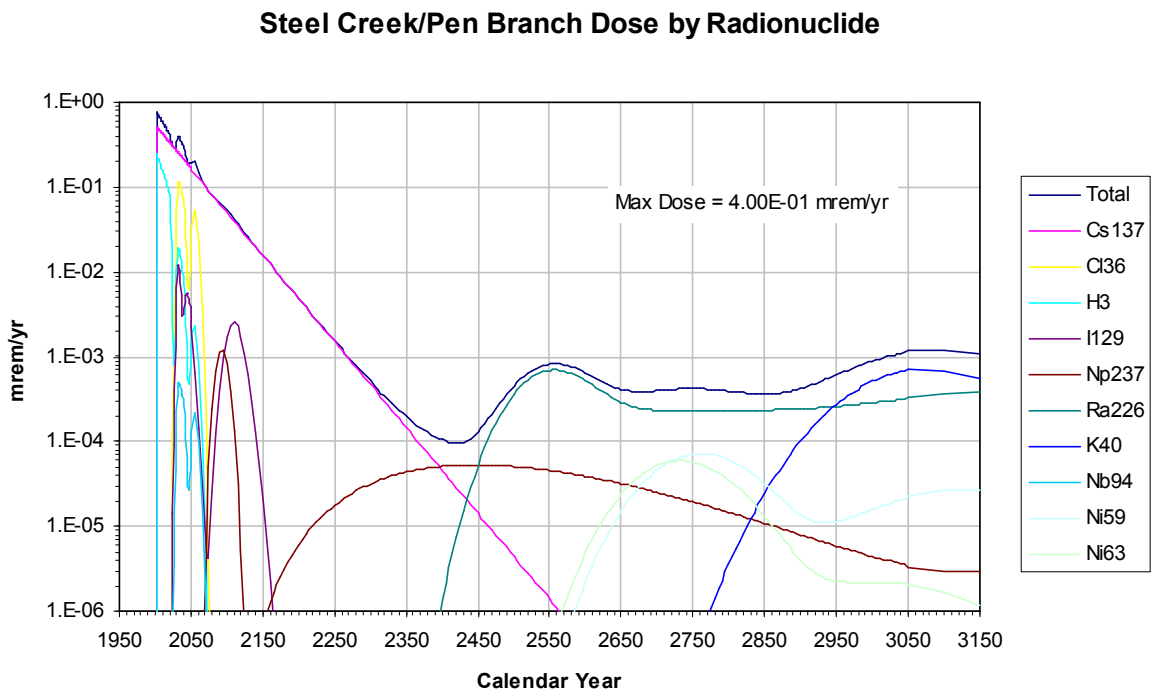


Figure E-130. **Total Direct Dose to Steel Creek/Pen Branch POA and Contribution from Major Radionuclides**

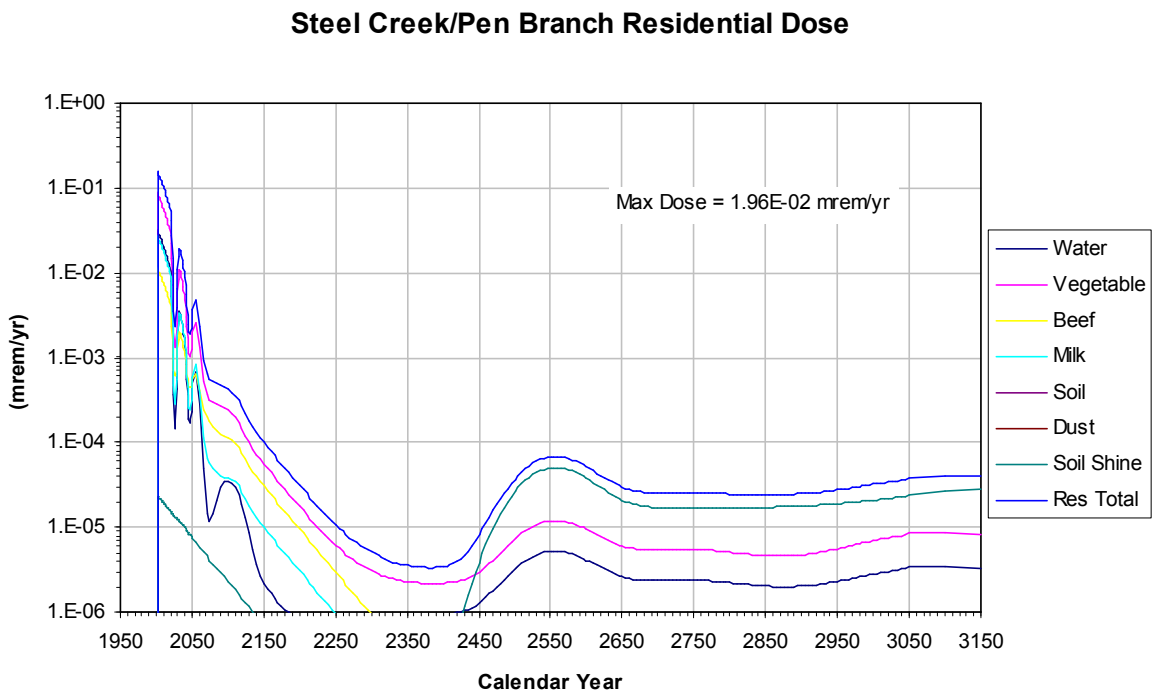


Figure E-131. **Residential Direct Dose to Steel Creek/Pen Branch POA by Pathways**

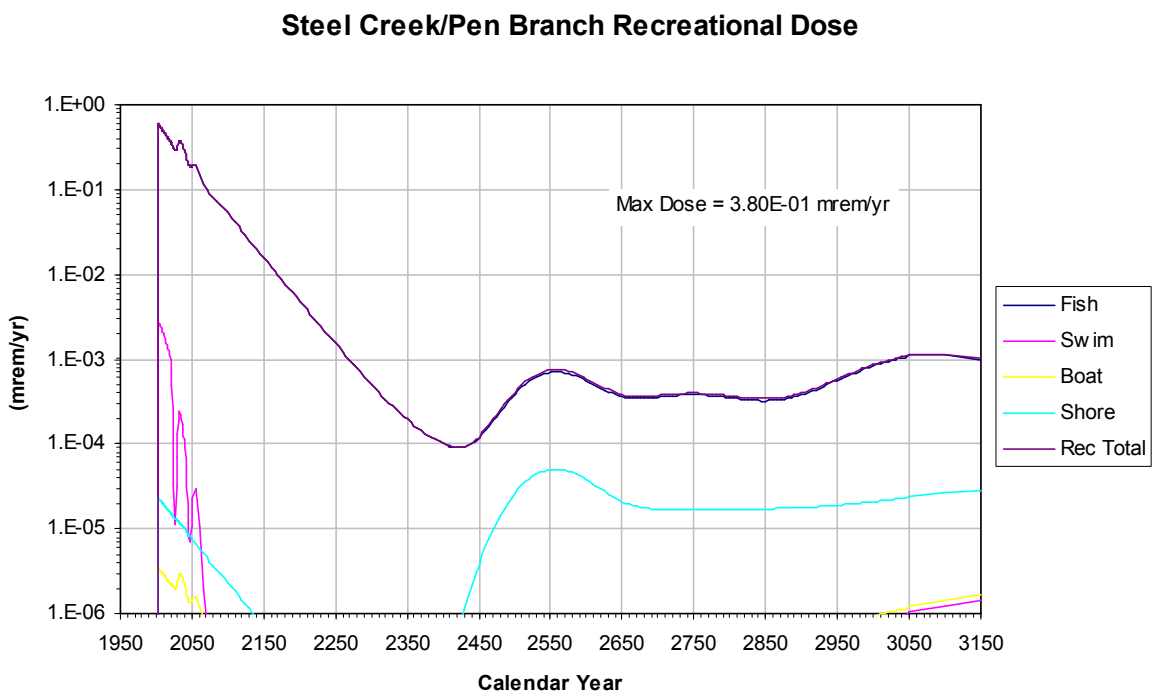


Figure E-132. **Recreational Dose to Steel Creek/Pen Branch POA by Pathways**

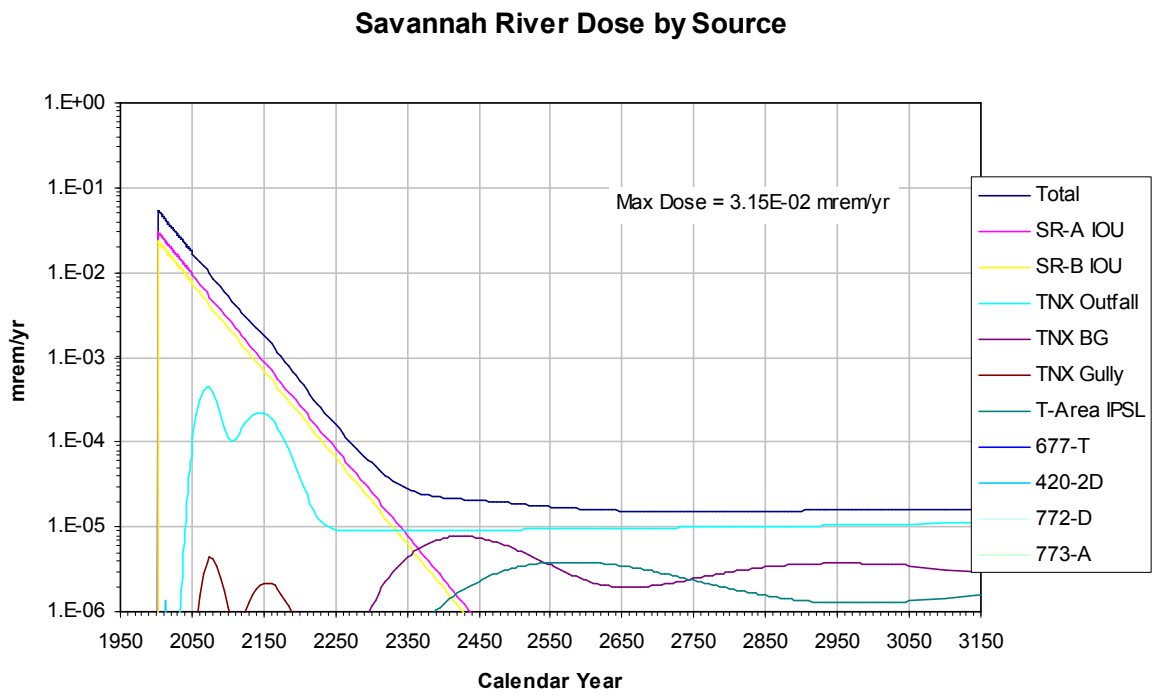


Figure E-133. **Total Direct Dose to Savannah River and Contribution from Major Sources**

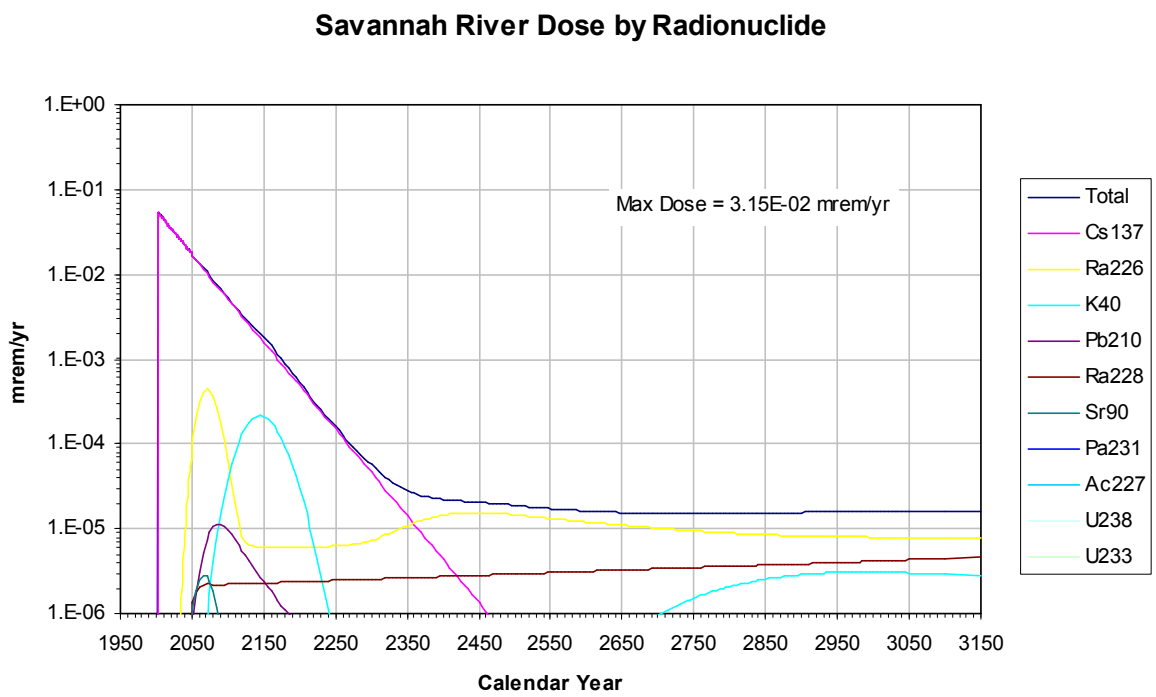


Figure E-134. **Total Direct Dose to Savannah River and Contribution from Major Radionuclides**

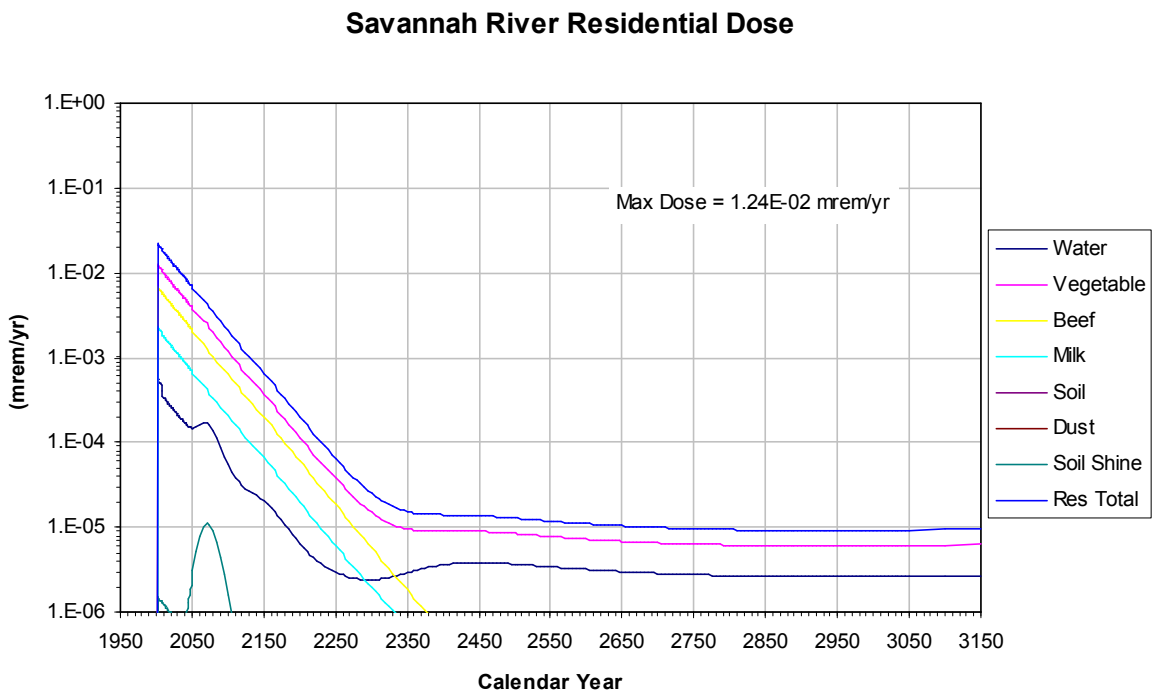


Figure E-135. **Residential Direct Dose to Savannah River by Pathways**

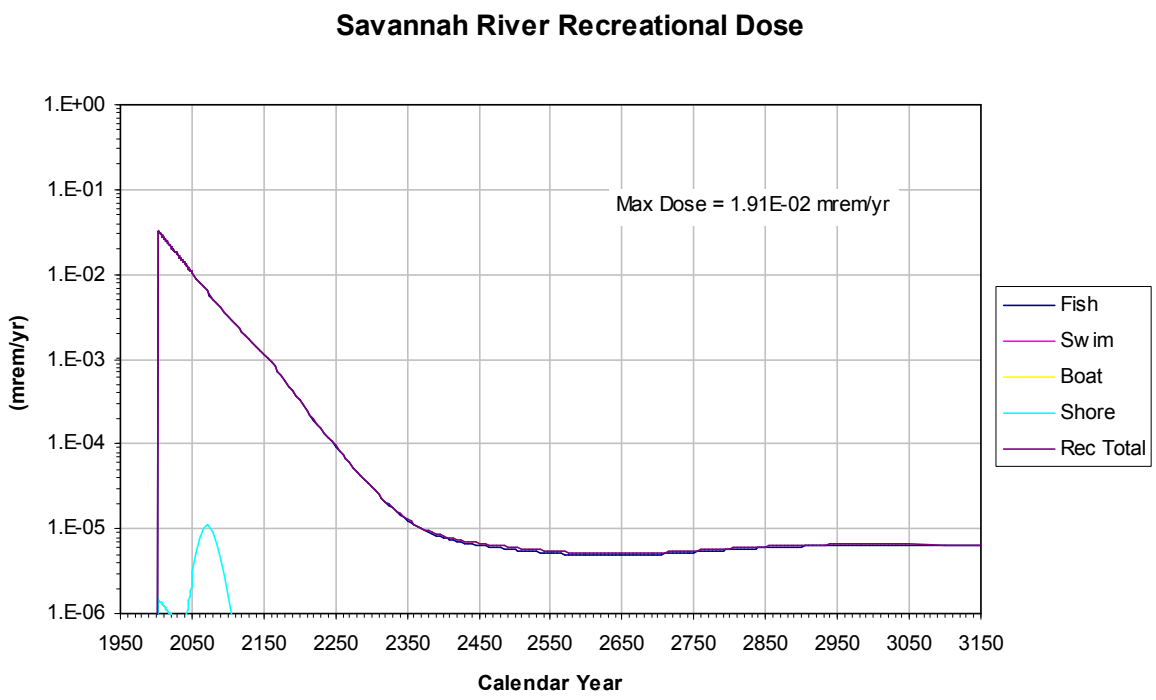


Figure E-136. **Recreational Direct Dose to Savannah River by Pathways**

E.2 SENSITIVITY OUTPUT RESULTS (FROM SMITH ET AL. 2009B)

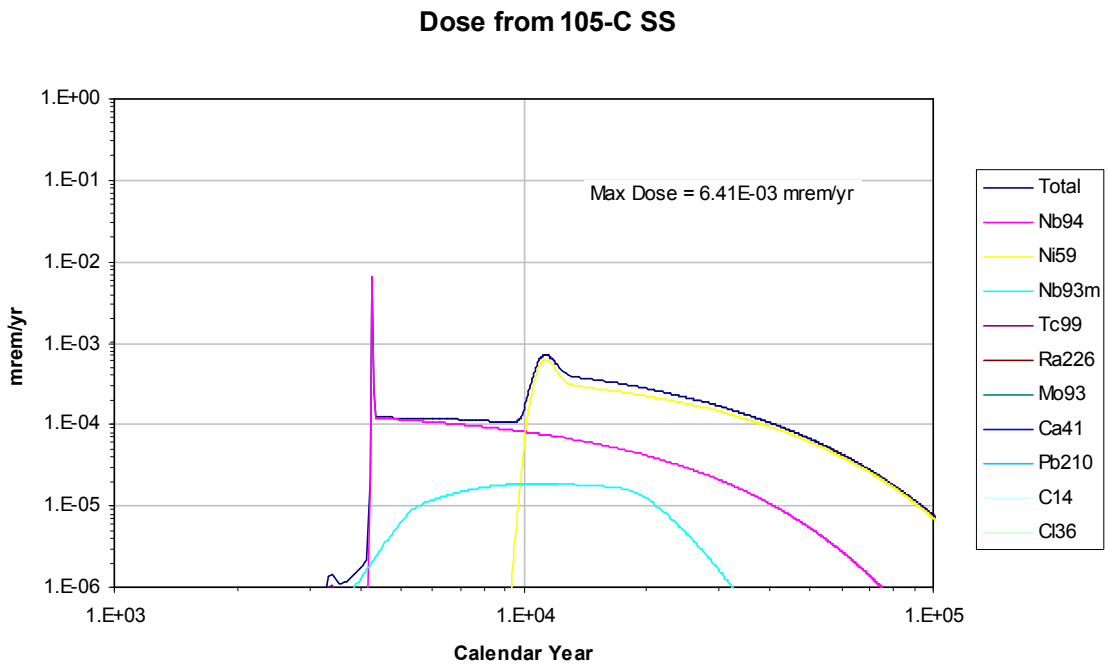


Figure E-137. **105-C SS Total Direct Dose by Isotope**

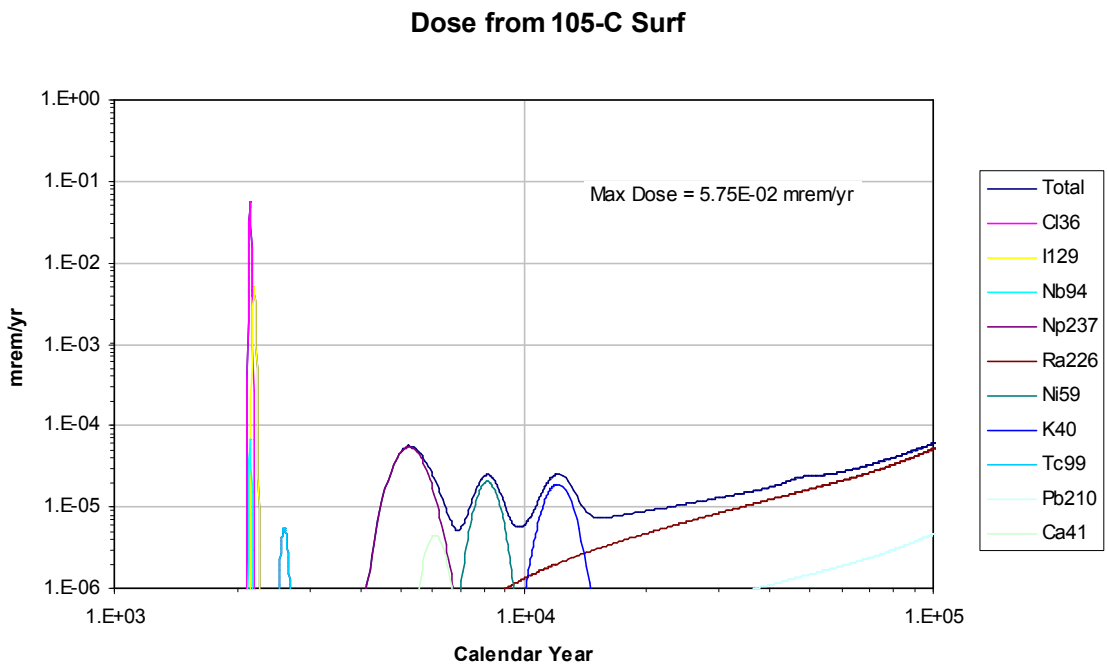


Figure E-138. **105-C Surf Total Direct Dose by Isotope**

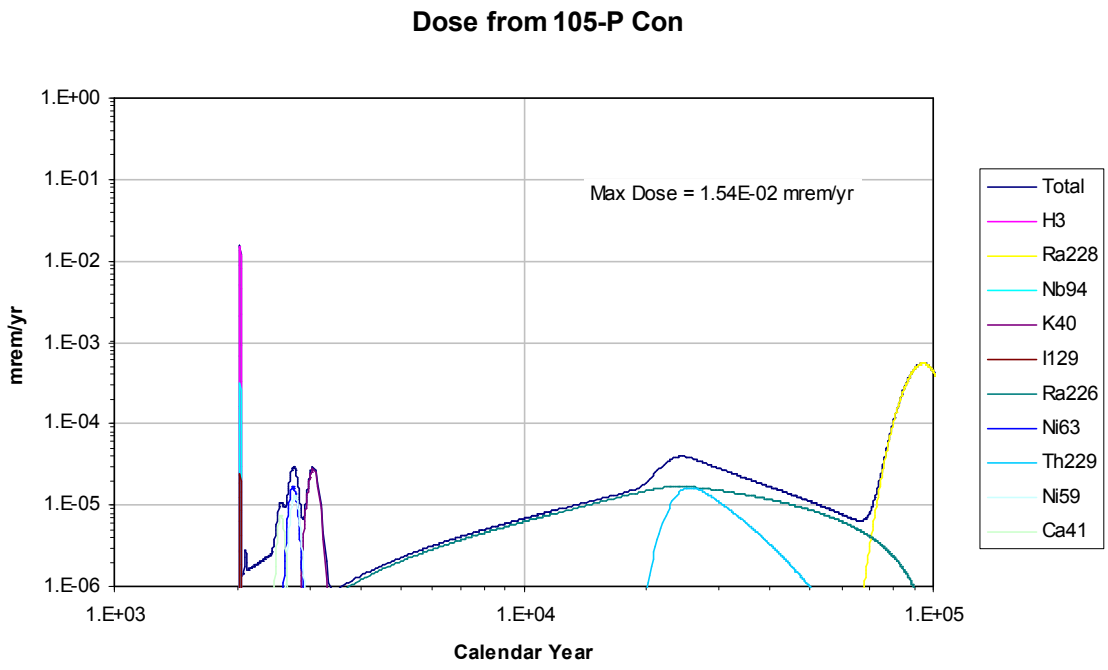


Figure E-139. **105-P Con Total Direct Dose by Isotope**

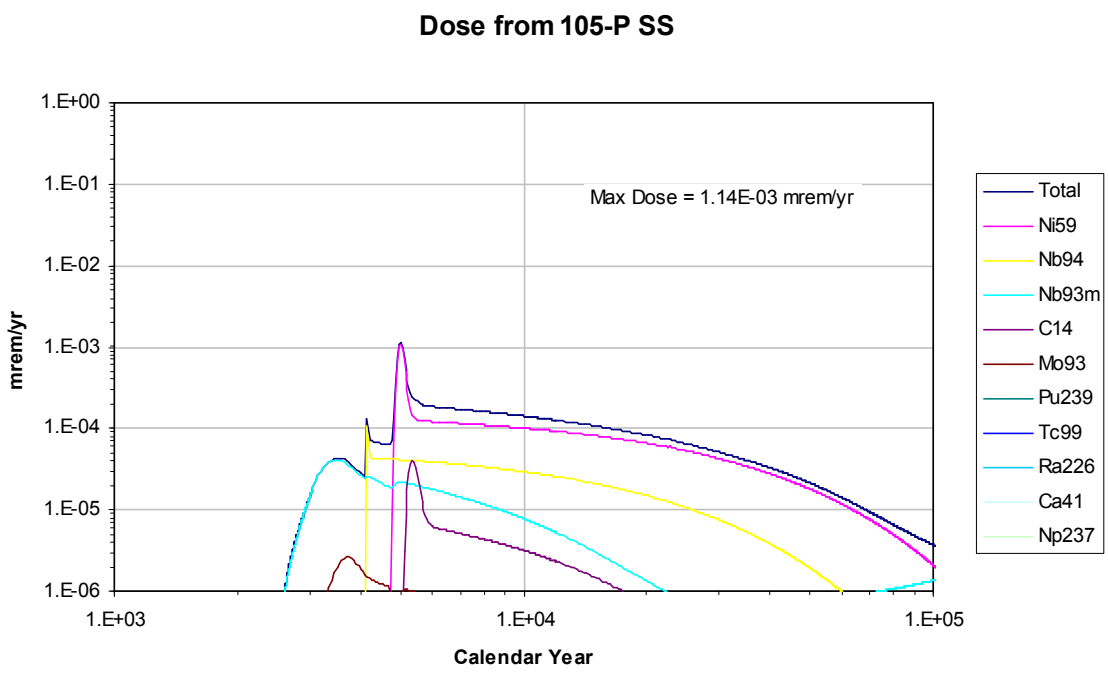


Figure E-140. **105-P SS Total Direct Dose by Isotope**

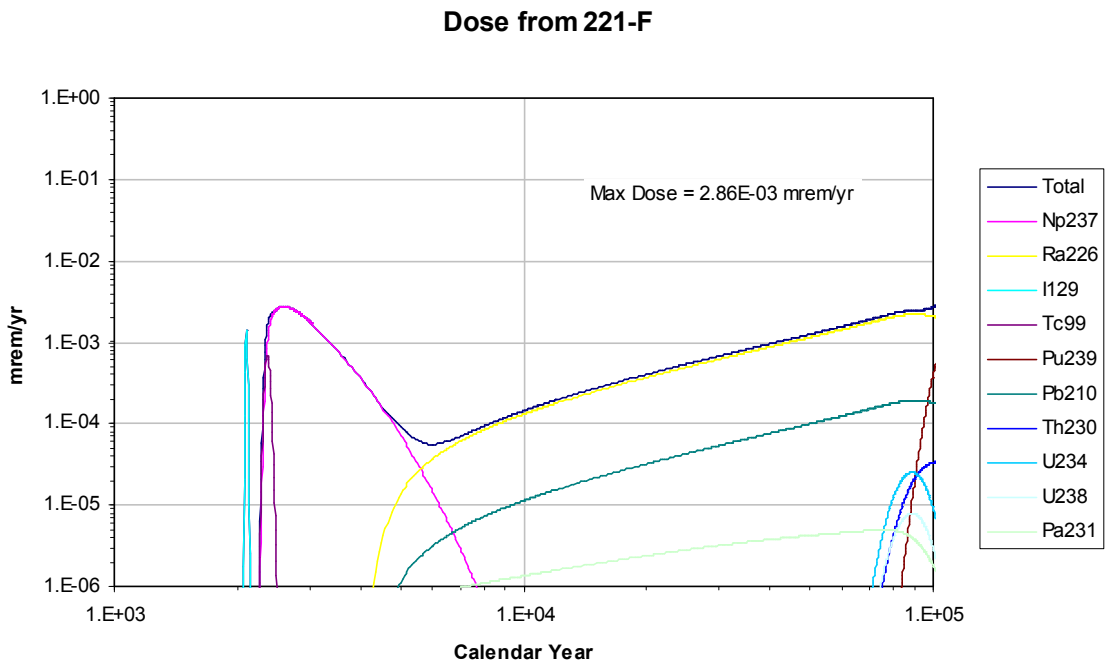


Figure E-141. **221-F Total Direct Dose by Isotope**

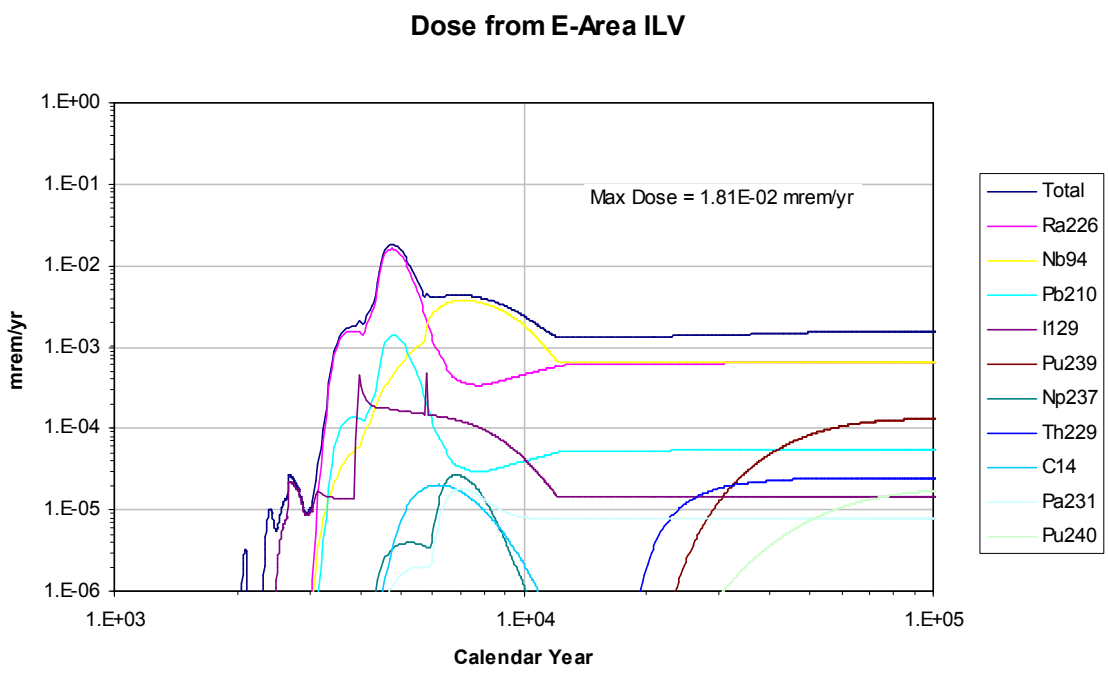


Figure E-142. **E-Area ILV Total Direct Dose by Isotope**

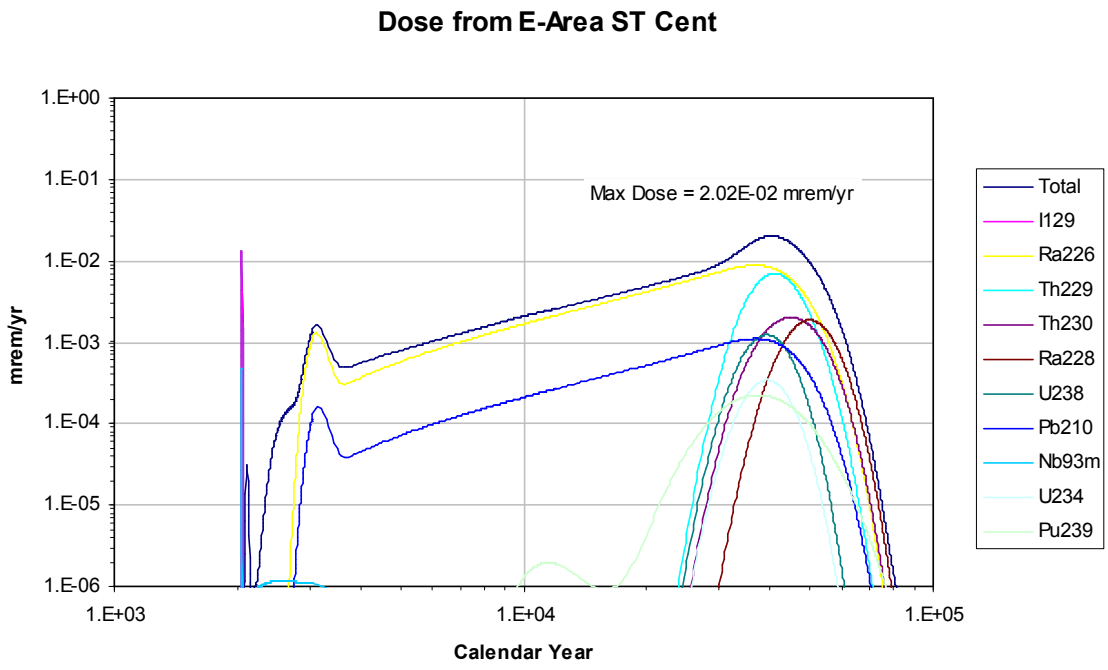


Figure E-143. **E-Area ST Cent Total Direct Dose by Isotope**

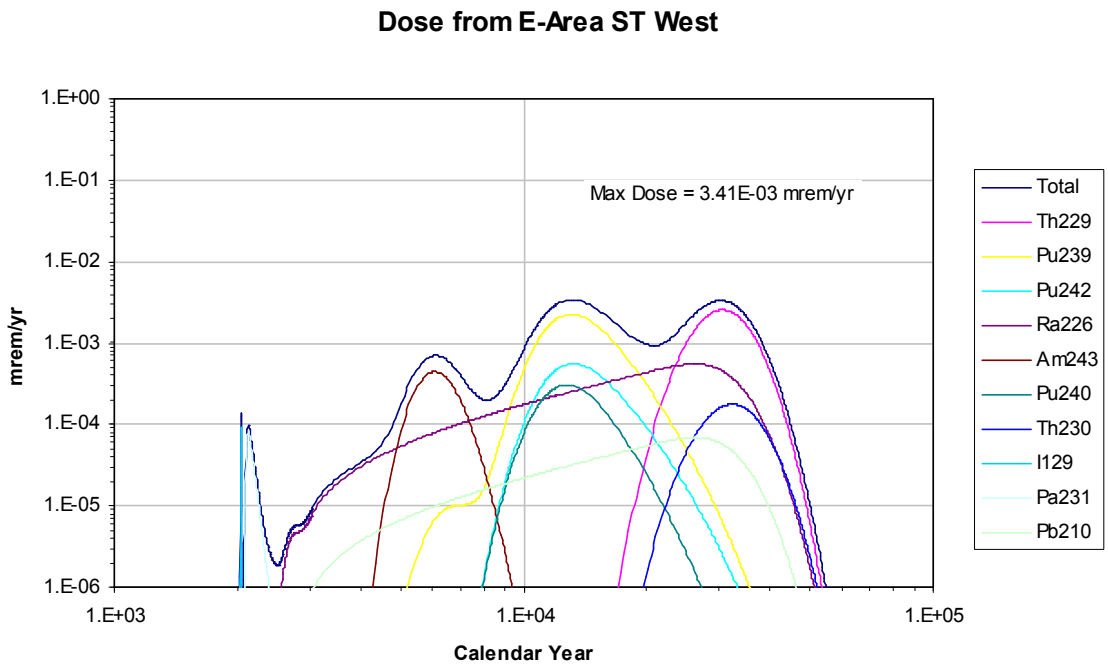


Figure E-144. **E-Area ST West Total Direct Dose by Isotope**

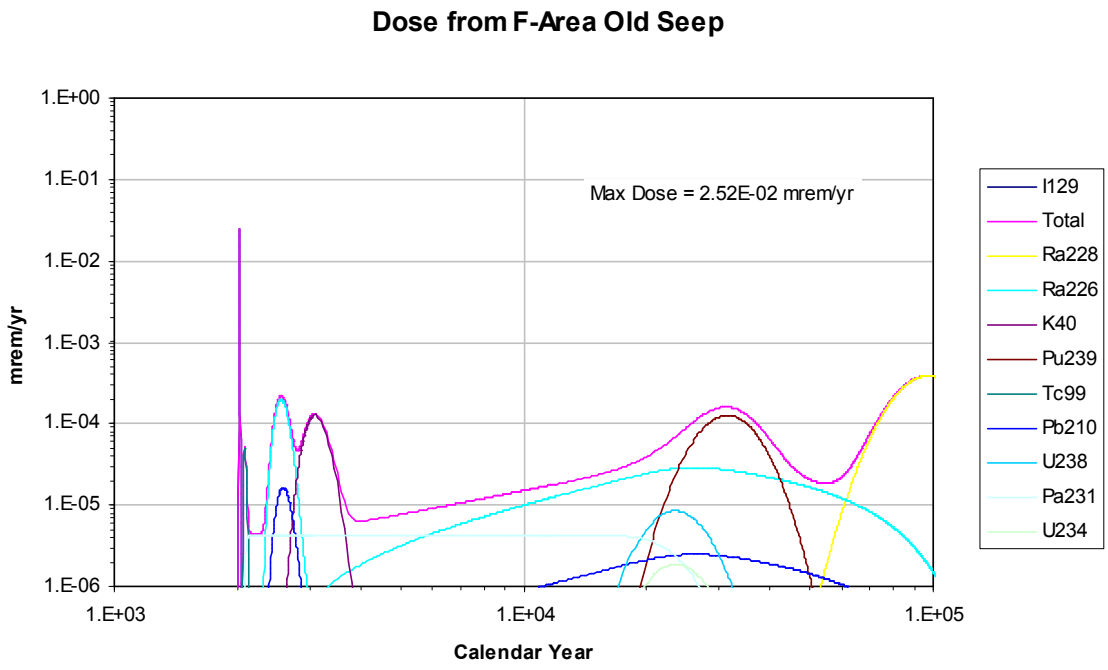


Figure E-145. **F-Area Old Seep Total Direct Dose by Isotope**

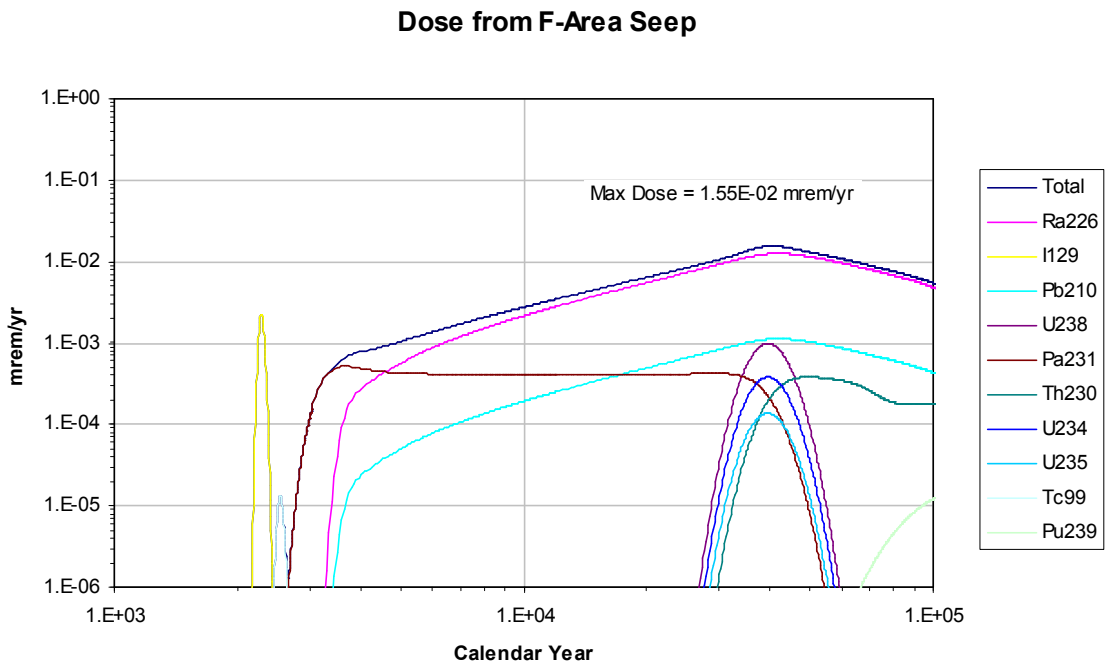


Figure E-146. **F-Area Seep Total Direct Dose by Isotope**

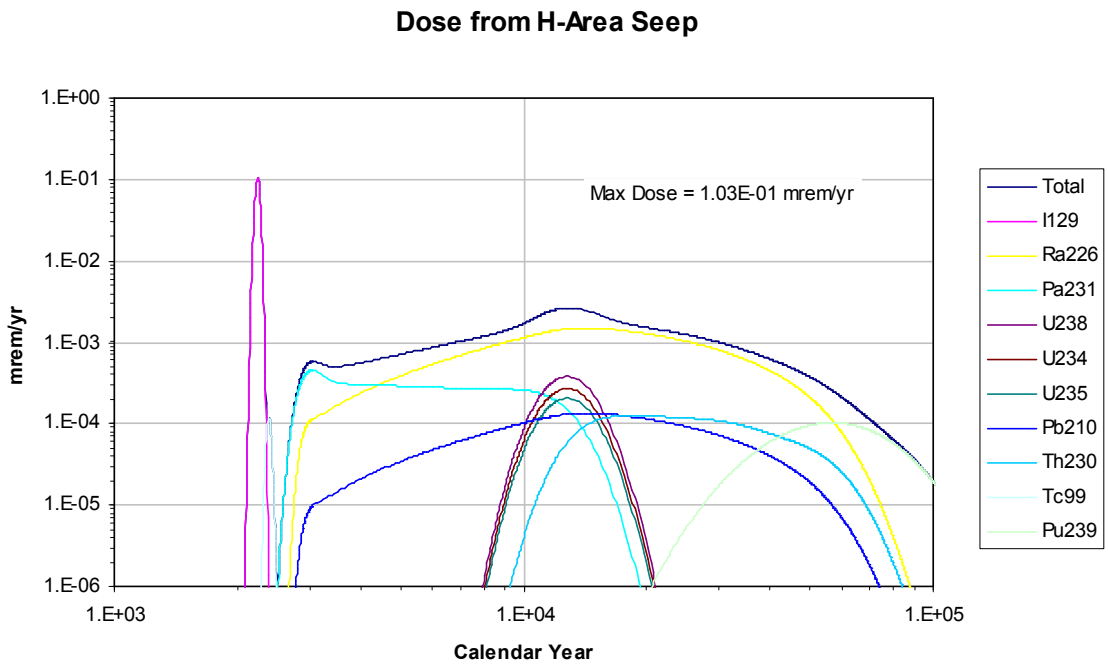


Figure E-147. **H-Area Seep Total Direct Dose by Isotope**

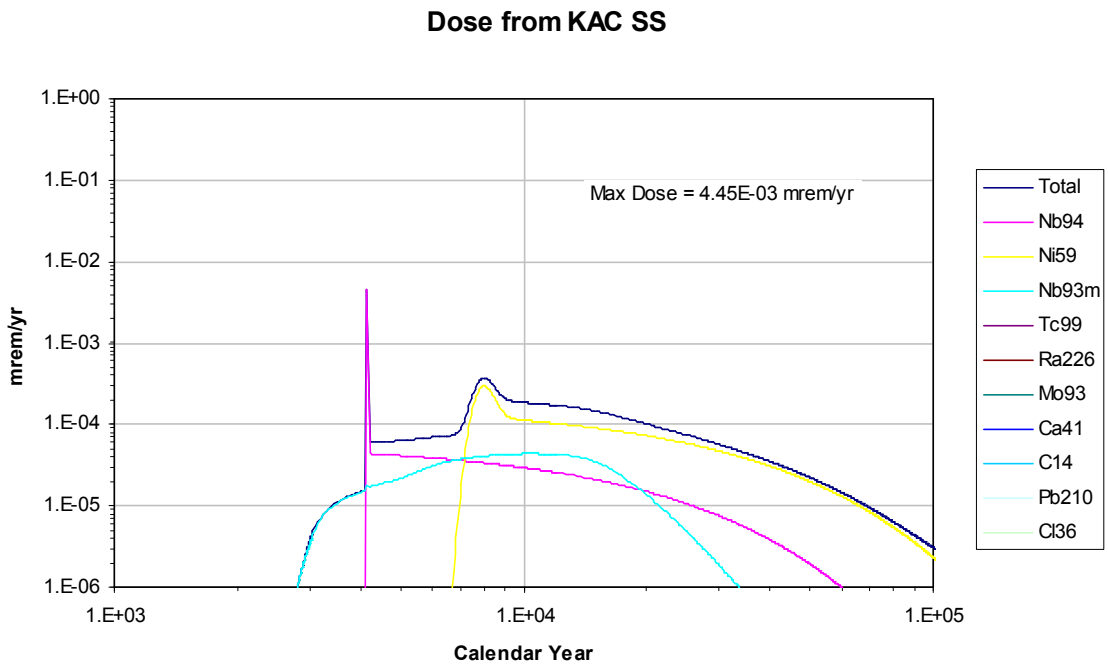


Figure E-148. **KAC SS Total Direct Dose by Isotope**

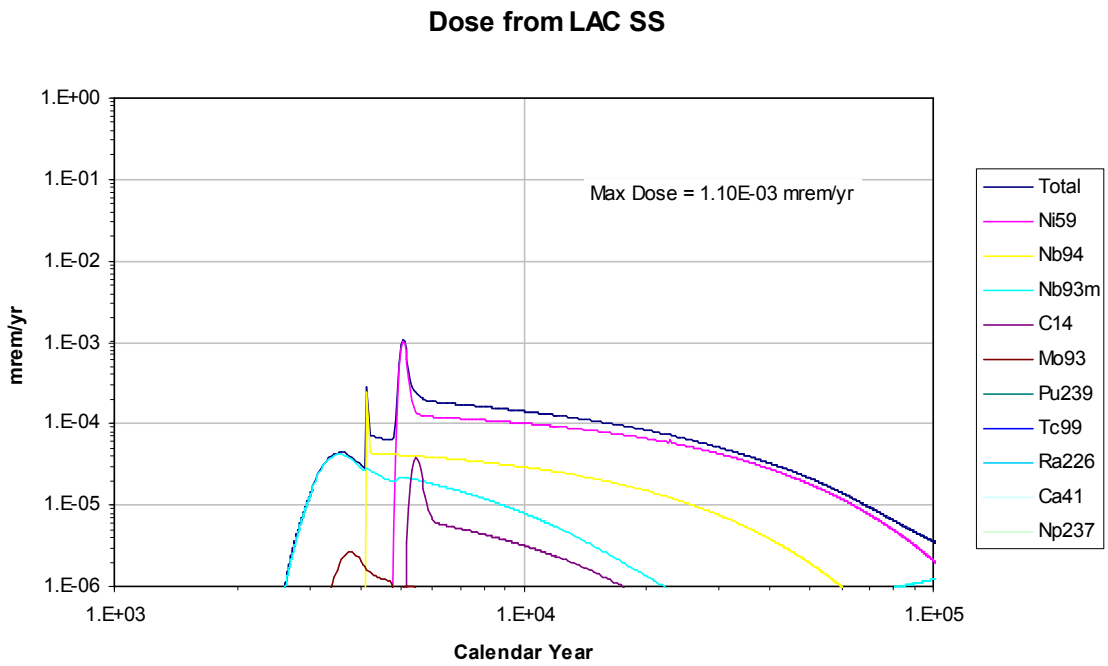


Figure E-149. **LAC SS Total Direct Dose by Isotope**

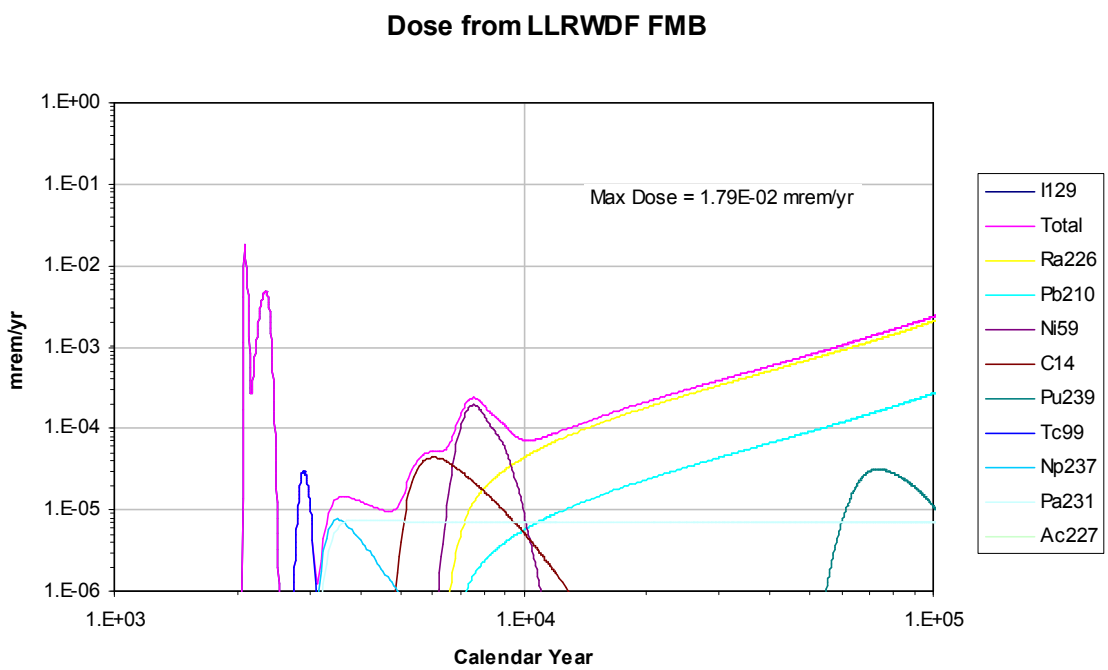


Figure E-150. **LLRWDF FMB Total Direct Dose by Isotope**

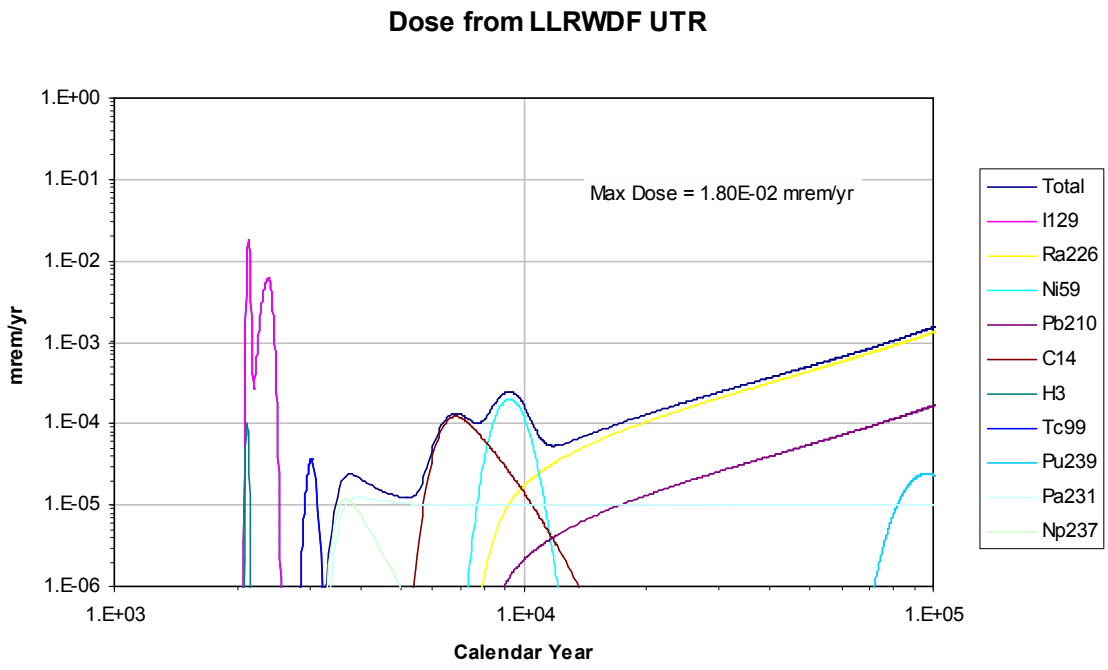


Figure E-151. **LLRWDF UTR Total Direct Dose by Isotope**

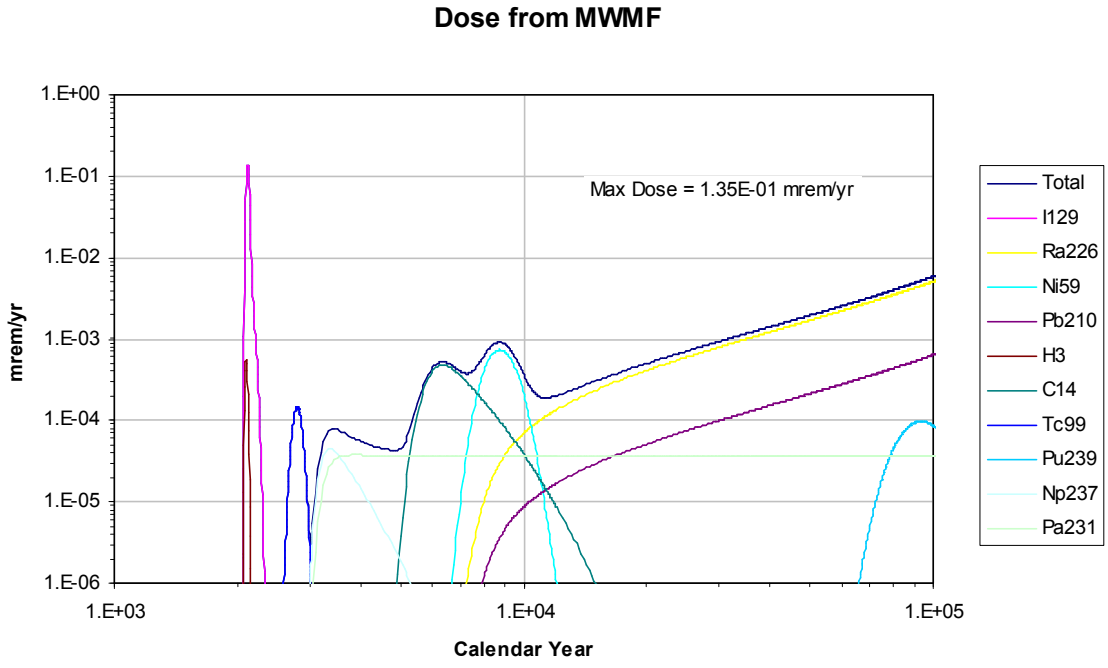


Figure E-152. **MWMF Total Direct Dose by Isotope**

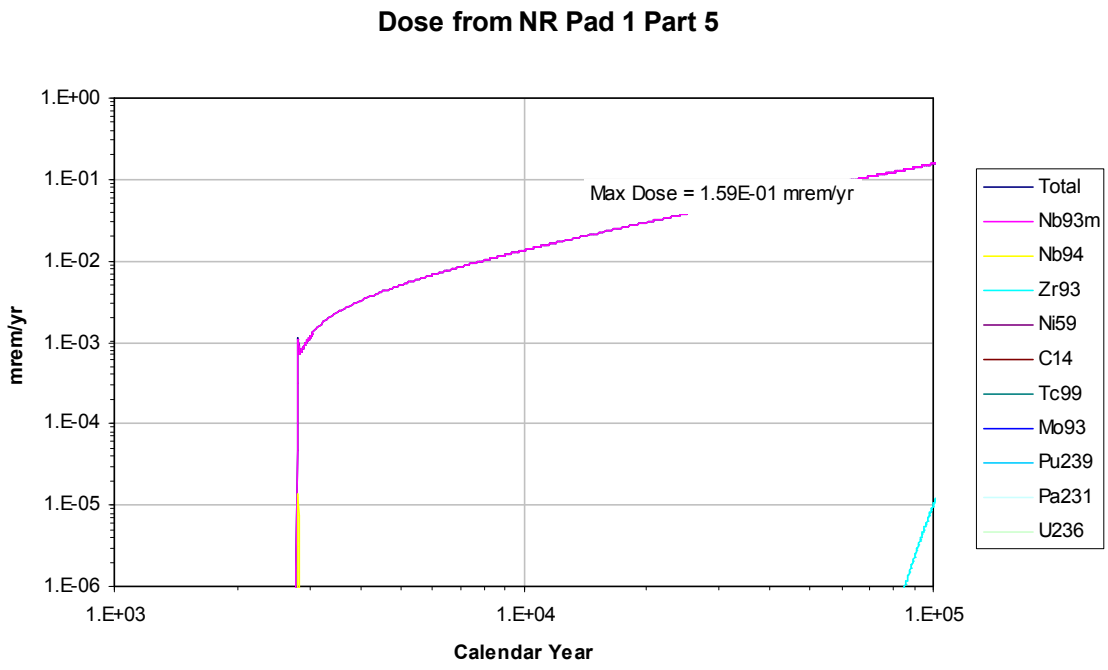


Figure E-153. **NR Pad 1 Part 5 Total Direct Dose by Isotope**

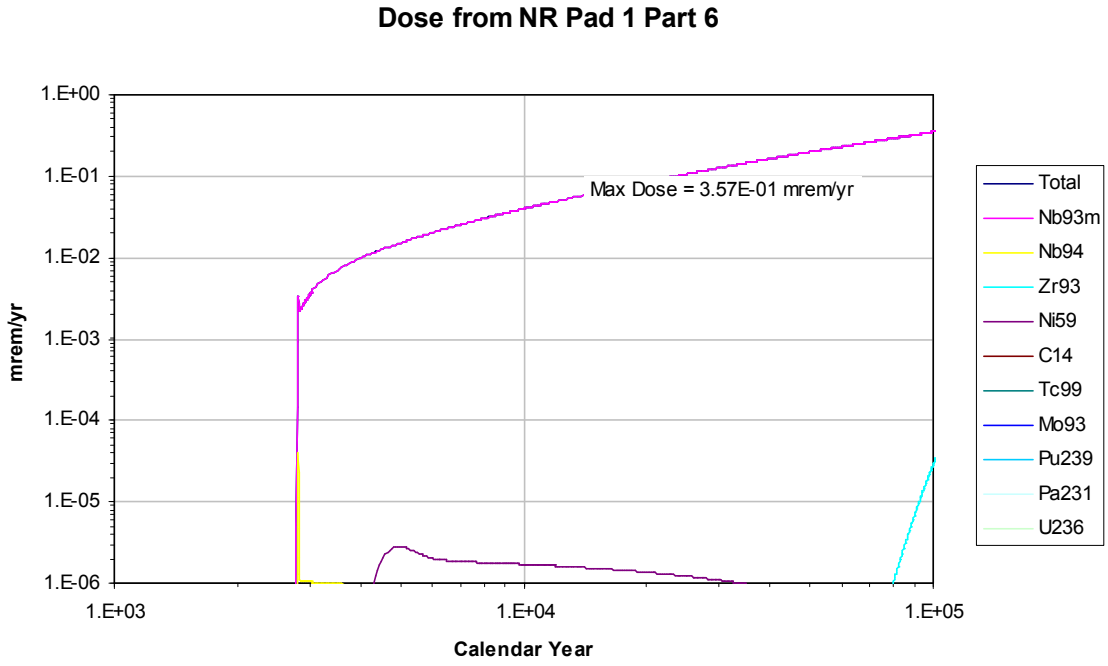


Figure E-154. **NR Pad 1 Part 6 Total Direct Dose by Isotope**

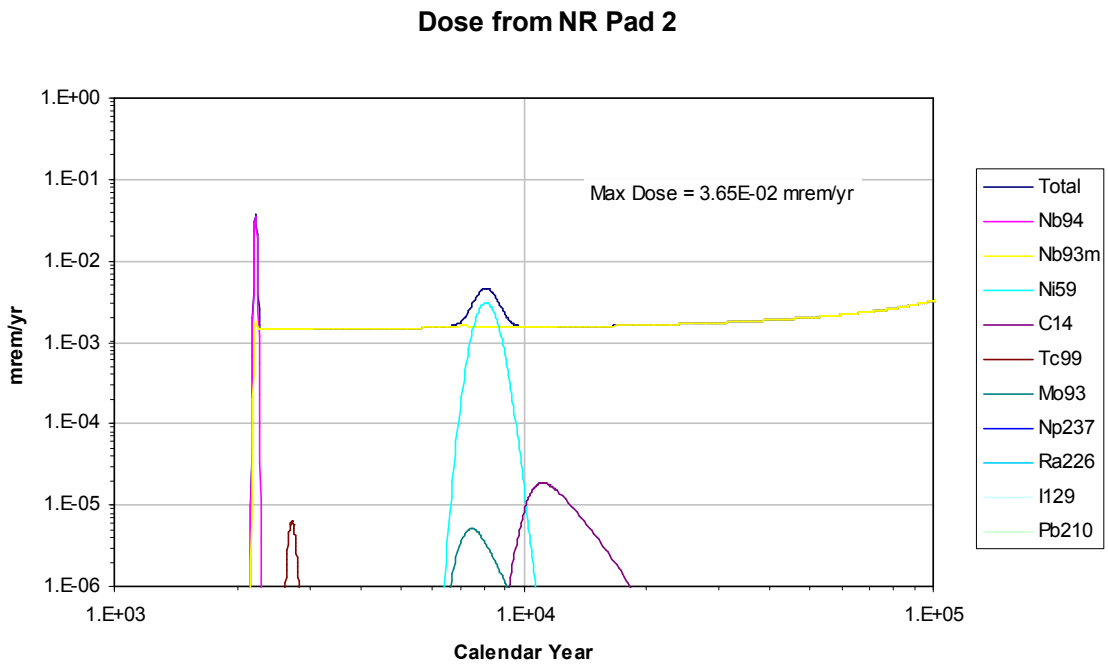


Figure E-155. **NR Pad 2 Total Direct Dose by Isotope**

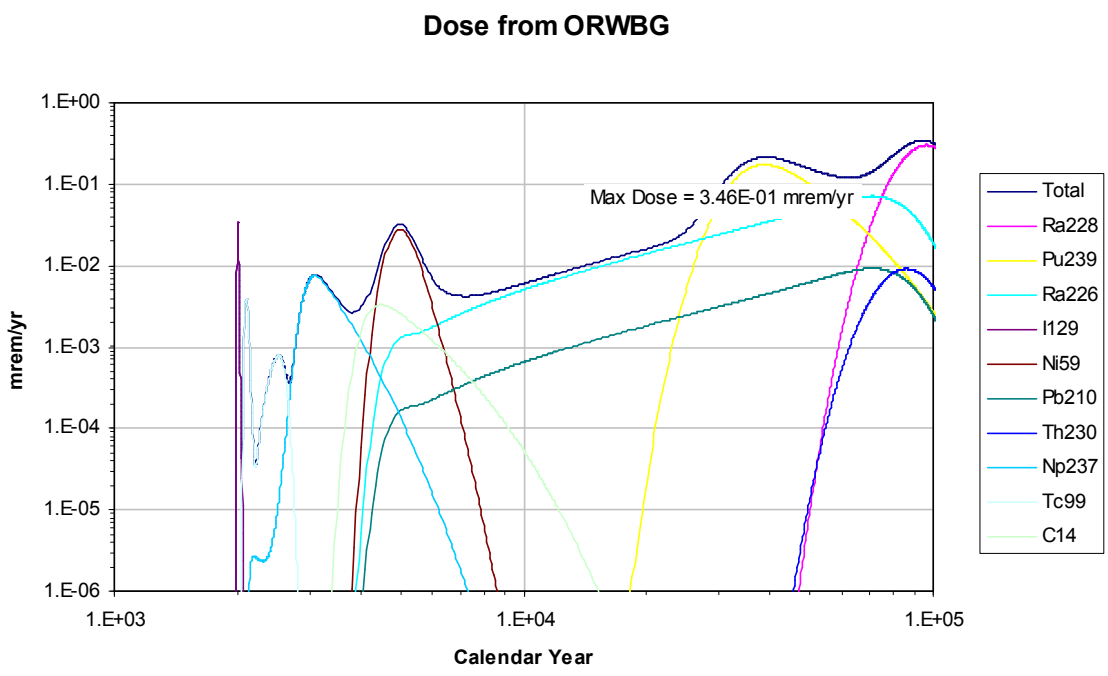


Figure E-156. **ORWBG Total Direct Dose by Isotope**

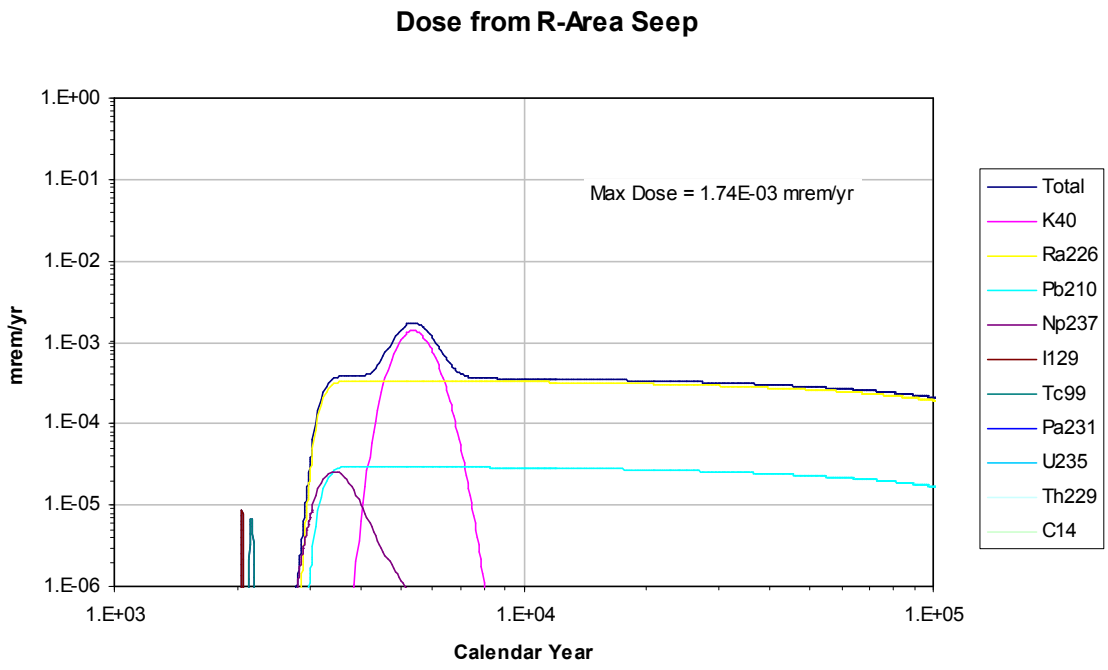


Figure E-157. **R-Area Seep Total Direct Dose by Isotope**

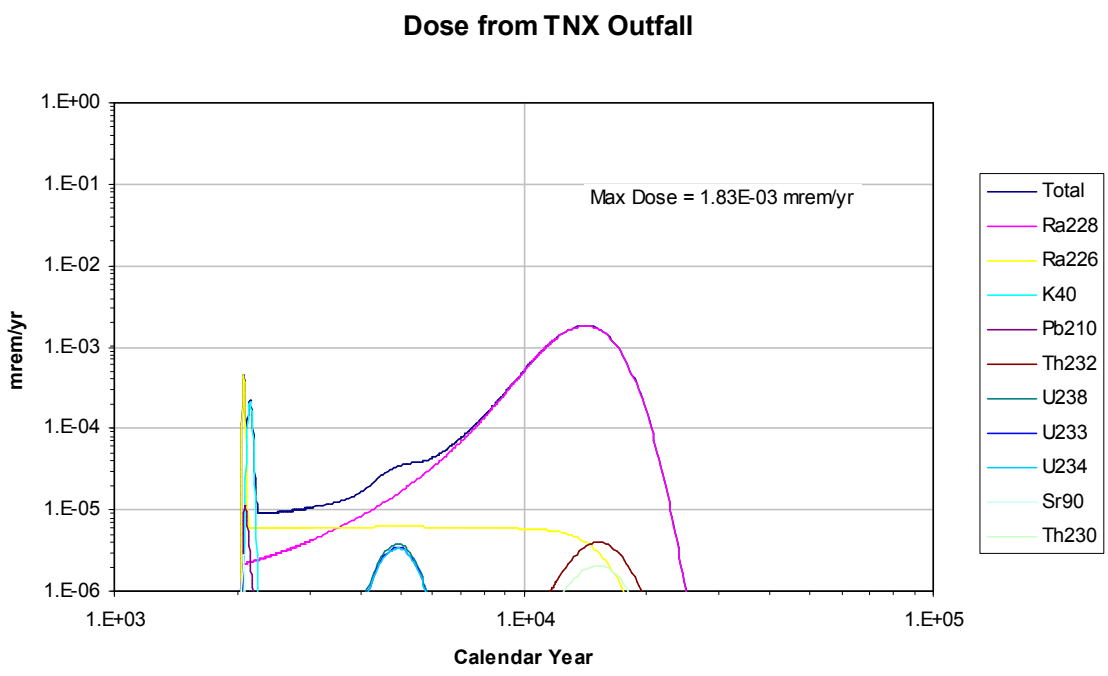


Figure E-158. **TNX Outfall Total Direct Dose by Isotope**

E.3 REFERENCES

Smith, F. G. III, Hiergesell, R. A., Swingle, R. F., Hamm, L. L., and Phifer, M. A. 2009a. *Savannah River Site Composite Analysis: Base Case Deterministic Calculations*, SRNL-STI-2009-00390, Rev. 0, Savannah River National Laboratory, Aiken, SC, July 22, 2009.

Smith, F. G. III, Hiergesell, R. A., Swingle, R. F., Hamm, L. L., and Phifer, M. A. 2009b. *Savannah River Site Composite Analysis: Sensitivity and Uncertainty Calculations*, SRNL-STI-2009-00436, Rev. 0, Savannah River National Laboratory, Aiken, SC, July 30, 2009.

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F. APPENDIX F: LIST OF KEY ASSUMPTIONS

F.1 KEY CA OPERATIONS ASSUMPTIONS

- The Savannah River Site (SRS) land use plan is documented within the Savannah River Site Long Range Comprehensive Plan (DOE 2000) and the SRS End State Vision (DOE 2005). See Sections 4.2.3.1 and 4.2.3.2 for more detailed information concerning the SRS land use plan. This plan states that the entire site will be owned, controlled, and maintained by the federal government, most likely the Department of Energy (DOE), in perpetuity, as established by Congress. Site boundaries will remain unchanged and the site will be used for industrial purposes for future DOE and non-DOE missions. Three land use zones have been established that concentrate future industrial land use operations toward the center of the site to form a central Site Industrial zone for continuing missions. The Site Industrial Support and General Support Zones will accommodate uses of decreasing intensity, particularly as they approach the site's boundaries. This will tend to reconfigure SRS land use and consolidate more intensive activities at the center of the site to minimize the effect on surrounding communities, produce less impact to ecosystems, maintain controlled site access, and ensure the integrity of the established safety and security buffer. Unrestricted public access and residential land use will be prohibited, thus minimizing potential liability from public ownership of land. Due to these land use controls the first publicly accessible location where radionuclide contaminated media originating from SRS can be contacted is at the mouth of the SRS streams, which empty directly into the Savannah River, and include Upper Three Runs Creek, Fourmile Branch, Steel Creek, and Lower Three Runs Creek. The CA Points of Assessment (POA) and the resulting projected CA dose to the public rely upon prohibiting unrestricted public access and residential land use over the entire site (See Section 4.2). Any change in the SRS land use plan that eases the prohibition of unrestricted public access and residential land use over any portion of the existing SRS site will require a reevaluation of the CA POAs and potentially a revision of the CA.
- The SRS End State Vision (DOE 2005) provides a vision for the disposal of all SRS DOE Office of Environmental Management (EM) nuclear material and waste hazards permanently, the decommissioning of all SRS EM facilities and the remediation of all SRS inactive waste sites (see Section 4.2.3.2). As such this document outlines the planned end states for these facilities (demolition or in situ disposal) and waste sites (planned remedial actions). This document does not specifically address facilities with enduring missions such as the SRS Tritium Facilities. Projected end states for the SRS Tritium Facilities have been assigned consistent with that of similar structures addressed in the end state vision. Appendix C Table C-1 lists all of the sources considered within the CA and their projected end state. However the projected end state has only been taken into considered for those facilities/waste sites for which CA source release modeling was performed. These CA source release facilities/waste sites and their projected end states are outlined in Table F-1. A change in the end states of the facilities/waste sites listed in Table F-1 could affect the CA results.

Table F-1. CA Source Release Facilities and Projected End State

Facility / Waste Site	Projected End State
E-Area Intermediate Level Vault (ILV)	Geosynthetic Closure Cap
E-Area Low Activity Waste Vault (LAWV)	Geosynthetic Closure Cap
E-Area Low Level Radioactive Waste Disposal Facility (LLRWDF) / Mixed Waste Management Facility (MWMF)	Geosynthetic Closure Cap (18 acres) / Clay Closure Cap (58 acres)
E-Area Old Radioactive Waste Burial Ground (ORWBG)	Geosynthetic Closure Cap
Z-Area Vault 2	In Situ Disposal / Geosynthetic Closure Cap
F-Area Seepage Basin	Clay Closure Cap
H-Area Seepage Basin	Clay Closure Cap
Naval Reactor Component Disposal Area (NRCDA), 643-7E	Geosynthetic Closure Cap
C-Reactor Building (105-C)	In Situ Disposal
K-Area Complex (KAC)	In Situ Disposal
L-Area Complex (LAC)	In Situ Disposal
P-Reactor Building (105-P)	In Situ Disposal
R-Reactor Building (105-R)	In Situ Disposal
H-Area Canyon / HB Line	In Situ Disposal
Building 232-H	In Situ Disposal
H-Area New Manufacturing Facility (HANM)	In Situ Disposal
H-Area Old Manufacturing Facility (HAOM)	Demolish to Concrete Slab
Tritium Extraction Facility (TEF)	In Situ Disposal
TEF Remote Handing Building, 264-2H	In Situ Disposal

- The Federal Facilities Agreement (FFA) for the Savannah River Site (FFA 1993) is an annually updated agreement between DOE, Environmental Protection Agency (EPA) and South Carolina Department of Health and Environmental Control (SCDHEC), which describes the agreed upon requirements for environmental protection at SRS (see Section 3.5.5). It addresses waste site remediation, facility decommissioning, and high-level radioactive waste tank system closure. As such the changes to the FFA which affect the projected end states for the facilities/waste sites listed in Table F-1 could affect the CA results.

- The Appendix A inventory tables provide the projected end state inventories for all of the facilities/waste sites modeled within the CA. A change to these projected inventories or discovery of an unanalyzed inventory could affect the CA results.

F.2 KEY MODELING ASSUMPTIONS

- One-dimensional transport through the vadose zone and aquifer is adequate. Flow through the vadose zone and aquifer is actually three-dimensional. The flow is approximated as one-dimensional in the CA model using the average flow path lengths and velocities. The full three-dimensional flow will tend to disperse the transported material in both time and space; therefore, the one-dimensional treatment is likely conservative.
- The actual distribution of clay in the aquifer is not important. Preliminary testing showed that the clay distribution is not important for parent isotopes but does affect the results obtained for daughter isotopes. The CA model assumes that it is sufficiently accurate to lump the aquifer clay in a region near the middle of the path. Only clay within designated aquitards is considered taken into account within the CA; clay within aquifers is not.
- Perfect mixing in the SRS streams and Savannah River. Segregation of flow in either the streams or river could create regions of higher contaminant concentration that could lead to higher doses, if that particular water were used. However segregation of flow within the streams at the POAs is highly unlikely, due to the length of flow in the streams from the sources to the Savannah River (as seen in Figure 4.2 the stream length of flow is greater than 5 miles). As outlined in Section 3.2.9.1.2, under normal Savannah River flow conditions, water entering the Savannah River from SRS streams thoroughly mixes with the Savannah River in a few thousand feet. Prior to SRS stream water thoroughly mixing with the Savannah River it hugs the South Carolina bank of the Savannah River from which is unlikely that a resident on the Georgia side would withdraw water.
- Uniform tritium plume distribution throughout the aquifer. More thorough investigation of well information may allow better definition of the plume shape and location. However, the groundwater tritium plumes do not appear to be significant dose sources.
- Uniform distribution of inventory over the disposal site area. The CA model assumes that it is sufficiently accurate to assume that the inventory is uniformly distributed throughout the disposal site and that a single transport path originating from near the centroid of the entire disposal site adequately captures the transport behavior. The actual distribution of the waste inventory could be accounted for more accurately by subdividing the disposal site into smaller units.
- The source inventory appears instantaneously at the date given in inventory report (i.e., prior releases from the source are neglected). In the CA modeling, the estimated closure inventory was released at the date the inventory was specified in the inventory report. In actuality, material will have been released since the start of disposal at the source location. The effects of these prior releases are neglected in the model.

- The radionuclides identified by the screening process (Taylor et al. 2008) are sufficient to capture the CA dose. The screening process resulted in 49 radionuclides to be analyzed in the CA. Three daughter isotopes with half-lives greater than three years were added to the CA screening species resulting in a total of 52 radionuclides used in the CA modeling. The screening analysis started with a comprehensive list of 826 radionuclides; however, only one nominal screening case and three sensitivity cases were run. Expert judgment was used to both reduce and expand the list of isotopes identified through the screening exercise resulting in the original 49 species. The screening tended to identify isotopes with relatively long half-lives and/or low K_ds. The screening results were in general agreement with the list of key radionuclides used for ELLWF and FTF PAs thereby increasing confidence in the results.
- The CA modeling assumes that the generic release modeling, where the source inventory is released on the ground and no credit is taken for transport barriers, results in a generally conservative or equivalent dose estimate. It is possible that impeding the transport of some isotopes would allow the ingrowth of daughters that might cause somewhat higher doses in some cases.
- Steady-state 3D aquifer flow models were employed where the surface conditions were assumed to have negligible impact on overall aquifer flow patterns over the +1000 year simulations. Surface changes such as covers may impact local aquifer conditions. Average rainfall conditions were assumed.

F.3 REFERENCES

DOE (Department of Energy). 2000. *Savannah River Site Long Range Comprehensive Plan*, Savannah River Operations Office, U.S. Department of Energy, Aiken, SC, December 2000.

DOE (Department of Energy). 2005. *Savannah River Site End State Vision*, Savannah River Operations Office, U.S. Department of Energy, Aiken, SC, July 26, 2005.

FFA (Federal Facility Agreement). 1993. *Federal Facility Agreement for the Savannah River Site*, Administrative Docket No. 89-05-FF (Effective Date: August 16, 1993).

Taylor, G. A., McDowell-Boyer, L., Lee, P. L., and Wilhite, E. L. 2008. *Radionuclide Screening Model for the Savannah River Site's Composite Analysis*, SRNS-STI-2008-00117, Rev. 0, Savannah River National Laboratory, Savannah River Nuclear Solutions, Aiken, SC, September 30, 2008.

G. APPENDIX G: DOE-SR COMMENT RESOLUTION

Comment Resolution to the Department of Energy – Savannah River Operations Office (DOE-SR) Review of the Savannah River Site DOE 435.1 Composite Analysis, Revision B

The Department of Energy – Savannah River Operations Office (DOE-SR) assembled a team to review and comment on the Savannah River Site (SRS) DOE 435.1 Composite Analysis (CA), Revision B. Table G-1 below provides the name and affiliation of each of the DOE-SR SRS CA review team members. The initials (see Table G-1) of the team member who made a comment are provided at the beginning of that comment. The Savannah River National Laboratory (SRNL) resolution for each comment, which has been approved by the DOE-SR SRS CA review team, is provided immediately below that comment. Resolution of the comments resulted in production of Revision 0 of the SRS CA. The comments have been divided into two categories:

- Those made in response to an applicable DOE Low-Level Waste Disposal Facility Federal Review Group (LFRG) criterion
- General comments

Comments made in response to an applicable LFRG criterion are provided in Section G.1, and such comments are preceded by the applicable criterion (from *Radioactive Waste Management Manual* or the *Low-Level Waste Disposal Facility Federal Review Group Manual*, Revision 3). General comments are provided in Section G.2.

Table G-1. DOE-SR SRS CA Review Team Members

Team Member	Affiliation	Initials
Bruce Crowe	NSO EM Science Advisor	BC
Sonny Goldston	SRNS RI&BM	SG
Ginger Humphries	SRNS RI&BM	GH
Kevin Mitchell	Strata-G	KM
John Patterson	Strata-G	JP
Howard Pope	Independent contractor	HP
Sherri Ross	DOE-SR WDPD	SR
Mike Simmons	DOE-SR WDPD	JMS
Danny Smith	Mentat (supporting EM-41)	DS

NSO EM = Nevada Site Office Environmental Management
SRNS RI&BM = Savannah River Nuclear Solutions Regulatory Integration and Business Management
WDPD = Waste Disposition Programs Division

G.1 CRITERION SPECIFIC COMMENTS AND RESPONSES

Criterion IV P. (3) from *Radioactive Waste Management Manual, Chapter 4 Low-Level Waste Requirements,* DOE M 435.1-1, 7/09/99

IV P. (3) Composite Analysis. For disposal facilities which received waste after September 26, 1988, a site-specific radiological composite analysis shall be prepared and maintained that accounts for all sources of radioactive material that may be left at the DOE site and may interact with the low-level waste disposal facility, contributing to the dose projected to a hypothetical member of the public from the existing or future disposal facilities. Performance measures shall be consistent with DOE requirements for protection of the public and environment and evaluated for a 1,000 year period following disposal facility closure. The composite analysis results shall be used for planning, radiation protection activities, and future use commitments to minimize the likelihood that current low-level waste disposal activities will result in the need for future corrective or remedial actions to adequately protect the public and the environment.

JMS-In order to satisfy HLW tank closures and Saltstone disposal we should clearly communicate 10,000 year results as well as the 1,000 year case.

Response: The summary tables in the Executive Summary and in Section 10 have been changed to present results for both the CA 1,000-year period of assessment and the 10,000-year period of assessment. Figures showing the results have also been added. Associated text has also been added to clarify the discussion.

DS-TYPO: Section 2.1, first line of paragraph 3 – the date cited should be June, 1998, not June 1988.

Response: The date in Section 2.1, paragraph 3, first line has been revised to June 1998.

DS-The CA review criteria should be extracted from the most recent version of the *Low-Level Waste Disposal Facility Federal Review Group Manual*, which is Revision 3 (June 2008), not Revision 2 (October 2006). I do not recall making any changes to the text of the CA criteria when I led the team that prepared the most recent revision of the LFRG Manual. I did a cursory comparison of each criterion in this table with the corresponding Revision 3 criterion and did not find any differences. However, all of the identifying numbers are different (e.g., 3.3.1 instead of 3.2.1; 3.3.1.1 instead of 3.2.1.1), and a word-by-word comparison should be performed to confirm my quick check. The citation in the cell to the left should be changed to Revision 3, June 2008. Any other references in the CA to Revision 2 of the LFRG Manual should be changed to Revision 3 and the correct date of publication (e.g., citations on pages 2-2, 5-4, 5-5, and 12-8 of the CA). The citation for a draft of Revision 3 of the LFRG Manual in the list of references (page 12-8) should be deleted.

Response: All references to the LFRG Manual have been revised to:

DOE (Department of Energy). 2008. Low-Level Waste Disposal Facility Federal Review Group Manual, Revision 3, U.S. Department of Energy, Washington, DC, June 2008.

Criteria 3.3.1 through 3.3.11.1 from *Low-Level Waste Disposal Facility Federal Review Group Manual*, Revision 3, June 2008

3.3 CA Review Criteria

3.3.1 Site and Facility Characteristics

3.3.1.1 The CA provides a coherent presentation of the relevant descriptive information concerning the disposal site, its location on the DOE site, and its proximity to other sources of radioactive material. The sources of radioactive material are described including relevant features that could influence radionuclide release and migration.

GH-Section 3.1, add N-Area to waste management bullet and add H-Area to Other radioactive missions…

Response: N-Area and H-Area have been added to the bullets.

GH-Section 3.1.3, 2nd paragraph, 2nd sentence starts with "All reactor…"; this sentence is confusing please reword.

Response: The first three sentences of the second paragraph of Section 3.1.3 have been revised to read as follows:

The SRS production reactors were designed and constructed by DuPont. The reactors initially used a combined fuel and target assembly, consisting of aluminum-clad natural uranium. In 1968 the reactors converted to highly-enriched uranium fuel assemblies and separate depleted uranium target assemblies (DOE 1997a). The fuel and target assemblies were loaded and unloaded into the reactor vessel remotely.

GH-Section 3.1.3, 4th paragraph, 2nd sentence delete "many times".

Response: The phrase "many times" has been deleted from Section 3.1.3 4th paragraph 2nd sentence.

GH-Section 3.1.4.1, 3rd par, next to last sentence – doesn't H Canyon still process some Pu?

Response: This section has been revised and rearranged as follows to make it clear that H Canyon currently processes some Pu:

Liquid separations involved liquid extractions to purify plutonium, uranium and other elements (Reed et al. 2002). F Canyon and H Canyon are the two areas largely devoted to wet separation processes. Wet processes involved dissolving reactor fuel and target assemblies, and chemically treating the resulting solution to separate unwanted from wanted material. A chemical process developed for use at the SRS, called PUREX, has become the standard for Pu and U separation. A variation on the PUREX process, called HM, was used in processing enriched uranium (Reed et al. 2002). The process buildings were designed to allow work to be performed remotely. Waste from the wet separations process was transferred to the radioactive liquid waste storage tanks at both F- and H-Area.

F Canyon was constructed in the early 1950s, and began operation in 1954 (SRS 2008c). The building that housed F Canyon also housed FB Line, and together, these were the two main processing facilities at F-Area (WSRC 2007a). F Canyon and FB Line were used primarily to recover Pu239 from reactor target assemblies and depleted U238 was recovered as a by-product. The production mission of FB Line and F Canyon was completed in March 2002, although stabilization and packaging of legacy nuclear materials for long-term storage was carried out until 2005 (SRS 2008c, 2008d). Currently, F-Area is undergoing decommissioning. F-Area support facilities outside the F Canyon Building include the U Processing Facility (FA line), the F-Area Materials Storage (FAMS) facility (See Section 3.4.6 for additional FAMS information), a Naval Fuels Fabrication Facility, Central Laboratories, a Mock-up/Fabrication Facility, and the F-Area Tank Farm (FTF) (SRS 2006).

H Canyon came online in July of 1955 (Reed et al. 2002), and similar to F Canyon, housed HB Line in the same building. Historically H Canyon and HB Line were used primarily to recover U235, Np237, and Pu238 from reactor assemblies (Reed et al. 2002; SRS 2008a). More recently H Canyon has been used to convert weapons-grade highly enriched uranium (HEU) to low-enriched material suitable as fuel in commercial power reactors; disposition research reactor spend nuclear fuel; and process and stabilize scrap Pu (WSRC 2007a; SRS 2008e). HB Line has three current process lines: a line to recycle legacy Pu scrap for purification and concentration to a solid form; a line to produce solid oxide material from Np237 or Pu239 nitrate solutions; and a line to produce Pu238 oxide from nitrate solutions. H Canyon and HB Line remain in operation at the SRS (WSRC 2007a). H-Area facilities outside the H Canyon Building include the Tritium Facilities, the H-Area Tank Farm (HTF), the Receiving Basin for Offsite Fuels (RBOF), the Effluent Treatment Project (ETP) facilities, the Consolidated Incineration Facility (CIF), and various other chemical, industrial, administrative, laboratory, and storage facilities (SRS 2006, WSRC 2007a).

F-Area and H-Area Inactive Process Sewer Lines (FIPSL and HIPSL) extend from Building 221-F and H-Canyon to their respective seepage basins in F- and H-Area. The FIPSL and HIPSL were vitrified clay lines utilized from 1955 to 1982 to transport hazardous and low-level radioactive wastes from Separation, Tritium, and Waste Management facilities in F- and H-Area. Due to deterioration and leakage, the vitrified clay lines were abandoned in place and replaced with a set of high-density polyethylene (HDPE) lines (WSRC 1993a).

GH-Section 3.1.4.1, paragraph 5, add at the end of the last sentence "and are currently planned for the future".

Response: This paragraph has been revised and rearranged along with the rest of Section 3.1.4.1 and the paragraph in question has been revised as follows to make it clear that decommissioning of F Canyon is on-going:

F Canyon was constructed in the early 1950s, and began operation in 1954 (SRS 2008c). The building that housed F Canyon also housed FB Line, and together, these were the two main processing facilities at F-Area (WSRC 2007a). F Canyon and FB Line were used primarily to recover Pu239 from reactor target assemblies and depleted U238 was recovered as a by-product. The production mission of FB Line and F Canyon was completed in March 2002, although stabilization and packaging of legacy nuclear materials for long-term storage was carried out until 2005 (SRS 2008c, 2008d). Currently, F-Area is undergoing decommissioning. F-Area support facilities outside the F Canyon Building include the U Processing Facility (FA line), the F-Area Materials Storage (FAMS) facility (See Section 3.4.6 for additional FAMS information), a Naval Fuels Fabrication Facility, Central Laboratories, a Mock-up/Fabrication Facility, and the F-Area Tank Farm (FTF) (SRS 2006).

GH-Section 3.1.5.7, 2nd paragraph add the specific list of F and H Area seepage basins.

Response: Section 3.1.5.7, 2nd paragraph has been revised to include the specific list of F and H Area seepage basins as follows:

In the Separations Areas, three unlined F-Area seepage basins (904-41G, 904-42G, and 904-43G) cover approximately 6.5 acres. These basins received approximately 1.8 billion gallons of LLW solutions originating from the processing of uranium slugs and irradiated fuel from 1954 through 1988. The low activity effluents contained a wide variety of radionuclides and dissolved metals. The four unlined seepage basins in H-Area (904-44G, 904-45G, 904-46G, and 904-56G) cover approximately 15.5 acres, and received about 1.6 billion gallons of waste solutions similar to that in the F-Area basins. In addition to those waste solutions, they received effluent from tritium facilities and the RBOF. The seepage basins essentially functioned as designed, but the acidic nature of the basin influent caused mobilization of metals and radionuclides that resulted in groundwater contaminant plumes.

GH-Section 3.1.5.7 last paragraph, last sentence is confusing – please rewrite.

Response: Section 3.1.5.7, last paragraph has been rewritten as follows to add clarity:

In T-Area, wastewater from pilot-scale tests for the DWPF and the General Separations Area (GSA) was discharged through a process sewer line to the Old TNX Seepage Basin from the mid-1950s until 1980. The old seepage basin received waste containing mercury, other heavy metals, depleted uranium and other radionuclides at low levels. The old basin was closed and backfilled with clean soil in 1981. In 1980, a new basin became operational and discharges to the new basin were carefully regulated to exclude hazardous and/or radioactive wastes (WSRC 1999b). In 1988 the non-hazardous and non-radioactive discharges to the new basin were rerouted to the TNX Effluent Treatment Plant (SRS 2003h).

HP-Section 3.1.5 Waste Management (E-, F-, H-, N-, S-, and Z-Area) add mixed waste to the first sentence.

Response: Mixed waste has been added to Section 3.1.5, first paragraph, first sentence.

HP-Section 3.1.5.1, "The E-Area LLW Burial Grounds is comprised …." –move this section to the first paragraph otherwise it is confusing on what facility you are talking about.

Response: The Section 3.1.5.1 paragraph beginning with "The E-Area LLW Burial Grounds is comprised …" has been moved to become the first paragraph of the section.

HP-Section 3.1.5.1, "The E-Area facility is" – unclear which facility (ies) you are talking about. Clear and consistent nomenclature should be maintained in this section.

Response: "The E-Area facility" has been changed to "The E-Area LLW Burial Grounds" in order to maintain consistent nomenclature.

HP-Section 3.1.5.1, "In 1998, SRS installed a soil cover over a majority of the ORWBG to reduce surface radiation levels" – I would add that it was closed under RCRA/CERCLA.

Response: The last two sentences of the third paragraph of Section 3.1.5.1 have been revised to read as follows:

> *As part of an Interim Action RCRA/CERCLA ROD, SRS installed a soil cover over a majority of the ORWBG to reduce surface radiation levels in 1998. As part of a Final RCRA/CERCLA ROD, SRS installed a final geosynthetic closure cap over the ORWBG to reduce storm water infiltration to the waste layer and mitigate contaminant migration to the groundwater in November 2007 (SRS 2007a).*

GH-Section 3.1.6, last paragraph, shouldn't "…are being remediated" be changed to "have been remediated?

Response: No, Section 3.1.6, last paragraph, first sentence should not be changed from, "…are being remediated" to "have been remediated", because, according to Cathy Lewis of ACP, the groundwater is still under remediation.

No change to the CA was required.

GH-Table 3-1, change 2006 Population Estimate to 2007.

Response: Table 3-1 "2006 Population Estimate" has been changed to "2007 Population Estimate".

GH-Section 3.2.6.1.1, Figure 3-11 should be revised to show Pen Branch intersecting Steel Creek.

Response: Figure 3-11 shows the major SRS tributaries to the Savannah River, the Savannah River itself, and their respective floodplains as part of the SRS topographic discussion. Figure 3-11 was obtained from the pdf of the SRS High Level Waste Tank Closure EIS and as such can not be readily modified. While the figure does not show Pen Branch intersecting Steel Creek, it also does not show Pen Branch as a direct tributary to the Savannah River. Therefore for purposes of the topographic discussion the Figure is adequate. However the text has been revised to acknowledge that Pen Branch enters Steel Creek prior to discharging to the Savannah River.

GH-Sections 3.2.9.2.1.7, 3.2.9.2.1.8, and 3.2.9.2.1.9 should be connected to the Upper Three Runs from a format standpoint.

Response: The information in current Sections 3.2.9.2.1.7, 3.2.9.2.1.8, and 3.2.9.2.1.9 have been added to the end of Section 3.2.9.2.1.1, Upper Three Runs.

GH-Figure 3-30, suggest delete this figure. It's a different format and duplicative of UTR.

Response: Figure 3-30 has been deleted from reference in association with current Sections 3.2.9.2.1.7, 3.2.9.2.1.8, and 3.2.9.2.1.9, which will be added to the end of Section 3.2.9.1.1, Upper Three Runs and Figure 3-11 has been referenced instead. However Figure 3-30 has not been deleted, but it has been moved to the end of Section 3.2.9.3.1, which appropriately references this figure. Figure 3-30 is also appropriately referenced in Section 3.2.10.2.

HP-Section 3.4.4.1, General Performance Assessment and Facility Information for the Z-Area Saltstone Disposal Facility, add the SDF is permitted by SCDHEC under Class 3 landfill regulations.

Response: Section 3.4.4.1, 1st paragraph, 2nd sentence has been revised to read as follows:

> *The SPF is permitted as a wastewater treatment facility per the SCDHEC Regulations R.61.67, and the SDF is permitted as a Class 3 Industrial Solid Waste Landfill per SCDHEC Regulations R.61.107.16.*

SG-Section 3.4.5.3 TRU Pad 1 PA Requirements
Section 3.4.6.2 F-Area Materials Storage PA Requirements
Comment: Normally, the requirements from 40CFR191 are stated as below. It may be better to trace the statements made in the CA to each of these items: Containment, Assurance, Individual Protection, and Groundwater.
"The 40 CFR 191 requirements are as follows: containment requirements (CRs), assurance requirements, individual protection requirements (IPRs), and groundwater protection standards. The results of the SA indicate that there is a reasonable expectation of meeting all the requirements of 40 CFR 191."

Response: The 40CFR191 requirements outlined in burger dots in Sections 3.4.5.3 and 3.4.6.2 have been labeled as containment requirements (CRs), assurance requirements, individual protection requirements (IPRs), and groundwater protection standards as appropriate.

JMS-3.1.5.7, page 3-16: Statement that radionuclides had not moved further than two feet deep is incorrect. Granted, a majority of the Cesium 137 was held up within the first two feet but four feet was removed from some basins while still leaving activity behind at greater depths. Suggest re-wording.

Response: Section 3.1.5.7, 5th paragraph, 4th sentence has been revised to read:
In 1983, characterization of the area showed that the bulk of the radionuclides were contained within the upper few feet of the bottom of the basins.

JMS-3.4.5.1, page 3-221: In the 1st paragraph it should state that a PA "may be" prepared, not "is being". In the 3rd paragraph the soil cover "is" being removed, not "may".

Response: Section 3.4.5.1, 1st paragraph, 3rd sentence has been revised to read:
The PA may be prepared to evaluate the on-site disposal of TRU Pad 1 waste against both the 40 CFR Part 191 standards and DOE Order 435.1-1 performance measures for disposal of LLW.
Section 3.4.5.1, 3rd paragraph, 2nd sentence has been revised to read:
Although TRU waste on TRU Pad 1 is buried under an interim soil cover, it is considered an operational facility because the waste is considered to be stored rather than disposed.

DS-Section 3.2.9, page 3-84, paragraph 2 – The reference to Figure 3-11 should be Figure 3-20; Figure 3-11 does not indicate the location of Steel Creek Landing and Figure 3-20 does show its location.

Response: Section 3.2.9, page 3-84, 2nd paragraph, 3rd sentence: the reference has been changed from Figure 3-11 to 3-20.

DS-Section 3.2.9, pages 3-86 and 3-87, Figures 3-21 and 3-22 – These figures are very informative but it is difficult to identify relevant locations such as Steel Creek Landing. Key reference points should be indicated with arrows, labels, or some similar device.

Response: Labels have been added to Figure 3-21 denoting the location of Fourmile Branch, Pen Branch, Savannah River, Savannah River swamp, and Steel Creek Landing. Labels have been added to Figure 3-22 denoting the location of Fourmile Branch and Savannah River.

DS-Chapter 3 – Many of the locations and landmarks cited are not shown on a map or otherwise described. Examples include page 3-85, paragraph 3 reference to the Highway 301 Bridge; page 3-90 references to Highway 278, SRS Road A, and SRS Road C; page 3-82, Table 3-15 references to the SRS Boat Dock and Steel Creek at Hattiesville Bridge. The information associated with those locations will be much more useful if the locations are defined.

Response: Figure 3-1 has been changed to indicate where Highways 125 and 278 are. The designations within the text using SRS Road numbers have been changed to refer to SRS areas, which are depicted on the various maps in the CA. Unfortunately, showing the SRS Road numbers on the maps makes the maps OUO.

DS-Page 3-90, Figure 3-23 – Define "Plant North."

Response: The following note was added to Figure 3-1 and is applicable all figures contained within this CA, which are based upon Plant North:

Note: During design of the SRS facilities in the 1950's, an SRS site grid system was established for facility location, based upon a Plant North, which was rotated 36°22' west from True North. This connotation applies to all figures contained within this CA that are based upon Plant North.

DS-Page 3-185, Table 3-23 – The footnote "a" for four entries in the MCL column should be changed to "b" and the footnote "b" for two entries in the Major Contaminants column should be removed.

Response: Table 3-23 footnotes have been changed as outlined in the comment above.

3.3.2 Radioactive Sources/Release Mechanism

3.3.2.1 The CA identifies all sources of radioactive material in the ground that could contribute to the potential future doses from the LLW disposal facility. Sources selected for the CA and the reasons for excluding any source are justified. Potential sources of radioactive material to be considered include wastes disposed of prior to 1988, other LLW disposal facilities, transuranic waste or alpha LLW disposal, buildings, tanks, cribs, spills, ditches, seepage basins, and leaks.

JMS-(Same as previous comment) 3.1.5.7, page 3-16: Statement that radionuclides had not moved further than two feet deep is incorrect. Granted, a majority of the Cesium 137 was held up within the first two feet but four feet was removed from some basins while still leaving activity behind at greater depths. Suggest re-wording.

Response: Section 3.1.5.7, 5th paragraph, 4th sentence has been revised to read:
In 1983, characterization of the area showed that the bulk of the radionuclides were contained within the upper few feet of the bottom of the basins.

DS-Table 8-1 does not include several facilities and sub-unit facilities, which are listed as potential CA sources in the FY2008 Annual Review (*E-Area Low-Level Waste Facility Performance Assessment and Composite Analysis*, SRNS-RP-2008-01314, Revision 0, January 2009). Examples include the Soil and Debris Consolidation Facility, H-Area inactive process sewer lines, and the F-Area sand filter. These items should be included in the *Savannah River Site DOE 435.1 Composite Analysis* with a rationale for not considering them in the CA if they are insignificant sources.

Response: The Soil & Debris Consolidation Facility listed in Table 11 of the FY2008 Annual Review (SRNS-RP-2008-01314) does not exist. The possibility of such a facility had been under evaluation with SCDHEC and EPA for the excavation and consolidation, in one location, of radioactively contaminated RCRA/CERCLA waste from approximately 20 waste sites (WSRC 1997). Such a facility was never approved nor built at SRS. Therefore the next PA/CA annual review will delete the reference to the Soil & Debris Consolidation Facility from the list of CA source locations.

WSRC (Westinghouse Savannah River Company). 1997. Alternative Screening Report Radioactive Soils/Debris Consolidation Facility / Off Unit Disposal, WSRC-RP-96-893, rev. 1, Westinghouse Savannah River Company, Aiken, SC. May 1997.

The H-Area inactive process sewer lines listed in Table 11 of the FY2008 Annual Review (SRNS-RP-2008-01314) are listed as HIPSL within Table 8-1 of the CA and the associated inventory for HIPSL is provided in Table A-39 of volume II of the SRS CA.

The F-Area Sand Filter was included in the CA Inventory as part of the F-Canyon Complex inventory (see Table A-27). This has been made clear in the Section 8.2.4.3 F-Canyon and FB Line write-up. The H-Area Sand Filter, which was included in the inventory separate from the H-Canyon (see Table A-45), resulted in a maximum generic dose of 1.2E-08 mrem/yr (see Table 9-19). The H-Area Sand Filter is associated with H-Canyon, for which CA generic release modeling resulted in a maximum generic dose of 1.25 mrem/yr; whereas the F-Canyon CA generic release modeling included the F-Area Sand Filter and together they resulted in a maximum generic dose of 2.8E-03 mrem/yr. Therefore it would be anticipated that if the CA generic release modeling of F-Area Sand Filter had been performed separately, it would have resulted in a maximum generic dose significantly less than that for the H-Area Sand Filter (i.e., <<1.2E-08 mrem/yr); that is alone it would have been an even more insignificant dose than the combined F-Canyon dose.

Even though all the facilities listed in the comment have been adequately addressed within the CA, additional CA inventory maintenance will be conducted. Section 11.1 outlines current CA maintenance items outlined within the 2009 SRNS PA/CA Maintenance Plan (SRNS 2009). Section 1.2.9 of SRNS 2009 includes the annual review and update of the SRS CA Radionuclide Inventory Database. Additionally there are several items in Table 11-2 pertinent to potential future improvements to the SRS CA Inventory. In addition SRNL is actively pursuing the re-evaluation of SRS facility and waste site lists to identify any facilities or waste sites with a projected end state inventory, which may have been overlooked in the 2009 CA Inventory. This activity has been added to Table 11-2.

3.3.2.2 The CA identifies and quantifies all radionuclides present in the LLW disposal facility and all other contributing sources of radioactive material that could contribute significantly to the total potential dose. Inventory estimates included in the analysis are justified. The estimates of radionuclide species and inventories in the sources selected for consideration are derived from referenced documentation or data summaries presented in the CA and are based on existing records, process knowledge, or site investigations (e.g., Remedial Investigations, Feasibility Studies). Any radionuclides that are screened from the analysis are identified and their exclusion justified as being insignificant contributors to the total dose estimated in the analysis.

HP-Section 3.5.9, Safety Basis Documentation, the Reactor safety basis should be added to this list.

Response: Section 3.5.9 is not a listing of all SRS safety basis documentation, but it provides a listing of those safety basis documents that were used to establish inventories for the CA. The inventories for the SRS Reactors were not based on safety basis documents, but upon SRNL and ACP (D&D) inventory specific documents. Therefore the Reactor safety basis documentation should not be listed in Section 3.5.9.

No change to the CA was required.

BC-A CA screening model was implemented in GoldSim™. The GoldSim™ model implements the major components of the conceptual model of the SRS site with conservative assumptions in multiple modeling steps. Model results are carried out through dose analysis using a screening criterion of 1% of the dose constraint (i.e., 0.3 mrem/yr). Each step of the screening process is traceable through the GoldSim™ model and the screening results are compared with PA results from multiple areas. The CA meets the model requirements of Criteria 3.2.2.2.

Response: We concur; no further response was required.

JP-Page 8-1 establishes the hierarchy for inventory estimates. The use of sampling and analysis data should be of primacy. Existing documented inventories may be only perpetuating dated and/or unreliable process knowledge. Overall, having an approach to document inventories across the SRS in a common manner was an excellent idea.

Response: We established the inventory development hierarchy to focus on using available data to facilitate including the entire SRS in the current analysis. Many of the inventories, such as those of closure sites, have been developed from sample analysis data. As the CA is maintained, we will pursue actions to reduce uncertainty, which will include refining the inventories of the most significant sources, which may include collecting additional samples for analysis. See Section 11.1 where the current CA Maintenance Plan Section 1.2.9, Maintain the SRS CA Radionuclide Inventory Database, is discussed.

No change to the CA was required.

3.3.2.3 The known physical and chemical characteristics of the radioactive materials considered in the CA, the site characteristics, and the effects of the Comprehensive Environmental Responses, Compensation, and Liability Act (CERCLA) actions prescribed in the Record of Decisions (RODs) or similar binding agreements such as those associated with Decontamination and Decommissioning (D&D), are included in the generation of the source terms and the transport of the radionuclides. Extrapolations are made and justified from known data to estimate radionuclides and inventories where clear information does not exist.

JP-Assess inventory loading probabilistically to determine sensitivity to downstream doses.

Response: Future work, discussed in Section 11 of the CA, will focus on reducing uncertainty. One element of this is to develop distributions for inventories of significant sources. The initial effort will focus on the Lower Three Runs Integrator Operable Unit. Table 11-2 has been revised to show emphasis on developing distributions for the most significant sources.

JP-Use sampling data or take more samples – need empirical data in general. Documented inventories may simply be bad PK assumptions being perpetuated so I'm not sure if it is the best number to use as the CA has done as a rule.

Response: We established the inventory development hierarchy to focus on using available data to facilitate including the entire SRS in the current analysis. Many of the inventories, such as those of closure sites, have been developed from sample analysis data. As the CA is maintained, we will pursue actions to reduce uncertainty, which will include refining the inventories of the most significant sources, which may include collecting additional samples for analysis. See Section 11.1 where the current CA Maintenance Plan Section 1.2.9, Maintain the SRS CA Radionuclide Inventory Database, is discussed.

No change to the CA was required.

3.3.2.4 Source terms and flow and transport models in the CA are commensurate with the available data consistent with the PA, incorporate the important characteristics identified in the PA, and provide outputs consistent with the PA.

JMS-9.1.3, page 9-4: The base case conceptual model assumes the residential POA will not have access to on-site streams because of prohibition by the current land use plan. However, typically we are not allowed to credit institutional controls past 100 years. Please validate using a land use plan as the basis for the residential POA. I realize; however, sensitivity runs have been performed assuming a loss of institutional controls. May be the original statement could have a caveat as such added.

Response: The prohibition on crediting institutional controls no more than 100 years is a feature of the requirements for a PA, not a CA. The FORMAT AND CONTENT GUIDE FOR U.S. DEPARTMENT OF ENERGY LOW-LEVEL WASTE DISPOSAL FACILITY PERFORMANCE ASSESSMENTS AND COMPOSITE ANALYSES, December 7, 1999, states in Section C.4:

> *The public dose limit applies only to members of the public. Thus, it applies only beyond the boundary of land controlled by DOE. Currently, land controlled by DOE extends to the boundary of the entire DOE site. However, the land controlled by DOE for purposes of radiation protection of the public may be assumed for the composite analysis to shrink in the future and should be consistent with site-specific plans required by DOE policy for land and facility use. Site-specific plans for land and facility use should be referenced in the composite analysis. If plans for long-term land and facility use are not available, reasonably conservative assumptions should be made (and justified) to determine the point(s) of assessment for the composite analysis (see Section C.2.4).*

No change to the CA was required.

GH-Section 3.4.1.5 should Cs137 be add to the list of radionuclides?

Response: The list of radionuclides provided in Section 3.4.1.5 is the correct list of radionuclides for which groundwater transport calculations were performed within the 2008 ELLWF PA (see 2008 ELLWF PA Part B Table 1-1 and Part C Table 4-3). A radionuclide screening was conducted as part of the ELLWF PA. A conservative radionuclide screening model was implemented, following the NCRP methodology (NCRP 1996). This model started with 826 radionuclides, including Cs137, and screened out radionuclides that contribute less than one percent of the dose or concentration limits. In this ELLWF PA radionuclide screening process Cs137 was screened out from further consideration. Therefore groundwater transport of Cs137 was not analyzed within the 2008 ELLWF PA. The primary difference relative to Cs137 between the ELLWF PA and the SRS CA is that a significant amount of Cs137 exists within site streams, which does not require transport through the subsurface prior to reaching the CA exposure point. This makes Cs137 a necessary consideration for the CA, but it is not applicable with respect to groundwater transport in the ELLWF PA as discussed above.

No change to the CA was required.

KM-Page 7-13, Section 7.2.2.4, last sentence says "Dilution within the Savannah River is not included". However, other text talks about dilution in the Savannah River, such as in the next section, 7.2.2.5.

Response: Within Section 7.2, Radionuclide Screening, dilution in association with the Savannah River is discussed three times: 1) Section 7.2.2.4 last sentence; 2) Section 7.2.2.5 last sentence; and 3) Section 7.2.2.6 next to last sentence. The radionuclide screening was conducted assuming stream flow rates of 10 and 50 cfs as outlined in Table 7-5. The average flow rate of Fourmile Branch, the smallest SRS tributary to the Savannah River, is 32 cfs (see Table 9-17). Whereas the average Savannah River flow rate is approximately 10,000 cfs at SRS (see Section 3.2.9.1.1). Therefore the radionuclide screening was conducted at flow rates associated with SRS streams rather than that associated with the Savannah River. The dilution statements have been clarified to be consistent with this understanding of how the radionuclide screening was conducted relative to stream flow rates.

3.3.3 Performance Measures

3.3.3.1 The CA presents an assessment using the time of 1,000 years for exposures to hypothetical members of the public with all disposal facilities closed, D&D completed, and operations at the DOE site terminated. The assessment establishes a realistic case for comparison with the dose constraint (30 mrem/yr) and dose limit (100 mrem/yr).

HP-Section 1.0 last paragraph – instead of saying the performance measures are met at 10K years, provide what the measures are – like was provided for 1K yrs.

Response: The Executive Summary has been modified in response to this and other comments to show results for both the 1,000-year and the 10,000-year periods of assessment. The revised text has made it clear that the CA performance measures (i.e., 100 mrem/year primary dose limit and the 30 mrem/year dose constraint) are not exceeded even over the 10,000-year period.

HP-Section 3.4.3.3, H-Area Tank Farm Performance Assessment Requirements – add 100 mrem for the chronic intruder for DOE Order requirements.

Response: The DOE Order 100 mrem chronic intruder limit has been added to the list of requirements in both Sections 3.4.2.3 (FTF) and 3.4.3.3 (HTF).

3.3.4 Point of Assessment

3.3.4.1 The point of assessment is the publicly accessible point of maximum impact reasonably expected for future members of the public for the time period of assessment. The point of assessment is justified and is supported by land use plans or reasonably conservative assumptions that are justified.

HP-Agree

Response: We concur; no further response was required.

3.3.4.2 Changes in the point of assessment as a function of time are justified.

HP-Agree

Response: We concur; no further response was required.

3.3.5 Assumptions

3.3.5.1 Assumptions incorporated into the analysis, including those related to the radionuclides to be considered, the inventories of radionuclides, the source term evaluation and the transport of radionuclides, are identified, justified, and consistent with the conceptual model of site behavior presented in the PA conducted on the LLW disposal facility.

BC-The GoldSim™ model used for the radionuclide screening is a simplification of the base case deterministic GoldSim™ model and both implement the conceptual model of flow and transport in the SRS site. There are minor concerns with the integration of site models with a regional groundwater flow model and the documentation of the linkage with the regional model. This issue is judged not to be significant given the flow boundary control by the streams and the sensitivity scenarios evaluated for the effect of changes in the groundwater divide.

Response: SRNL concurs that the lack of linkage between current SRS groundwater models and a regional model is not a significant concern, given the hydrogeologic data supporting the boundary controls of the SRS groundwater models. Currently an SRS regional model encompassing the entire SRS does not exist, from which to produce boundary controls for SRS groundwater models with greater grid resolution that encompass smaller areas. The boundary controls for three of the SRS groundwater models (CKLP, GSA, and R (see Table 9-3)), used to establish flow path parameters for the CA, consist of a combination recharge/drain top surface boundary; a prescribed head with leakance coefficient across the Crouch Branch Confining Unit bottom boundary (see Figure 3-33); and no flow or prescribed head vertical boundaries based upon stream boundary control. The boundary controls for the AM model (see Table 9-3), also used to establish flow path parameters for the CA, consist of a combination recharge/drain top surface boundary; a no flow McQueen Branch bottom boundary (see Figure 3-33); and specified head or head-dependent flux vertical boundaries based upon stream boundary control.

Although the lack of linkage between current SRS groundwater models and a regional model is not a significant concern, Section 11.0 of the CA has maintenance plan items associated with production of an SRS regional model encompassing the entire SRS. Within Table 11-1, building an SRS-wide hydrologic model is listed as a potential ARRA funded CA maintenance item. Within Table 11-2 including low, average, and high water table conditions within the SRS-wide hydrologic model, so that distributions about the aquifer flow path parameters can be developed, is listed as a potential future work item. These Section 11 items have been reworded to make it clear that an SRS regional model encompassing the entire SRS should be developed from which boundary controls for smaller SRS groundwater models with greater grid resolution can be extracted.

3.3.5.2 The CA identifies results, objectives, constraints, or milestones of other DOE programs, Federal, state, or local statutes, or agreements (e.g., D&D programs, Formerly Utilized Sites Remedial Action Program (FUSRAP), CERCLA RODs) that may impact the analysis or conclusions of the CA.

HP-Section 3.4.4.1, General Performance Assessment and Facility Information for the Z-Area Saltstone Disposal Facility, add the SDF is permitted by SCDHEC under Class 3 landfill regulations.

Response: Section 3.4.4.1, 1st paragraph, 2nd sentence has been revised to read as follows:

The SPF is permitted as a wastewater treatment facility per the SCDHEC Regulations R.61.67, and the SDF is permitted as a Class 3 Industrial Solid Waste Landfill per SCDHEC Regulations R.61.107.16.

3.3.6 Modeling

3.3.6.1 The CA presents a reasonable methodology for estimating the transport of radionuclides to the point of assessment from all sources based on and consistent with the available site data.

BC-The CA presents a reasonable methodology for estimating the transport of radionuclides to the point of assessment from all sources based on and consistent with the available site data.

The base case deterministic model systematically traces radionuclides starting with simplified or site specific source release models, through vadose zone transport to the water table. Vadose zone transport is simplified for areas of limited data with more detail added at well studied sites incorporating barriers and backfill, and anticipated behavior of closure covers over time. Transport in the water table is represented for different flow path lengths to bounding surface streams, for variable rock properties with assigned K_ds. The effect of existing groundwater plumes is incorporated into the aquifer models. Streamflow is modeled with representative discharge volume and incorporates radionuclides released from streambed sources. Radionuclide flux as total curies of individual radionuclides per year are calculated at multiple points of assessment representing two major classes of receptor exposure scenarios. Results from the deterministic base case model are integrated and calibrated with results from approved PAs for multiple sources and sites (primarily for the GSA). The CA model meets the requirements for reasonable representation of flow and transport.

Response: We concur; no further response was required.

DS-Section 11.1 of the CA describes selected portions of the CA maintenance plan. The description for Section 3.2 of the maintenance plan states that the results of monitoring and R&D will be considered in the annual review of the CA. On the other hand, the FY 2008 Annual Summary for E Area explicitly states that monitoring results will be compared to the CA results with the ultimate purpose of validating the CA. Validation of the CA modeling with monitoring data should be included in plans for future work, and the language in Revision B of the draft CA should explicitly describe those plans.

Response: We agree. Once this CA has been approved the PA/CA Monitoring Plan will be revised to include monitoring specific to this revised CA to validate the CA results. Reference to this PA/CA Monitoring Plan requirement has been included in Section 11.

3.3.6.2 Analytical and numerical models selected are documented and verified either in referenced publications or in the appendices of the CA.

HP-Section 6.1.1.3 Software QA – mentions GoldSim™ but does not mention the remaining software packages that were used. I suggest a table with the software package and the QA that was applied in this section.

Response: The GoldSim™ software package (code) was the only software package actually utilized during production of the CA that required production of a Software Quality Assurance Plan (SQAP). The MODFLOW, FACT, and PORFLOW codes, discussed in Section 9.2.1, were not actually utilized for production of the CA. Rather existing flow fields obtained from previously developed groundwater models, which utilized these codes as documented within existing reports, were utilized as discussed in Section 9.2.1 in order to produce the aquifer flow parameters for the CA. As such it is not appropriate to discuss the QA associated with these codes within Section 6.1.1.3 of the CA. However Section 6.1.1.3 has been reworded to make it clear that GoldSim™ is the only code requiring treatment in Section 6.1.1.3. Additionally Table 9-3, which addresses these models and other codes, has been revised to add references to the code user manuals and QA/QC documentation and the Section 9.2.1 discussion of Table 9-3 has been revised for consistency with the revised table.

HP-Section 8.1.2.2, "The assessments were performed using the vadose zone contaminant fate and transport model Vadose Zone Contaminant Migration Model – Multi-Layered (VZCOMML©) (Rucker 1999) or equivalent models, such as the Residual Radioactive Materials Computer Code (RESRAD), the Seasonal Soil Compartment Model (SESOIL), Reactor Screening Software 1 (RST1) (WSRC 2006), and the Contaminant Migration Screening (CMScreen) model (WSRC 2006a)." What are the software QA requirements for these models? They should be listed in Section 6.1.1.3.

Response: The GoldSim™ software package (code) was the only software package actually utilized during production of the CA that required production of a Software Quality Assurance Plan (SQAP). The Vadose Zone Contaminant Migration Multi-Layered (VZCOMML©) Soil Screening Program (Rucker 1999; Rucker 2004; Rucker 2007) is utilized by Site Deactivation and Decommissioning (SDD) to determine whether, after deactivation and decommissioning (D&D), the facility remains (i.e., concrete slabs, low grade trenches and sumps, and building foundations) could potentially impact groundwater directly underneath the facility. Of the 273 facilities, which have already undergone D&D, only 38 had a radiological history and of these 38 only six had the potential to impact groundwater based upon the VZCOMML© determination. Results of these SDD determinations along with the associated references are provided within Appendix A (Table A.1-1) of Birk 2008 for each of the 273 facilities, which have already undergone D&D. These determinations were not conducted as part of the CA revision; rather this existing information was simply referenced and used to inform the CA in regards to which D&Ded facilities should be included in the CA inventory. As such it is not appropriate to discuss the QA associated with VZCOMML© within Section 6.1.1.3 of the CA. Residual Radioactive Materials Computer Code (RESRAD), the Seasonal Soil Compartment Model (SESOIL), Reactor Screening Software 1 (RST1) (WSRC 2006), and the Contaminant Migration Screening (CMScreen) model (WSRC 2006a) have not been used to date by SDD in order to determine if a D&Ded facility has the potential to impact groundwater. These are simply other programs that could be used. While Section 6.1.1.3 will not be revised to discuss the QA associated with VZCOMML©, Section

8.1.2.2 has been revised to make clear how the CA has made use of this existing SDD information.

SG-Section 6.1.1.3, Software Quality Assurance and P.9-32 figure 9-13 Changes made by….
Comment: Once the Conceptual Model for the CA was developed in GoldSim™, please explain the configuration control program to ensure no changes are made without testing and approval.

Response: The changes listed in Section 9.2.2.2 Figure 9-13 are not changes to the CA Conceptual Model. These are changes that were made to the CA Dose Module after it was received from the developers, Neptune and Company, Inc. that have nothing to do with the CA Conceptual Model. The CA Dose Module, as developed by Neptune and Company, Inc., has certain input data, including bioaccumulation factors (transfer factors), embedded within the model itself rather than in an external look-up table. Due to this, any necessary revisions to such CA Dose Module input data requires a revision to the CA Dose Module itself. The CA work performed on the carbon dynamics in Fourmile Branch, which resulted in a site-specific carbon-14 water-to-fish transfer factor, was not completed until after Neptune issued the CA Dose Module to SRNL. Therefore once the site-specific carbon-14 water-to-fish transfer factor was developed, it had to be entered into the CA Dose Module as a revision. Additionally once the CA Dose Module was received, SRNL performed a design check of the input data to the CA Dose Module. During that design check it was found that a transcription error had occurred in association with the calcium water-to-fish transfer factor and that the CA Dose Module as received from Neptune had the wrong value for this parameter embedded. Therefore this also required a revision to the CA Dose Module. Finally the function entitled, \Water_Dose\Drinking_Pathways_Dose\CheckSum_WateringDose, which was a CA Dose Module input data check and not part of the actual dose calculation, had to be deleted. This check function had to be deleted because it resulted in a "divide by zero error" when the input radionuclide concentrations to the CA Dose Module were zero in the initial years of the run. This non-critical change also required a revision to the CA Dose Module as received from Neptune. All these changes are documented within the CA Dose Module itself, the CA Base Case Deterministic Calculations report (Smith et al. 2009), and the CA in Section 9.2.2.2.

Control over the development of the SRS CA and the associated conceptual model was primarily maintained through the following:

- *The oversight, review, and approval authority of the CA Core Team over all work and associated reports produced by the individual task team as outlined in Section 6.1.1.1,*
- *The CA planning documents including the Program Plan (Phifer and Cook 2007), the Execution Plan (Phifer and Saldivar 2008a), the Quality Assurance Plan (Phifer and Saldivar 2008b), and the Criteria and Comments Matrices (Phifer et al. 2008),*

- *The CA document control as outlined in Section 6.1.1.2, and*
- *The CA document design checks as outlined in Section 6.4.*

CA development configuration control will be maintained through a Data Management Plan for Archiving the Savannah River Site Composite Analysis and establishment of an associated access controlled SRS CA Archive. The plan and actual archive are currently under development, with a scheduled completion coinciding with issuance of the SRS CA revision 0 to DOE-Headquarters on January 29, 2010. All CA related documents, models, files, etc. will be maintained within the SRS CA Archive so that the full documentation associated with the SRS CA is maintained. Future revisions to the SRS CA will require a separate archive to maintain configuration control. Section 6.1.3 of the CA has been renamed Records and CA Data Management and a brief discussion of the data management plan and archive has been provided.

BC-The GoldSim™ base case model is developed from and compared against results of numerical codes used at multiple locations in the SRS with many of the site modeling studies approved through a formal review process. The computer codes used in the modeling studies are generally accepted codes in the modeling community and the codes are described in the report and referenced in supporting appendices. The CA meets this review criterion.

Response: We concur; no further response was required.

3.3.6.3 Any analytical and numerical models used in the CA for analyzing the transport of radionuclides to the point of assessment are appropriate for the LLW disposal facility and all other contributing sources. The models used in the CA provide calculated results that are representative of the results calculated in the PA for similar wastes in similar disposal facilities.

BC-The GoldSim™ model is developed from, benchmarked against and integrated from an aggregate of models used in multiple PA and related studies of contaminated sites on the SRS.

Response: We concur; no further response was required.

3.3.6.4 Credits for CERCLA/Resource Conservation & Recovery Act of 1976 (RCRA) actions or other actions (e.g., D&D, tank closures) are represented in the conceptual models used in the CA, and are justified by supporting or referenced documentation.

BC-A notable strength of the GoldSim™ base model is the use of EXCEL spreadsheets for cataloging and feeding data into the simulation models. The data are fully traceable and referenced to multiple supporting documents. The SRS CA sets a high standard for transparency and quality of model input data.

Response: We concur; no further response was required.

3.3.6.5 The input data to the models are based on field data from the site, laboratory data interpreted for field applications, referenced literature sources which are applicable to the site, or related analyses performed for the PA. Any assumptions used to formulate input data are justified and have a defensible technical basis.

SG-Section 9.4.2.4.3 TRU Pad 1, P. 9-165
Comment: The case to dispose of TRU pad 1 onsite at SRS is the base case and is very well discussed as a potential even though the current plans are to ship this waste to WIPP. Please explain the different ways TRU Pad 1 and FAMS were analyzed.
Here are some quotes from the CA:
"The impacts on CA dose from two alternatives for the disposition of TRU Pad 1 inventory were evaluated. In the first case, it was assumed that 100% of the TRU Pad 1 inventory was shipped off site. This scenario has a negligible impact on the CA dose during the period of assessment. The maximum dose from TRU Pad 1 between the years 2025 and 3025 is only1.2E-12 mrem/yr because almost none of the plutonium radionuclides or daughters have migrated to the stream by that time. The second case assumes that 100% of the TRU Pad 1 inventory is transferred to H-Canyon/HB-Line for final disposal."
Section 9.4.2.4.5 Inventory Sensitivity Summary Page 9-169
"Shipping the TRU Pad 1 TRU waste to an off-site disposal facility has very little impact upon the CA dose, because leaving it in place only results in a maximum total dose of 1.2E-12 mrem/yr during the 1000-year CA assessment period"
"Transferring the TRU Pad 1 TRU waste to HB-Line has very little impact upon the H-Canyon/HB-Line maximum total dose during the 1000-year CA assessment period."

Response: Section 9.4.1 has been revised to clarify the base case inventory scenarios assumed for both TRU Pad 1 and FAMS and to clarify the two sensitivities conducted for TRU Pad 1 and FAMS. A new first paragraph has been added to Section 9.4.2.4.3 to clarify the base case and two sensitivity inventory scenarios conducted for TRU Pad 1. Section 9.4.2.4.1 associated with the FAMS sensitivities was considered clear enough in regard to the FAMS base case and two sensitivities and therefore no changes were made to this section.

3.3.6.6 Intermediate calculations are performed, and the results are presented to demonstrate the CA calculations are representative of the site and are consistent with results presented in the PA for similar situations.

BC-An additional strength of the GoldSim™ model is its module structure using object oriented model design. Simulation results are fully traceable to their parameter elements and intermediate stage and final results of all components of the model can be displayed and evaluated.

Response: We concur; no further response was required.

3.3.6.7 The conceptual model used for the CA is consistent with the representation of the conceptual model used in the PA, and includes the major mechanisms affecting the transport of radionuclides at the DOE site. The components of the conceptual model for the CA are reasonably represented in the analysis of the LLW disposal facility and other contributing sources.

BC-The GoldSim™ base model is divided into source, source release, vadose zone, aquifer, stream and POA model components which incorporate all the major aspects of radionuclide release, flow and transport for the SRS. The model is derived from and fully consistent with the multiple PA models developed for tank waste and radioactive waste disposal sites.

Response: We concur; no further response was required.

3.3.7 Exposure Pathways and Dose Analysis

3.3.7.1 The CA provides a complete discussion of all important exposure pathways for the evaluation of potential doses to a hypothetical, individual member of the public at the point of exposure for any time during the period of assessment. The exposure pathways identified in the CA should be consistent with the exposure pathways in the PA. The exposure pathways considered in the CA include only those pathways that are related to the exposure of individual members of the public at the point of assessment and are justified.

SG-Section 1.0, Executive Summary, Table 1-1 and paragraph above, P.1-2
Comment: Exactly how was Excel used to perform aggregation of source doses? Does this mean that Table 1-1 dose of ~3.0 mrem/yr is a combination of all the doses from each AP?
"The CA model uses the GoldSim™ code to calculate the transport from each source to its respective POA and the resulting doses at each POA. Aggregation of source doses was performed using Excel. Table 1-1 shows that the maximum cumulative dose from all residual radioactive material over the 1,000-yr AP is about three mrem/yr, indicating compliance with the 100-mrem primary dose limit, and 30-mrem dose constraint."

Response: The Executive Summary paragraph in question has been revised as follows:

The CA model uses the GoldSim™ code to calculate the transport from each source to its respective POA and the resulting doses at each POA. *Table 1-1 and Figures 1-1 and 1-2 summarize the CA results. As shown in Table 1-1, the maximum dose over the 1,000-yr AP occurs at the Lower Three Runs (LTR) POA and is about three mrem/yr, indicating compliance with the 100-mrem primary dose limit, and 30-mrem dose constraint. This maximum dose occurs at the projected site end-state date of 2025 and is due primarily to Cs137 contained within the sediment of the LTR streambed. Because Cs137 is the primary contaminant, and it has a relatively short half-life of 30 years, the LTR POA dose quickly declines from its 3 mrem/yr high in 2025 to approximately 0.1 mrem/yr in 2150, at which point LTR is no longer the controlling POA. Extending the AP to 10,000 years does not increase the maximum dose.*

The sentence that states, "Aggregation of source doses was performed using Excel", has been eliminated, because it may lead to confusion, which can not be clarified concisely, and because it is only referring to one particular aspect of the modeling and calculation mechanics, which is discussed in more detail within the CA. The Executive Summary is not the appropriate location for such a discussion.

3.3.7.2 The dose analysis performed for the CA is consistent with that performed for the PA for similar exposure pathways and similar exposure scenarios.

KM-Page 9-214, Section 9.5.1, Fig 9-127. For clarity, suggest adding another input circle to figure that points only to the "CA" circle. The new input circle should represent C-14 bioaccumulation factor and stream and river flows that were used for CA, and not PAs, as stated in Sec. 9.5.1, 4th sentence.

Response: Figure 9-127 has been revised by adding another input circle that points only to the CA.

KM-Page 9-217, Table 9-59. Should this table include a row for C-14 bioaccumulation factor and corresponding reference since this data is used for CA, but not PAs.

Response: The following row has been added to Table 9-59:

Transfer factors, dose conversion factors, exposure parameters, and consumption rates	*Phifer et al. 2009*	*Y*

The C-14 bioaccumulation factor is included within the data type "transfer factors, dose conversion factors, exposure parameters, and consumption rates", which are used by the PAs. Both the FTF and SDF PAs use a C-14 water-to-fish bioaccumulation factor (or transfer factor), and the most recent PA (i.e., SDF) uses the same C-14 water-to-fish bioaccumulation factor that the CA does. It is planned that future PAs will use the dose module developed for the CA that has been revised to include all of the PA requirements.

3.3.8 Sensitivity/Uncertainty

3.3.8.1 The sensitivity and uncertainty analysis considers factors such as alternative land use plans, CERCLA/RCRA actions or other actions (e.g., D&D, tank closures), radionuclide inventories, site and facility characteristics, and transport parameters to provide reasonable estimates of potential doses at the point of assessment for the period of the assessment. The maximum projected dose over the period of the assessment (at least 1,000 years) is presented at the point of assessment.

HP-Section 9.4.3.4, "all five POAs are shown focusing in on the 1000-year CA assessment period from 2025 to 3025". Should uncertainty analysis be performed at 10K yrs?

Response: The CA base case deterministic results were run out to 10,000 years as shown in the new Table 1-1 below (this information has been added to Section 1.0 and 10.0 due to other DOE-SR comments). As shown in the table, deterministic results over the 1,000-year AP are significantly greater than deterministic results over the subsequent 9,000 years. As shown in Figures 9-112 through 9-119 the uncertainty analysis was conducted to 4025 (i.e., a 2000-year period beyond the assumed end state date of 2025). The dose bounds are greatest during the first 1,000 years and are generally decreasing after the first 1,000 years. For these reasons uncertainty analysis to 10,000 years is not considered as important as the uncertainty analysis to 1,000 years.

However as outlined in Section 11.0, SRS recognizes the need to be able to conduct a fully probabilistic analysis to 10,000 years. Therefore SRNL is actively pursuing the ability to utilize the GoldSim™ distributed processing (DP) capability in order to expand our uncertainty analysis by assembling a large cluster of Windows machines off the SRS network for CA uncertainty work (see Table 11-2). The associated future work item in Table 11-2 has been revised to state that one purpose of the item is to be able to expand the CA uncertainty analysis to 10,000 years.

Table 1-1. Maximum Cumulative Doses at each POA during CA Period of Assessment

Point of Assessment	Maximum Cumulative [1] Dose mrem/yr 2025 to 3025	Maximum Cumulative [1, 5] Dose mrem/yr 3025 to 12050	Major Contributing Source [2]	Major Contributing Radionuclide	Major Exposure Scenario/ Pathway
Upper Three Runs	1.06	0.40	H-Canyon	Np237	Recreational/ Fish Ingestion
Fourmile Branch	2.16	0.14	FMB IOU [3]	Cs137	Recreational/ Fish Ingestion
Steel Creek/Pen Branch	0.42	0.05	SC IOU	Cs137	Recreational/ Fish Ingestion
Lower Three Runs	2.97	0.05	LTR IOU	Cs137	Recreational/ Fish Ingestion
Savannah River	0.17 [4]	0.05 [4]	LTR IOU	Cs137	Residential/ Vegetable Ingestion

[1] *Sum of doses from the residential and recreational exposure scenarios, using the respective stream flow rate for recreational dose and the Augusta, GA, river flow rate, unless otherwise noted, for residential dose.*

[2] *See Table C-1 for Source Identification corresponding to abbreviations given below.*

[3] *IOU stands for Integrator Operable Unit, which are the stream and river beds.*

[4] *Both residential and recreational doses are cumulative from all sources; the highway 301 bridge flow was used.*

[5] *In all cases, the maximum dose in the 9,000 years beyond the 1,000 year assessment period occurred in year 3025.*

BC-Alternative model results representing multiple assessments of parameter sensitivity analysis and model output uncertainty are described throughout the discussion of model development and model results in Chapters 7 and 9. The base case GoldSim™ model is easily modified and alternative simulations run to explore the effects of parameter and model uncertainty. Section 9.4 summarizes and presents the results of 12 sets of alternative scenarios selected for sensitivity and uncertainty analysis of model results. The 12 sets of scenarios represent a broad range of investigations of parameter sensitivity and model uncertainty and meet the CA review criteria for sensitivity and uncertainty analysis. Topics for consideration in future CA studies which could strengthen and expand the sensitivity and uncertainty analyses include:

1. Systematic exploration and categorization of model output for the multiple model components (source, vadose, aquifer, stream, exposure) with focused sensitivity and uncertainty analysis on the most sensitive model components.
2. More careful delineation/separation and focused exploration of parameter sensitivity and model uncertainty.
3. Better definition of source/inventory uncertainty and incorporation of that information into future sensitivity and uncertainty analysis.
4. More systematic structuring with full justification of stochastic parameterizations (hierarchy of parameters using measures of variance ranking) and incorporation into the probabilistic GoldSim™ model.
5. Application of newly evolving methods of global sensitivity using probabilistic output from the probabilistic GoldSim™ model (after completion of #4).

Response: All good suggestions that will be considered in future CA/PA work. It would certainly be valuable to determine the relative importance of the various model parameters. Applying a systematic evaluation of parameter sensitivity and model uncertainty may identify some parameters with little impact on model results that could be fixed and eliminated from the uncertainty analysis which would improve computational efficiency. We do have some concern about the amount of effort that would be required for a more extensive sensitivity and uncertainty analysis. Table 11-2 has been revised to outline potential future work that utilizes the SRNL statistical group to help design a systematic approach to model sensitivity and uncertainty analysis and to help interpret the results.

Sensitivity calculations were performed to assess the changes in inventory required to reach dose limits. These showed that the inventory estimates would have to be significantly greater than the estimated values to challenge the limits. However, variation in the inventory was not included in the uncertainty analysis. A stochastic distribution for the inventory of many SRS sources may be difficult to construct and, in many cases, will likely have to be based on expert opinion. The current Table 11-2 potential work item dealing with inventory uncertainty has been revised to focus on developing inventory distributions in a systematic manner starting with the most significant sources, and it has also been revised to establish defensible criteria to categorize whether sources require a distribution or not.

3.3.8.2 The calculated results and the sensitivity or uncertainty analysis results are used to evaluate meeting the dose constraint of 30 mrem/year and the dose limit of 100 mrem/year at the point of assessment over the period of assessment.

HP-Section 9.4.2.9 "The fifth and sixth columns in Table 9-54 indicate the 12 sources where the maximum dose during the second (approximately 10,000 year) time period exceeded the maximum dose during the CA period of assessment…" Suggest changing "maximum" to "maximum base case".

Response: Suggested change to Section 9.4.2.9 has been made.

3.3.9 ALARA & Options Analysis

3.3.9.1 For analyses that exceed the dose constraint of 30 mrem/year but are less than the dose limit of 100 mrem/year, an options analysis is provided which identifies alternatives that could be conducted to reduce the dose to less than the dose constraint. The options analysis, using the ALARA process, considers alternatives which are technically feasible and demonstrated to be effective in reducing doses to the public at the point of assessment over the period of the assessment.

HP-agree

Response: We concur; no further response was required.

3.3.9.2 For analyses that exceed the dose limit of 100 mrem/year, an options analysis, using the ALARA process, is provided which identifies alternatives that should be conducted to reduce the dose to less than the limit. The options analysis, using the ALARA process, considers alternatives which are technically feasible and demonstrated to be effective in reducing doses to the public at the point of assessment over the period of the assessment.

HP-agree

Response: We concur; no further response was required.

3.3.9.3 The ALARA process uses a cost-benefit analysis based on the cost of dose-reduction in accordance with DOE O 5400.5.

HP-agree

Response: We concur; no further response was required.

3.3.10 Results Integration

3.3.10.1 The results of the analysis for the source terms and transport of radionuclides, dose analysis, available site monitoring data, supporting field investigations, sensitivity or uncertainty analysis and options analysis are reasonable representations of the existing knowledge of the site, disposal facility, and contributing sources.

See General Strata-G Comment #6 (reproduced below).
6. KM-See section 10.2.1.4. It doesn't seem to compare the same things. For example, how many times less than 7Q10 flow are the RECREATIONAL percent of average annual flow? What about RESIDENTIAL for Lower Three Runs? Data is tabulated below as it was presented in Sec. 10.2.1.4.

River	% of Average Annual Flow for RECREATIONAL dose to reach 30 mrem/yr	How low flow must be to give a RESIDENTIAL dose at U.S. 301 Bridge of 30 mrem/yr
Savannah River	0.06%	42.5 cfs (100x less than 7Q10 flow)
Lower Three Runs	9.7%	-

Response: Section 10.2.1.4 has been revised to read as follows in order to present a more clear and concise comparison relative to the flow rate sensitivities:

Sensitivity to stream and river flow rates was assessed in multiple ways. One way that it was assessed was to determine how low the annual average flow in each of the SRS streams would have to be to cause the recreational dose at the stream mouth POA to reach 30 mrem/year. This is a pertinent statistic, because the base case dose at all of the SRS stream mouth POAs is dominated by the recreational dose. The results were 3.1%, 7.1%, 1.3%, and 9.7% of annual average flow for UTR, FMB, PB/SC, and LTR, respectively. The annual average flow would have to be reduced by at least a factor of ten before the recreational dose in these SRS streams would exceed 30 mrem/year.

Another way that sensitivity to flow rates was assessed was to determine how low the flow in the Savannah River at the US Highway 301 Bridge would have to be to cause the combined recreational and residential dose to reach 30 mrem/year. The Savannah River flow rate would have to be reduced to 56.8 cfs before the combined recreational and residential dose would reach 30 mrem/year. The Savannah River annual average flow at the US Highway 301 Bridge is 10,175 cfs. The annual average flow in the Savannah River at the US Highway 301 Bridge would have to be reduced by approximately 180 times before the combined recreational and residential dose would reach 30 mrem/year.

These results show that no credible reduction in flow will cause the CA dose to approach the 30 mrem/year dose constraint.

KM (editorial)-Page 9-3, Section 9.1.3, 6th sentence, "The conceptual model…to the POAs". Delete duplicate "one of" in that sentence.

Response: Duplicative "one of" in the 6th sentence of Section 9.1.3 has been deleted.

KM-Page 10-2, Section 10.1, Table 10-1, 5th row in table, 1st Savannah River line. Should "SR-A IOU" in this row be "LTR IOU" instead, based on text on page 10-3, last paragraph of Section 10.1, 4th sentence?

Response: The next to last paragraph of Section 10.1 has been revised to read as follows:

For the Savannah River POA both the recreational and residential doses were calculated using all sources with the annual average flow rate of the Savannah River at the US Highway 301 Bridge. The residential dose represents 68% of the total maximum cumulative dose of 0.17 mrem/year. This dose is dominated by Cs-137 from the Lower Three Runs IOU.

The 5th row (1st Savannah River line with a dose of 0.14 mrem/year in Table10-1 has been deleted, due to its being inconsistent with the cumulative nature of the remaining values within the table and in order to avoid confusion.)

3.3.10.2 The analysis, results, and conclusions of the CA provide a reasonable representation of the disposal facility and other contributing sources for determining the appropriate actions to be taken for the protection of public health and environment. The analysis and results of the CA are consistent with comparable results of the PA and provide a defensible and complete basis for an acceptable decision by DOE.

DS-The use of GIS for presentation of CA results and related information could dramatically enhance their clarity and utility. Such a project should be included in the future work plans described in Table 11-2, page 11-8. A GIS project is described in the FY 2008 E Area Annual Summary (see Section 4.4.2.4).

Response: We agree. Development of a GIS system utilizing the CA model and results has been envisioned for some time. We have added such a project to Table 11-2.

3.3.10.3 The conclusions of the CA address and incorporate any constraints resulting from other DOE programs or from any Federal, state, and local statutes or regulations or agreements that would influence the calculated results or the options analysis.

HP-agree

Response: We concur; no further response was required.

3.3.10.4 Implementation of the conclusions from the options analysis can be reasonably accomplished at the disposal facility or the other contributing sources.

HP-agree

Response: We concur; no further response was required.

3.3.11 QA

3.3.11.1 The CA discusses QA measures applied to the preparation of the analysis and its documentation. The CA includes appendices or references to published documents that provide a basis for the discussions in the CA.

HP-agree

Response: We concur; no further response was required.

G.2 GENERAL COMMENTS AND RESPONSES

General Comments from StrataG

1. DS-In general, it should not be assumed that all readers of the CA will be intimately familiar with the Savannah River Site and its jargon. All terms, locations, and landmarks should be defined to enhance the clarity and utility of the CA. Location examples include "Highway 301 Bridge" (page 3-85), "SRS Road A" (page 3-90), and "Steel Creek at Hattiesville Bridge" (page 3-82). The term "Plant North" appears in Figure 3-23 on page 3-90 but is not defined.

   Response: Figure 3-1 has been changed to indicate where Highways 125 and 278 are. The designations within the text using SRS Road numbers have been changed to refer to SRS areas, which are depicted on the various maps in the CA. Unfortunately, showing the SRS Road numbers on the maps makes the maps OUO. The following note was added to Figure 3-1 and is applicable all figures contained within this CA, which are based upon Plant North:

   Note: During design of the SRS facilities in the 1950's, an SRS site grid system was established for facility location, based upon a Plant North, which was rotated 36°22' west from True North. This connotation applies to all figures contained within this CA that are based upon Plant North.

2. DS-This CA is site-wide in scope, so it is associated with all of the performance assessments for the Savannah River Site. Nonetheless, in some instances there are references to "the PA" (e.g., page 11-3 "…as part of the PA Annual Summary" and page 11-4 "…in the PA revision…"). Such references should cite a specific PA, if appropriate, or refer to "the performance assessments" if a specific PA is not being referenced.

   Response: Sections of the CA, which have significant discussions of SRS PA work (i.e., Sections 1.0, 2.2, 3.4, 7.24, 9.37, 9.5, and 11), have been reviewed and have been corrected so that it is clear where specific PAs and PAs in general are being discussed.

3. DS-A list of documents evaluated for review comments received on the 1997 CA appears on page 5-5. Additional documents that could be evaluated and listed are the review reports on the Annual Summaries prepared for E Area and Z Area. The review reports for the Fiscal Year 2007 Annual Summary and Fiscal Year 2008 Annual Summary have been prepared by DOE Headquarters and transmitted to the Savannah River Assistant Manager for Environmental Management.

   Response: The following has been added to the end of Section 5.2.4 on page 5-5:

   Subsequently, the following documents were evaluated for review comments.

   - *Low-Level Waste Disposal Facility Federal Review Group Annual Summary Review for the Savannah River Z Area (Saltstone); and*
   - *Low-Level Waste Disposal Facility Federal Review Group Annual Summary Review for the Saltstone Disposal Facility, 9/17/2009.*

   None requiring action in the CA revision were found.

4. DS-Section 11.1, page 11-2, paragraph 3 – This paragraph describes the planned evaluation of near-surface disposal of transuranic waste against the performance requirements of 40 CFR 191. The statement that the evaluation will demonstrate that the "… material does not need the degree of isolation afforded by a geologic repository" suggests that the conclusion has already been reached and that the evaluation is being performed to backfill the facts. A more suitable statement would be, "This disposal will be assessed per 40 CFR 191 performance requirements to determine whether this material needs the degree of isolation afforded by a geologic repository."

   Response: Section 11.1, third paragraph, has been revised as requested.

5. KM-General Comment – Consider clarifying K_d and K_ds, which are used interchangeably, in the following locations in document.
   - K_d is defined in abbreviations list as "distribution (or sorption) coefficient". Consider adding "or partition" after "or sorption" to the definition.
   - Page 7-15, Section 7.2.2.5.4, 1st paragraph, 1st sentence, consider adding "(or distribution)" after "sorption".
   - Page 9-67, Section 9.3.1.6.8.3, 2nd sentence, consider adding "(or distribution)" after "partitioning".

     Response: Revisions have been made so that the first time K_d was used in a major section (i.e., 1.0, 2.0, 3.0, etc.) it was denoted as follows: "distribution (or sorption or partition) coefficient (K_d)". After the first time K_d was used in a major section it has been simply denoted as "K_d". The notation "distribution (or sorption or partition) coefficient (K_d)" has been used in the abbreviations and Sections 3.4.1.5, 7.2.2.4, 8.2.4.7.2, 9.1.1, 11.1, and 13.1; everywhere else K_d has been simply used.

6. KM-See section 10.2.1.4. It doesn't seem to compare the same things. For example, how many times less than 7Q10 flow are the RECREATIONAL percent of average annual flow? What about RESIDENTIAL for Lower Three Runs? Data is tabulated below as it was presented in Sec. 10.2.1.4.

River	**% of Average Annual Flow for RECREATIONAL dose to reach 30 mrem/yr**	**How low flow must be to give a RESIDENTIAL dose at U.S. 301 Bridge of 30 mrem/yr**
Savannah River	0.06%	42.5 cfs (100x less than 7Q10 flow)
Lower Three Runs	9.7%	-

Response: Section 10.2.1.4 has been revised to read as follows in order to present a more clear and concise comparison relative to the flow rate sensitivities:

Sensitivity to stream and river flow rates was assessed in multiple ways. One way that it was assessed was to determine how low the annual average flow in each of the SRS streams would have to be to cause the recreational dose at the stream mouth POA to reach 30 mrem/year. This is a pertinent statistic, because the base case dose at all of the SRS stream mouth POAs is dominated by the recreational dose. The results were 3.1%, 7.1%, 1.3%, and 9.7% of annual average flow for UTR, FMB, PB/SC, and LTR, respectively. The annual average flow would have to be reduced by at least a factor of ten before the recreational dose in these SRS streams would exceed 30 mrem/year.

Another way that sensitivity to flow rates was assessed was to determine how low the flow in the Savannah River at the US Highway 301 Bridge would have to be to cause the combined recreational and residential dose to reach 30 mrem/year. The Savannah River flow rate would have to be reduced to 56.8 cfs before the combined recreational and residential dose would reach 30 mrem/year. The Savannah River annual average flow at the US Highway 301 Bridge is 10,175 cfs. The annual average flow in the Savannah River at the US Highway 301 Bridge would have to be reduced by approximately 180 times before the combined recreational and residential dose would reach 30 mrem/year.

These results show that no credible reduction in flow will cause the CA dose to approach the 30 mrem/year dose constraint.

7. JP-The dose at L3R is calculated to be 2.97 mrem. It is not clear in the Executive Summary or main text but my understanding is that this dose occurs at year 2025. Suggest this point be made clear. Table 9.47 states that the L3R dose is sensitive by a factor of 10 to get to the 30 mrem threshold. The CA recognizes throughout the document that inventory uncertainty is a key issue yet no sensitivity analysis or uncertainty analysis was performed specific to the inventory (i.e., L3R IOU) that contributes to the L3R dose of 2.97. If some sensitivity or uncertainty analysis has been done maybe clarify to the reader that it has in a few appropriate places in the document. Bottom-line is DOE needs to be reassured that the inventory is not, in fact, off (i.e., greater) by more than a factor of 10.

   Response: The Executive Summary has been revised to show the maximum doses at each POA over both the 1,000-year and 10,000-year assessment periods. Figures have been provided that clearly show the 2.97 mrem/year dose at L3R occurs at 2025, the beginning of the assessment period. Future work, discussed in Section 11 of the CA, will focus on reducing uncertainty. One element of this is to develop distributions for inventories of significant sources. The initial effort will focus on the Lower Three Runs Integrator Operable Unit.

8. JP-Throughout the CA there is no explicit usage of Data Quality Objectives or Data Quality Assessment principles as provided in EPA Guidance although many parts of the document follow such principles. DQO is required by DOE M 435.1 for characterization. The evaluation of radionuclide inventory is exactly like characterizing a building or set of drums and given the importance (i.e., weighting) of inventory to the CA results my opinion is that it would be useful. DQOs follow a 7-step process that documents the decision required (e.g., what is the inventory estimate for Area XYZ) and establishes decision framework and confidence limits to achieve the decision (e.g., what values for Area XYZ inventory can we accept?). For example, scaling to Cs-137 was used for some inventory characterization (i.e., L3R IOU) but no information (at least that I could find) clarified how many samples were used to establish scaling ratios. Ratios to Cs-137 often need multiple data points (especially for heterogeneous waste forms) in order to understand scaling relationships fully. The use of the CA Core Team and various CA working groups were excellent examples of gathering input and necessary guidance throughout the process. If the LFRG has documented a decision not to require use of DQOs in PAs and/or CAs that decision should be referenced.

Response: The use of the Data Quality Objectives process in developing a Composite Analysis was originally agreed to between DOE and the Defense Nuclear Facilities Safety Board. Consequently, the FORMAT AND CONTENT GUIDE FOR U.S. DEPARTMENT OF ENERGY LOW-LEVEL WASTE DISPOSAL FACILITY PERFORMANCE ASSESSMENTS AND COMPOSITE ANALYSES, December 7, 1999, states in Section C.2.1:

> *The data quality objectives (DQO) process (Ref. 15) should be applied to the planning and implementation of the composite analysis, and use of the DQO process should be described in this section. The DQO process is used to specifically identify the data and quality of data needed to make decisions with acceptable levels of uncertainty.*

This was reflected in the LFRG Manual, issued in 1999, LOW-LEVEL WASTE DISPOSAL FACILITY FEDERAL REVIEW GROUP MANUAL, Revision 1, November 1999, in CA review criterion 1: "The CA includes a discussion of how the Data Quality Objectives (DQO) process was used as a flexible planning tool and applied to the CA preparation."

However, in 2006, the LFRG revised their manual and removed the criteria calling for the use of the DQO process.

Thus, the DQO process was not used in the preparation of the revision of the SRS CA.

It should be noted however, that the SRS Area Closure organization does use the DQO in acquiring environmental data for closure actions. These closure actions include many of the sources of residual radioactive material assessed in the SRS CA revision.

No change to the CA was required.

9. JP-Clarify (expand) Executive Summary discussion when the 2.97 mrem dose occurs (i.e., it happens in 2025 – even though text on Page 4-5 states land use will be controlled for at least 50 years). Also, since Cs-137 is the dose driver additional discussion that states how it will decrease proportionally will decay every 30 years or so is relevant to reassuring compliance and the importance of land-use controls. Table 9-27 stated the dose was 2.94 not 2.97 (perhaps a typo?)

   Response: The 9th paragraph of the executive summary has been revised as follows to address the first part of the comment:

   > *The CA model uses the GoldSim™ code to calculate the transport from each source to its respective POA and the resulting doses at each POA. Table 1-1 and Figure 1-1and Figure 1-2 summarize the CA results. As shown in Table 1-1, the maximum dose over the 1,000-yr AP occurs at the Lower Three Runs (LTR) POA and is about three mrem/yr, indicating compliance with the 100-mrem primary dose limit, and 30-mrem dose constraint. This maximum dose occurs at the projected site end-state date of 2025 and is due primarily to Cs137 contained within the sediment of the LTR streambed. Because Cs137 is the primary contaminant and it has a relatively short half-life of 30 years, the LTR POA dose quickly declines from its 3 mrem/yr high in 2025 to approximately 0.1 mrem/yr in 2150, at which point LTR is no longer the controlling POA. Extending the AP to 10,000 years does not increase the maximum dose.*

   Table 9-27 provides the maximum LTR POA dose from each LTR POA source. The 2.94 mrem/year maximum LTR POA dose presented in Table 9-27 is not a total maximum LTR POA dose, but it is the maximum dose to the LTR POA from a single source (i.e., the LTR IOU or LTR streambed). The total maximum LTR POA dose, including the LTR streambed, is 2.97 mrem/year. Therefore Table 9-27 is correct in what it presents and no change to the CA is required in this regard.

10. JP-Page 3-2 says the M-Area Settling Basin cleanup began under RCRA in 1981. Suggest checking that date as DOE in general did not acknowledge the applicability of RCRA to DOE activities until after the 1984 Federal court decision Hodel vs. EPA. Further, Page 3-5 seems to indicate the M-Area Settling Basin operated through 1985.

    Response: DOE-SR and DuPont began working with the South Carolina Department of Health and Environmental Control (SCDHEC), as a matter of comity, on SRS Resource Conservation and Recovery Act (RCRA) issues in 1981. However DOE-SR did not officially recognize SCDHEC's RCRA regulatory authority over SRS until after the 1984 Federal court decision Hodel vs. EPA. Because this might be a little confusing, the Section 3.1, 5th paragraph, 2nd sentence has been revised as follows:

    > *By 1980, a five-year Restoration Program for process facilities on the site was launched. In 1981, chlorocarbon groundwater contamination was discovered at the M-Area Settling Basin, and an M-Area groundwater cleanup program was initiated that resulted in operation of a full-scale groundwater treatment system by 1985 (Reed et al. 2002).*

11. JP-Page 3-10 discusses the final closure cap on ORWBG in 2007. What kind of influence did the cap have on fate and transport assumptions or if no credit was afforded explain why? This may be discussed elsewhere but the answer would be useful to note at this point or let the reader know where to find the answer.

    Response: Section 3.0, Background Information, provides general SRS information such as historic operations, hydrogeology, Performance Assessments, related documentation, etc. As such Section 3.0 simply lays the groundwork and background necessary to understand the context within which the SRS CA was prepared. As such it is not appropriate to discuss details of the actual SRS CA analysis within Section 3.0. This would simply lead to confusion, because such a detail would not be able to be put in context of the overall SRS CA analysis within Section 3.0 and such analysis details would then need to be provided for other facilities within Section 3.0. How the ORWBG closure cap was considered within the SRS CA analysis is discussed in Sections 9.3.1.6.2, Table 9-22, and 9.3.7.1.

    No change to the CA was required.

12. JP-Page 6-8 (bottom) states: "…with the use of existing documented inventories considered most certain and inventories developed from SARs considered least certain." This approach does not make sense to me as *existing documented inventories* could mean many things including propagating erroneous PK information. SARs are now managed under 10 CFR 835 PAAA rules and thus I would argue their pedigree would be amongst the best data to use although I have known SAR data to be overly conservative (i.e., reporting to maximums instead of the mean or some other values).

    Response: The Safety Analysis Reports develop information on the hazards of a facility during its operation. Thus, the SAR inventory information represents the inventory of radioactive material in the facility while operational. The CA is an analysis of the post-operational impacts. The SAR inventories were used by assuming that a fraction of the operational inventory would be retained in the closed facility. The assumed fraction is the uncertain quantity.

    A footnote to this effect has been added.

13. JP-Figures 7-1, 7-2, and 7-3 are excellent. Lots of information but easy to follow logic.

    Response: We concur; no further response was required.

14. JP-Table 7-3 "Preliminary CA Source Data" is interesting. I think the purpose is to just show everything is less than 10 M curies assumed for screening but is the data in Table 7-3 used again? What is the goodness of the data? Very large curie values are purported with intuitively a lot of variability in each number. Suggest some footnotes or additional text to better understand how this table is used in this section and/or later in the document if relevant.

Response: The purpose of Table 7-3 is simply to show that the assumed 10 MCi bounds all of the CA sources. The data in Table 7-3 is not used anywhere else in the CA. The following has been added, "These inventories represent the sum of the inventories of all radionuclides in each source", after the sixth sentence of the first paragraph in Section 7.2.2.5.5, and the following has been added, "The data in Table 7-3 is not used further in the CA", at the end of that paragraph.

15. JP-Page 9-2 states "Inventory for each individual source" were evaluated for sensitivity but I could find only that specific inventories (those covered in other PAs) were evaluated and not each individual source (e.g., L3R OIU).

    Response: This sensitivity was performed for all 152 sources evaluated in this CA. However those 135 sources with an inventory multiplier greater than 1000 were simply denoted as such and were not specifically listed. Changes to Section 9.1.1 and 9.4.2.4.2 have been made to clarify this sensitivity.

    Section 9.1.1, 2nd paragraph, 3rd burger dot has been revised to read as following:

    - Source inventory (inventory required to reach 30 mrem/yr)
      Section 9.4.2.4.2, 1st paragraph has been revised to read as follows:
      This sensitivity study determined the overall inventory multiplier that would be required to reach a dose of 30 mrem/yr for individual sources contributing to each POA while keeping all of the other sources for that POA at the base case inventory. A specific inventory multiplier was determined for all PA sources (i.e., ELLWF facilities, FTF, HTF, and SDF) and all other sources with an inventory multiplier of 1000 or less. Those sources with an inventory multiplier greater than 1000 are simply denoted as such and include all sources other than those listed in Table 9-44 and Table 9-45. The results for PA sources and those sources with an inventory multiplier of 1000 or less are given in Table 9-44 and Table 9-45 as a number that the best estimate source inventory would have to be multiplied by to give a total dose at the POA of 30 mrem/yr. Of the 152 sources evaluated in this CA, 135 had inventory multipliers greater than 1,000; 14 had inventory multipliers between 100 and 1000; and only the three sources highlighted in Table 9-44 and Table 9-45 (H-Canyon, FMB IOU and LTR IOU) had inventory multipliers less than 100.

16. JP-Page 9-2: It would be useful and informative to assess inventories probabilistically in addition to the 6 items (bullets) studied.

    Response: Future work, discussed in Section 11 of the CA, will focus on reducing uncertainty. One element of this is to develop distributions for inventories of significant sources. The initial effort will focus on the Lower Three Runs Integrator Operable Unit. Table 11-2 has been revised to show emphasis on developing inventory distributions for the most significant sources.

General Comments from Bruce Crowe

1. BC-Identified weaknesses in the report do not invalidate the general report conclusions and are categorized as either secondary issues or as areas for future improvements. They include:
   a. Incomplete description of the regional groundwater flow system and the basis for establishing boundary conditions and fluxes for the multiple groundwater flow models used to justify the GoldSim™ deterministic base case. This is mostly a weakness of the foundation PA and site studies and not necessarily the GoldSim™ model used for the CA.

      Response: Section 3.2.10 provides an overview of SRS hydrogeology. Further details concerning SRS hydrogeology can be found in the many references cited within Section 3.2.10. Sections 9.2 and 9.2.1 discuss the four SRS groundwater models, which have been used to establish flow path parameters from each radionuclide source location to the respective surface water discharge. The four SRS models are discussed in some detail within Section 9.2.1 (see Tables 9-3 and 9-4). Further details, including boundary conditions, for each of the models are discussed within the associated model reports (see Table 9-3 for flow model references). The boundary controls for three of the SRS groundwater models (CKLP, GSA, and R (see Table 9-3)), used to establish flow path parameters for the CA, consist of a combination recharge/drain top surface boundary; a prescribed head with leakance coefficient across the Crouch Branch Confining Unit bottom boundary (see Figure 3-33); and no flow or prescribed head vertical boundaries based upon stream boundary control. The boundary controls for the AM model (see Table 9-3), also used to establish flow path parameters for the CA, consist of a combination recharge/drain top surface boundary; a no flow McQueen Branch bottom boundary (see Figure 3-33); and specified head or head-dependent flux vertical boundaries based upon stream boundary control.

      Section 11.0 of the CA has maintenance plan items associated with production of an SRS regional model encompassing the entire SRS. Within Table 11-1 building an SRS-wide hydrologic model is listed as a potential ARRA funded CA maintenance item. This item includes the following as necessary as outlined within Table 11-1: synthesize existing hydrologic data; identify data gaps; build database; collect field data; create field sample archive; determine surface impacts on infiltration and water table elevation; and build model and advanced visualization tools. Development of such an SRS regional model will necessarily result in a more complete description of the regional groundwater flow system in and around SRS. Additionally Section 11 has been reworded to make it clear that an SRS regional model encompassing the entire SRS should be developed from which boundary controls for smaller SRS groundwater models with greater grid resolution can be extracted.

b. Incomplete assessment of transient groundwater conditions and cyclic changes in the groundwater water levels under longer-term drought and wetter conditions.

Response: As outlined in 9.4.3.2, aquifer sand and clay velocity probability density functions were developed by performing the aquifer flow parameters analysis (Hamm et al. 2009) with three GSA Flow Model solutions at high, low and nominal water table levels. These functions were applied in the uncertainty analysis to address the overall uncertainty associated with travel time through the aquifer (i.e., both uncertainties in pore velocity as well as travel length) for each material type. This in effect did consider cyclic changes in the groundwater levels under longer-term drought and wetter conditions.

Section 11.0 of the CA has maintenance plan items associated with production of an SRS regional model encompassing the entire SRS. Within Table 11-1 building an SRS-wide hydrologic model is listed as a potential ARRA funded CA maintenance item. Within Table 11-2 including low, average, and high water table conditions within the SRS-wide hydrologic model, so that distributions about the aquifer flow path parameters can be developed, is listed as a potential future work item. This should more fully address cyclic changes in the groundwater levels under longer-term drought and wetter conditions than was done in the current CA revision.

No changes were required to the CA.

c. Complicated and varying analysis of time of compliance (exactly what compliance intervals are required by 435.1 and how should they be compared with the performance measures of the CA?). All of the required analyses are presented but it is not clear what performance measures are applicable for 10,000 years and for time to peak dose. This is a continuing problem for all PAs and CAs and requires consideration in revisions of 435.1.

Response: Comment noted.

We agree that the situation in this CA is a bit confusing. The DOE 435.1 required period of assessment is 1,000 years. However, because of the high-level waste facilities at SRS and their review by the NRC, we felt it necessary to include the results of assessment at 10,000 years, which is the NRC Headquarters staff recommended period of assessment.

Text in the Executive Summary and in Section 4.3 has been added to make it clear that the 1,000-year assessment period is mandated by DOE 435.1-1, the analysis to peak dose is recommended in the PA/CA Standard Format and Content Guide, and the analysis to 10,000 years is done to provide potentially helpful information for those facilities regulated by Section 3116 of the 2005 National Defense Authorization Act.

d. The sensitivity and uncertainty analysis (SA) meets the requirements of the CA. However, the overall approach for SA could be better tied to the conceptual model(s) of flow and transport to ensure all key components of the system are fully evaluated. Uncertainty in inventory estimates for the multiple interacting sources should be significant and this issue is not fully considered in the CA (it is probably not required for a CA). A systematic assessment of inventory uncertainty should be a priority topic for future work, given the number and complexity of the inventory sources at SRS. This work would fit well into the SRS GoldSim™ model effort and could be merged with the probabilistic GoldSim™ work. The probabilistic GoldSim™ modeling studies are not yet well structured and global sensitivity analysis is not conducted.

Response: All good suggestions that will be considered in future CA/PA work. It would certainly be valuable to determine the relative importance of the various model parameters. Applying a systematic evaluation of parameter sensitivity and model uncertainty may identify some parameters with little impact on model results that could be fixed and eliminated from the uncertainty analysis which would improve computational efficiency. We do have some concern about the amount of effort that would be required for a more extensive sensitivity and uncertainty analysis. Table 11-2 has been revised to outline potential future work that utilizes the SRNL statistical group to help design a systematic approach to model sensitivity and uncertainty analysis and to help interpret the results.

Sensitivity calculations were performed to assess the changes in inventory required to reach dose limits. These showed that the inventory estimates would have to be significantly greater than the estimated values to challenge the limits. However, variation in the inventory was not included in the uncertainty analysis. A stochastic distribution for the inventory of many SRS sources may be difficult to construct and, in many cases, will likely have to be based on expert opinion. The current Table 11-2 potential work item dealing with inventory uncertainty has been revised to focus on developing inventory distributions in a systematic manner starting with the most significant sources, and it has also been revised to establish defensible criteria to categorize whether sources require a distribution or not.

2. BC-Section 3.28 provides full descriptions of the numerous infiltration and water balance studies conducted at SRS over a 30-year interval. What is missing in the summary section 3.2.8.2 is an integration of the results of the studies. There is integration of the results of infiltration data in the modeling section but it would be useful in this section to clearly state the conclusions through the progression of studies. It would also be useful to critique and constrain the results of the multiple decades of study and point toward preferred conclusions particularly for the critical infiltration ranges.

Response: The following concluding paragraph to Section 3.2.8.2 has been added to put the infiltration and water balance studies in perspective relative to background infiltration:

Based upon the results of these background water balance and infiltration studies, the nominal annual average background infiltration at SRS has been historically taken as 15 inches/year for modeling purposes (Shine 2008). Additionally the 1000-year average background infiltration rate standard deviation has been determined to be 0.17 inches/year, based upon an analysis of this historical data and a 100-year simulation history of background infiltration (Phifer et al. 2007; Shine 2008).

3. BC-I found section 3.2.10 to be generally useful for an overview perspective of the groundwater hydrology. The general conceptual model of heads controlled by creek levels with bounding creeks forming groundwater islands seems generally sound. What seems missing is a general discussion of the relative importance of groundwater underflow versus local recharge. Does SRS have a regional scale model which provides information to balance water flows for the multiple groundwater islands? The USGS MODFLOW model described by Clark and West could serve that purpose. Does SRS use the USGS model or adapt their own model? Description of particle track results in later sections appears to rely on the USGS model or the more recent GeoTrans model. Doesn't SRS maintain and assess their own regional scale model for particle track studies? How sensitive is the regional groundwater flow to transient conditions including both seasonal changes and longer term changes associated with drought and cycles of wetter conditions? Does local recharge dominant creek flows or is regional groundwater flow locally important? Do fluctuations in water table levels change the effects of confining units and the role of aquifer units on the SRS site? These topics should be covered in this section and should lead to an overall description of uncertainty in the conceptual model of the flow system. I found the descriptions of the hydrology of individual sites in the final sections 3.2.10 to be very good but am left wondering how affected the individual sites are by regional flow and transient conditions?

   Response: The USGS developed their regional model to address state of Georgia concerns with the potential for migration of tritium contaminated groundwater from SRS into Georgia (USGS 2006). The USGS used their regional model to assess the potential for trans-river flow of contaminants from the SRS into the underlying aquifers and beneath the Savannah River into Georgia. As such the USGS regional model was concerned primarily with migration of tritium to deeper aquifers that might have the potential to underflow the Savannah River. The USGS model was less concerned with flow in upper aquifers, which discharge to on-site streams. However the bulk of transport from SRS radionuclide sources to the discharge points occurs in the upper aquifers and discharge to on-site streams. Therefore the CA utilized four existing SRS groundwater models to establish flow path parameters for the CA, which were designed to evaluate contaminant migration from SRS source locations to their predominant discharge locations. While these SRS groundwater models are not regional models, they were constructed in context of the overall regional hydrogeology in particular in regard to assigned boundary conditions. The individual model references can be consulted for more complete model information including hydrogeological context and boundary conditions than provided in Section 9.2.1 of the CA.

Currently an SRS regional model encompassing the entire SRS does not exist. However as previously stated by the reviewer the lack of linkage between current SRS groundwater models and a regional model is not a significant concern, given the hydrogeologic data supporting the boundary controls of the SRS groundwater models. However SRS does recognize the benefits of an SRS regional model to help answer questions in regard to the impacts of transient drought and wet conditions on groundwater flow patterns and associated discharge locations. As outlined in Section 11.0 of the CA has maintenance plan items associated with production of a SRS regional model encompassing the entire SRS.

Within Table 11-1, building an SRS-wide hydrologic model is listed as a potential ARRA funded CA maintenance item. Development of such an SRS regional model will necessarily result in a more complete description of the regional groundwater flow system in and around SRS. Within Table 11-2 it is stated that as a potential future work item the SRS regional model should include low, average, and high water table conditions, so that distributions about the aquifer flow path parameters can be developed. These Section 11 items have been reworded to make it clear that an SRS regional model encompassing the entire SRS should be developed from which: 1) boundary controls for smaller SRS groundwater models with greater grid resolution can be extracted; and 2) the impacts of transient drought and wet conditions on contaminant transport can be evaluated.

4. BC-Figure 3-84 would be more convincing with particle tracks. Arrows from the facilities on Figure 3-96 showing the expected drainages would help the discussion of the figure.

   Response: Section 3.0, Background Information, including Figures 3-84 and 3-96, provides general SRS hydrogeology information, which was in existence when work on the revised SRS CA began, to simply lay the groundwork and background necessary to understand the context within which the SRS CA was prepared. Particle tracking from each SRS CA source was performed as part of the SRS CA and can be found in Appendix D, Figures D-1 through D-32. It is not considered appropriate to discuss actual work performed for the CA in Section 3.0, because Section 3.0 was designed to provide background information available at the time that preparation of the revised SRS CA began.

   No change to the CA was required.

5. BC-Chapter 7 on screening is reasonable and I agree with the logic and use of modeling to screen the pathways and radionuclides. I looked through but did not do simulation runs for the sections of the GoldSim™ model used for the transport screening. I agree with the use of NCRP (1996) for radionuclide screening and secondary screening based on production processes. My only comment is that it would probably make more sense to screen first on production processes (why include a radionuclide if it was not produced?) then apply the NCRP screening.

Response: It was actually easier to use the NCRP screening first because that eliminated a very large number of radionuclides. Had we started with a review of production processes, we would have considered a number of radionuclides that are not listed in the NCRP set of radionuclides. It would have been much more cumbersome and we would have arrived, in the end, at the same list of radionuclides of importance to the CA.

No change to the CA was required.

6. BC-Chapter 8 is a long chapter and I only skimmed the content and gained an appreciation for the amount of work it took to integrate the information. You have your work cut out for tracking the inventory estimates through site closure! The IOUs are somewhat complex units and I noted that you referenced a 2001 report where the soil and sediment samples were collected. If I had more review time, I would have liked to examine the ERD (2001) to gain an understanding of how representative the sampling is for the range of IOUs. What would have been very useful in the inventory tables of Appendix A would be estimates of inventory uncertainty. This information would be very useful in your GoldSim™ models and it is something to consider for future work.

   Response: Future work, discussed in Section 11 of the CA, will focus on reducing uncertainty. One element of this is to develop distributions for inventories of significant sources. The initial effort will focus on the Lower Three Runs Integrator Operable Unit. Table 11-2 has been revised to show emphasis on developing inventory distributions for the most significant sources.

7. BC-Table 9-1 is a nice overview summary of the base case conceptual model and made it easier to review the document. I have to make a comment on the use of "conservative values" in the modeling section and get this out of my system. I'm generally opposed to conservatism in model parameters for a variety of reasons (nonlinearity, nonsystematic propagation of conservatism through multiple stages of CA calculations, unclear definitions of "acceptable" conservative values) but recognize their use in screening models and a CA calls for conservative assessments of interacting sources. The PA community in general is attempting to better quantify uncertainty (mostly through probabilistic analysis) and why not be more descriptive of what constitutes a conservative parameter? I strongly urge consideration of attempting to define exactly what is chosen as a conservative value by contrasting the chosen value with mean or expected values and perhaps defining where a conservative value is visualized to exist in a parameter distribution (i.e., > mean or median and possibly compared with percentile or quantile information if possible). That way the reader can make a decision whether an assumed value is acceptably conservative rather than read the ubiquitous and completely uninformative statement of "we assigned a conservative value."

Response: We agree that the assignment of a value as "conservative" is a value judgment. And, we agree that such judgments should be made transparent to the extent practical by describing the information used to make the judgment. However, often it is not practical or possible to lay out all the factors taken into account in the judgment.

The following has been added as a footnote on page 9-1 of the document where the last paragraph states, in part "…as a reasonably-conservative best estimate…". This is the first occurrence of the term "conservative" in the modeling section.

> *[1] The term "conservative" is used to describe a situation where insufficient data exists to properly quantify a parameter or to define a sub-set of a model. Assigning a "conservative" selection is a value judgment that is often made on the basis of expert opinion. In such situations it is not possible to lay out all the factors taken into account in the judgment.*

8. BC-Table 9-2 is the first reference I saw to a regional groundwater flow model other than the USGS model. What is the model and why isn't it described in the introductory sections? I assume, based on the reference that this is an A/M Area flow and transport model and not a regional groundwater flow model.

   Response: The AM model is a flow and transport model and not a regional flow model. The AM model was developed primarily as an aid to understand the migration of the current trichloroethylene (TCE) plumes in this location and to provide a means for assessing the effectiveness of remedial and management alternatives for the plumes. Therefore the Table 9-2 AM model name has been revised to eliminate the word, "Regional".

9. BC-Page 508. The GeoTrans (2009) model is referenced for the CA particle track analyses whereas a previous section of the report referenced the USGS model for particle tracking.

   Response: Background Section 3.2.10.1.9 discussed USGS regional modeling (MODFLOW–2000) that was conducted to address state of Georgia concerns with the potential for migration of tritium contaminated groundwater from SRS into Georgia (USGS 2006). The USGS used their regional model to assess the potential for trans-river flow of contaminants from the SRS into the underlying aquifers and beneath the Savannah River into Georgia.

The USGS model indicated that all locations of trans-river flow are restricted to the Savannah River's floodplain, where groundwater passes immediately prior to discharging into the river, even under conditions of enhanced groundwater withdrawals. The investigation confirmed that the Savannah River is a regional discharge zone, with groundwater proceeding towards the river from both the Georgia and South Carolina sides of the river. This pattern is observed within all aquifer zones. Where trans-river flow does occur, it is localized beneath the Savannah River floodplain and is primarily controlled by the position of meanders of the Savannah River within the floodplain. Virtually all of the westward trans-river flow originates as surface infiltration well to the east and south of the SRS and passes underneath the SRS before ultimately discharging upward into the Savannah River. Additionally, the particle tracking work confirmed that virtually all surface infiltration within the SRS discharges locally to site streams. The only exception was to identify one small area of less than 1 square mile, located immediately adjacent to the Savannah River on the Georgia side of the river, which received groundwater that originated within the SRS. Recharge linked to this area is from 6 separate model cells that total about 1 square mile within SRS and none of which are at contaminated locations.

WSRC-TR-2007-00008, Savannah River Site Environmental Reports for 2006
WSRC-TR-2006-00007, Savannah River Site Environmental Reports for 2005

CA Modeling Sections 9.2 and 9.2.1 discuss the SRS models, which have been used to establish flow path parameters from each radionuclide source location to the respective surface water discharge. The four SRS models are discussed in some detail within Section 9.2.1 (see Tables 9-3 and 9-4). These SRS models were used rather than the USGS model for several reasons: 1) availability; 2) the USGS model was designed primarily to evaluate the potential for trans-river flow (i.e., deep aquifers which discharge directly to the Savannah River); 3) mesh size versus flow path distance (USGS model used a minimum grid size of 0.33 mile by 0.33 mile (1742 feet by 1742 feet), while the CA flow path distance ranged from 440 to 50,000 feet with an average of 6,000 feet); 3) consistency with PAs; etc.

No change to the CA was required.

10. BC-My initial reaction looking at Figure 9-14 and inside the GoldSim™ model was why use cells for the saturated zone when GoldSim™ pipes were designed specifically for saturated flow and provide features that require more work to implement in cells? After reading through the all of Chapter 9, I assume that cells were used to provide the detail required by the multiple links to the saturated zone flow path. Were pipes considered and not used for a particular reason?

Response: As noted, cells were used to model the aquifer because flow from the vadose zone to the first 100 aquifer cells was distributed in a non-uniform manner based on the source footprint in relation to the average aquifer flow path. The model could use all or some part of the first 100 cells to define this flow distribution. The second group of 100 aquifer cells was subdivided into a region of 40 cells that could be used to define an equivalent aquitard region and a region of 60 cells that were assumed to be saturated sandy soil. Region and cell dimensions varied with source dimensions and aquifer path lengths. We could have replaced the last 100 cell segment with GoldSim™ pipes but elected to use cells throughout for consistency in the aquifer model.

No change to the CA was required.

11. BC-Figure 9-33. Some odd outlier tritium fluxes in the Sept-2005 data. Any explanation?

    Response: Looking back into the Environmental Protection Department records it was found that a phytoremediation pond, located in the General Separations Area (GSA), was drained for dam maintenance and upgrades in August, 2005. Because the pond is located on a drainage way that discharges into Four Mile Branch (FMB), an increase in tritium concentration in samples collected from FMB at downstream Station A7 was both expected and observed. The work on the dam was completed in the Spring of 2006, at which time a decrease in tritium concentration in samples collected at Station A7 was also expected and subsequently observed. Therefore, the construction work performed to drain the pond explains the observed flux differences and corresponds with the time period during which elevated fluxes to FMB are observed on the trend plot.

    The following sentence has been inserted at the end of the second paragraph of Section 9.3.1.6.7.2 immediately before Figure 9-33:

    > *With the exception of a handful of elevated tritium flux data points that occur from late 2005 and into early 2006, and which are linked to a specific construction-remediation project in the GSA, the data points on this graph conform relatively closely to the indicated trend line.*

12. BC-I had a listing in my notes about a statement in the report that some radionuclide species are in the CA which are not in the PA which seems odd. However, when writing up my review I couldn't find the page for this statement. As an interesting observation, I found that Word on different computers doesn't consistently assign the page numbers so I couldn't find the page during my review write-up.

    Response: It is believed that the reviewer is referring to the following statement within Section 9.3.1.6.6.1:

"Some of the CA species were not PA species of interest and therefore, no PA flux data exist. Inventories for those species not included in the PA flux, but among those analyzed in the CA, must be included in the input for the pertinent source."

The Saltstone Disposal Facility (SDF) PA species corresponded exactly with the CA species. However, for both the E-Area ILV and E-Area LAWV the following 11 CA radionuclides had inventories but did not appear in the PA flux tables: Am-242m, Cf-249, Cm-246, Cs-135, Cs-137, K-40, Ni-63, Pb-210, Pu-240, Sr-90 and Th-230. In these cases, the inventory not accounted for in the PA fluxes was included using the CA generic release mechanism. These sentences in Section 9.3.1.6.6.1 have been revised to be more specific regarding differences between PA and CA species in the use of PA fluxes.

13. BC-Page 9-63. I'm not sure I agree with uniformly introducing tritium along all 200 mixing cells. Why not take advantage of any available information you have from monitoring wells on the spatial variability of tritium in groundwater plumes?

    Response: In general, it was felt that tritium plume data was not adequate to define plume concentration profiles. The plume maps typically do not show much definition in terms of delineating concentrations within plumes. For example, the L-Area plume maps only delineate the regions thought to have concentrations > 20,000 pCi/L. In most cases, the plumes extend from the source to the stream discharges. For several of the areas which contained multiple sources of tritium (usually from basins, or reactor buildings, or both), the total inventory was placed into the single shortest groundwater path. Examples would include P-Area, L-Area and K-Area. This was viewed as a conservative way to handle groundwater plume inventories. Section 9.3.1.6.7.3 has been revised to better explain the rationale for this approximation.

14. BC-My preference for Section 9.3.5 would be to separate the model testing into verification activities (testing whether the code functions as expected) and validation recognizing that the ability to validate any transport model over time is very limited.

    Response: SRNL QA for computer codes typically defines verification as the sort of activity performed in our design checking process (i.e., having an independent reviewer check that a specific part of the coding is correct and functioning as expected). Design checking of the GoldSim™ CA model was performed and documented. We typically define validation as comparison of code results to data and/or results from other codes. The opening paragraph in Section 9.3.5 has been modified to clarify the complete CA model testing that was performed.

15. BC-Clearly the difference between the generic source release results and the detailed source release results are very significant and this should be a featured observation in the sensitivity and uncertainty analysis section. The generic source release approach certainly overestimates resulting doses.

Response: Moving Table 9-23 and the related discussion from Section 9.3.7 to Section 9.4.2.2 was considered. However considering the table renumbering and revision of associated references required in Section 9.3, this approach was rejected in favor of modifying the discussion in Section 9.4.2.2 to include the points noted by the reviewer. Such a modification to Section 9.4.2.2 has been made.

16. BC-As an overview comment, I think section 9.4 is adequate for the CA requirements/criteria. However, I was expecting something slightly different and have to add that much of the sensitivity approaches are scattered throughout the modeling sections (much of the sensitivity studies were conducted as part of the modeling work and not necessarily as a separate sensitivity analysis activity). What I was expecting was three things:
    a. A discussion of what parts of the conceptual model most control the dose results using a categorization by source, vadose, aquifer, streambed and dose components. Based on the generic versus source term differences, I would guess the source component of the model dominates the results. I also think it is useful to separate out the dose module and look at system sensitivity independent of the dose model since the rest of the model is process based whereas the dose model is controlled by exposure parameters.
    b. For each component of the conceptual model and for the total system, there needs to be an assessment of the importance ranking of parameters (true sensitivity analysis).
    c. An assessment of the uncertainty affecting each component of the conceptual model and identification of scenarios affecting those components of uncertainty (true uncertainty analysis).

    Response: We have perhaps used the terms "sensitivity analysis" and "uncertainty analysis" in the CA report in a manner that does not conform exactly to current statistical terminology. Our working definition of sensitivity analysis was testing the model response to discrete variations in model parameters where a distribution of possible values either can not be defined or does not exist. Similarly, we defined uncertainty analysis as testing the model response to random variations in parameters where a mean value and stochastic distribution are defined. These may not be rigorous definitions but served to categorize the two distinct types of evaluations that were performed. We agree that a more systematic sensitivity analysis, along the lines outlined in the other review comments, would be useful. The first paragraph in Section 9.4 has been revised to include a brief description of our approach to sensitivity and uncertainty analysis. Additionally Table 11-2 has been revised to outline potential future work that utilizes the SRNL statistical group to help design a systematic approach to model sensitivity and uncertainty analysis and to help interpret the results. The suggestion to investigate model sensitivity to the dose model calculations separately from that of the transport model is well taken. A Section 11.2, Table 11-2 potential future work item has been added to investigate model sensitivity to the dose model calculations separately from that of the transport model.

17. BC-Looking at the 12 scenarios selected for the sensitivity analysis, I think you have probably covered all of the important uncertainty components but I would have more confidence in this conclusion if you followed a more systematic approach to the overall structure of the sensitivity analysis. As an additional point, I would also call the scenario evaluations an assessment of uncertainty (not sensitivity) as I define uncertainty as variance in model output with changes in model input and sensitivity as importance ranking of model parameters.

    Response: Our working definition of sensitivity analysis was testing the model response to discrete variations in model parameters where a distribution of possible values either can not be defined or does not exist. Similarly, we defined uncertainty analysis as testing the model response to random variations in parameters where a mean value and stochastic distribution are defined. These may not be rigorous definitions but served to categorize the two distinct types of evaluations that were performed. We agree that a more systematic sensitivity analysis, along the lines outlined in the other review comments, would be useful. The first paragraph in Section 9.4 has been revised to include a brief description of our approach to sensitivity and uncertainty analysis. Additionally Table 11-2 has been revised to outline potential future work that utilizes the SRNL statistical group to help design a systematic approach to model sensitivity and uncertainty analysis and to help interpret the results.

18. BC-I had three issues with the sensitivity section. The first is that sensitivity specific model runs were not completed for the SA. I found it difficult to shift back and forth between the SA section and the descriptions of the model runs to assess the results. Second, from looking at many PAs and CAs across the complex, inventory uncertainty almost always dominates the SA but the CA calculations treat site specific inventories as fixed values and do not attempt to assess the uncertainty in the inventory estimates. Again, the differences in dose results for the generic versus source modeling approaches strongly suggests that inventory and inventory release approaches dominant the calculated doses. Third, assessment of the impact of the groundwater divide is a very useful section. I would like to see additional calculations of the impact of transient conditions particularly long-term changes in water table elevations.

    Response: See responses below:

    1. *Using the linear response of dose to changes in flow and inventory allowed us to evaluate many of the "sensitivity" cases without additional model runs. References back to the base case runs on which the sensitivity analyses were based may be difficult to follow. Having multiple authors contributing to this section of the report also led to some differences in style and structure in the final report. The description of the sensitivity analyses has been revised to better connect the results with the corresponding base case calculations.*

2. *While we agree with the comment that inventory uncertainty can dominate dose calculations, the nominal inventory used in the model calculations represents a best estimate value and distributions about this value are not currently available. However sensitivity calculations were performed to assess the changes in inventory required to reach dose limits. These showed that the inventory estimates would have to be significantly greater than the estimated values to challenge the limits. However, variation in the inventory was not included in the uncertainty analysis. A stochastic distribution for the inventory of many SRS sources may be difficult to construct and, in many cases, will likely have to be based on expert opinion. The current Table 11-2 potential work item dealing with inventory uncertainty has been revised to focus on developing inventory distributions in a systematic manner starting with the most significant sources, and it has also been revised to establish defensible criteria to categorize whether sources require a distribution or not.*

3. *As outlined in 9.4.3.2, aquifer sand and clay velocity probability density functions were developed by performing the aquifer flow parameters analysis (Hamm et al. 2009) with three GSA Flow Model solutions at high, low and nominal water table levels. These functions were applied in the uncertainty analysis to address the overall uncertainty associated with travel time through the aquifer (i.e., both uncertainties in pore velocity as well as travel length) for each material type. This was a consideration of long-term changes in water table elevations. Section 11.0 of the CA has maintenance plan items associated with production of an SRS regional model encompassing the entire SRS. Within Table 11-1 building an SRS-wide hydrologic model is listed as a potential ARRA funded CA maintenance item. Within Table 11-2, including low, average, and high water table conditions within the SRS-wide hydrologic model, so that distributions about the aquifer flow path parameters can be developed, is listed as a potential future work item. This should more fully address long-term changes in water table elevations than was done in the current CA revision. No change to the CA in relation to this point was required.*

19. BC-Finally, I have three concerns with the probabilistic section. First, it is very difficult to choose and eliminate parameters for assigning PDFs without following an iterative process of identifying the set of parameters which most control system variance. This generally requires multiple model passes as the parameter ranking can change with different combinations of assigning PDFs. It is critical that no important parameters are missed as assigning a fixed value essentially assigns no uncertainty to those parameters. I could not find any section that described the logic of choosing which parameters to assign PDFs and which parameters to treat as fixed values. Second, I am somewhat concerned with the consistent (but not always) assignment of three sigma about the mean for defining the min and the max of PDFs. I am glad to see truncation of the normal distributions but a standard three sigma approach can overestimate uncertainty particularly when you consider the affects of scaling and parameter averaging. You have to be sure you are assessing the PDF for a parameter at the model scale (equivalent to the standard error of the mean) and not based on the measurement variance of the parameter measurements. Too broad of parameter distributions can often result in spurious combinations of parameter tail values and lead to physically implausible model output. Third, you should complete global sensitivity analysis as an aid to the SA for your probabilistic model output using the probabilistic output.

Response: See response below:

1. *PDFs were essentially assigned to all model parameters where there was some basis for developing a distribution (e.g., distributions of stream flows based on long term USGS measurements) or distributions already existed. All other parameters, for which no current basis existed for development of a distribution, were held fixed (e.g., inventories). A systematic analysis evaluating which model parameters have the greatest impact on the results was not performed. This approach has been included as an item for future work in Table 11-2.*
2. *Use of the three sigma bounds was chosen to include 99.5% of the distribution in the analysis and did not appear to create any problems. However the comment is correct that using a three sigma bound may lead to extreme combinations of parameter values. In retrospect, a two sigma bound would have been a better choice and has been added to Table 11-2 for consideration as potential future work.*
3. *Table 11-2 has been revised to outline potential future work to be conducted to investigate the use of a global sensitivity analysis with help from the SRNL statistics group to design a systematic approach to model sensitivity and uncertainty analysis and to help interpret the results.*

H. APPENDIX H: LFRG REVIEW TEAM OBSERVATION RESOLUTION

Observation Resolution to the Department of Energy Low-Level Waste Disposal Facility Federal Review Group Review Team Review of the Savannah River Site DOE 435.1 Composite Analysis, Revision 0 Draft for DOE LFRG Review

The Low-Level Waste Disposal Facility Federal Review Group (LFRG) Review Team produced twenty-three observations from their review of the Savannah River Site (SRS) Composite Analysis (CA) (Carilli and Golian 2010). They identified one secondary issue through the consolidation of eighteen observations that the team concluded, when evaluated collectively, could potentially impact the integration of the Composite Analysis (CA) results.

- Nine observations involved missing information associated with the CA (see observations 1, 3, 9, 11, 12, 18, 19, 20, and 23).
- Nine observations were related to ensuring specific future work items listed in Chapter 11 were included in the CA maintenance plan (see observations 2, 4, 5, 6, 7, 13, 14, 16, and 21).

Observations identified below with one or two asterisks were consolidated into the secondary issue (one asterisk indicates "missing" information that was rolled up into the secondary issue and two asterisks indicate priority future work that was rolled up into the secondary issue). This secondary issue is important but not critical to the SRS CA acceptance. It describes areas for improvement that must be tracked and resolved through the SRS CA maintenance plan and reported through the site annual summary submitted to the LFRG.

The remaining five observations (i.e., 8, 10, 15, 17, and 22), which are not identified with an asterisk, represent observations that were not included in the secondary issue. These observations are conditions or practices, which while not a violation of an established requirement, indicate a less than optimal performance and are worthy of being raised to the attention of management for evaluation/improvement.

The LFRG Review Team observations and recommendations presented below were extracted from Carilli and Golian 2010. The Savannah River National Laboratory (SRNL) response provides the actions taken in response to the team's observations and recommendations, in particular any necessary revisions to the SRS CA are discussed.

<u>Observation 1</u>*: The CA acknowledges that the Vogtle Nuclear Power Station and Chem-Nuclear Barnwell Radioactive Waste Disposal Site are near the site boundaries.

Recommendation: A justification for not including these sources needs to be in the document.

SRNL Response:

The following paragraph has been added at the end of Section 10.3:

> *"DOE guidance for the Composite Analysis (DOE 1999) requires that impacts of commercial nuclear operations be considered, if warranted. Consistent with requirements in DOE Order 5400.5, doses from non-DOE activities need be considered only when (1) the dose to an individual member of the public from DOE activities exceeds 30 mrem in a year and (2) the dose from the non-DOE activities also exceeds 30 mrem in a year to the same individual. As indicated in Section 3.2.3.2, there are two non-SRS nuclear facilities within 50 miles of the SRS. These are the Chem-Nuclear Systems, Inc. Low-Level Radioactive Waste Disposal Facility in Barnwell, SC, and the Vogtle Electric Generating Plant, a nuclear power facility located across the Savannah River from the SRS. Radionuclide releases from these facilities or residual radioactive material left at these facilities when operations cease could impact members of the public that are potentially impacted by SRS. However, because the maximum dose from the CA is about three mrem/year, which is less than the dose constraint of 30 mrem/year, these non-SRS facilities need not be considered."*

A reference for DOE 1999 has also been added: DOE (Department of Energy). 1999. Format and Content Guide for U.S. Department of Energy Low-Level Waste Disposal Facility Performance Assessments and Composite Analyses, U.S. Department of Energy, Washington, DC, December 7, 1999.

Additionally, a footnote has been added on page 3-32 to the first mention of the Chem-Nuclear Systems and Vogtle facilities. The footnote states: "Although these non-SRS commercial nuclear operations may release radionuclides into the Savannah River and may leave residual radioactive material when operations cease, they are not considered in this SRS CA because the maximum dose projected in the CA is less than 30 mrem in a year (see Section 10.3)."

Observation 2**: The inventory of radionuclides other than Tritium (H-3) in GSA groundwater plumes has not been estimated and evaluated. Yet site monitoring data indicates that other radionuclides of concern (e.g., Strontium 90 (Sr-90), Technetium 99 (Tc-99), Cesium 137 (Cs-137), Iodine 129 (I-129)) are present in General Separations Area (GSA) groundwater and being released to Four Mile Branch.

Recommendation: The maintenance program should address the neglected inventory of detected radionuclides in GSA groundwater plumes and their impacts assessed through CA screening or dose assessment, as appropriate.

SRNL Response:

Previous Section 11.2, Potential Future Work Under Consideration, has been updated and renamed and previous Table 11-2, Potential Future Work Items, has been split and modified to address the future work identified as priority items by the LFRG SRS CA Review Team. The Table 11-2, Potential Future Work Items, which were categorized as priority future work items within Observations 2, 4, 6, 7, 13, 14, 16, and 21, were incorporated into a new Table 11-2, Priority Future Work Items. These future work items within the new Table 11-2 have been or will be incorporated into the SRNS PA/CA maintenance plan to be addressed as priority items.

Item 8 of the new Table 11-2 states the following in response to Observation 2:

> *"The inventory and inventory distribution for radionuclides within the F and H-Area Seepage Basins groundwater plumes, such as Strontium 90, Technetium 99, Cesium 137, Iodine 129, should be developed in addition to that for tritium. This inventory and its distribution should be evaluated within the SRS CA through screening or an actual dose assessment, as appropriate."*

Observation 3*: Various underground waste transfer pipelines among the high-level waste tanks, the Defense Waste Processing Facility, and the Saltstone Disposal Facility are present on site but how the inventory for these transfer lines was included in the CA was not clearly and explicitly documented.

Recommendation: How these sources of radioactive materials were included in the CA needs to be clearly and explicitly documented.

SRNL Response:

Section 8.2.4.7.2.2 for the FTF ancillary equipment inventory, Section 8.2.5.1.2.2 for the HTF ancillary equipment inventory, Section 8.2.8.1.2 for the DWPF inventory, and Section 8.2.19.1.2 for the Saltstone inventory have been revised to clearly indicate the following as appropriate:

- *The Table A-34 FTF ancillary equipment inventory includes half the inventory of the inter-area transfer line between FTF and HTF, and*
- *The Table A-38 HTF ancillary equipment inventory includes half the inter-area transfer line between FTF and HTF, the entire transfer line between HTF and the Defense Waste Processing Facility (DWPF), and the entire transfer line between HTF and the Saltstone Processing Facility (SPF).*

Observation 4**: Limitations of the current CA inventory and its uncertainty are well recognized by SRS. Section 11.1 of the CA discusses CA related maintenance items, which includes the maintenance of the SRS CA Inventory Database. Future work items of Table 11-2 also include items related to inventory updates and uncertainty assessments (Items 1, 3, 7, 13, 14, 15, 16, 17, and 21).

Recommendation: If not already initiated these items should be given top priority in the maintenance plan.

SRNL Response:

Previous Section 11.2, Potential Future Work Under Consideration, has been updated and renamed and previous Table 11-2, Potential Future Work Items, has been split and modified to address the future work identified as priority items by the LFRG SRS CA Review Team. The previous Table 11-2, Potential Future Work Items, which were categorized as priority future work items within Observations 2, 4, 6, 7, 13, 14, 16, and 21, were incorporated into a new Table 11-2, Priority Future Work Items. These future work items within the new Table 11-2 have been or will be incorporated into the SRNS PA/CA maintenance plan to be addressed as priority items.

Items 1, 3, 7, 13, 14, 15, 16, 17, and 21 from the previous Table 11-2 have been incorporated into the revised Table 11-2, Priority Future Work Items, as items 5, 1, 6, 3, 2, 9, 7, 4, and 10, respectively. Items 1 and 5 have already been specifically added to the 2010 SRNS PA/CA maintenance plan, and work on these items have been initiated. Items 2, 3, 4, 6, 7, 9, and 10 will be incorporated into the future 2011 SRNS PA/CA maintenance plan as outlined within the 2010 SRNS PA/CA maintenance plan (SRNS 2010) Section 3.2.2, Address Issues Assigned to Maintenance from the LFRG Review of the SRS CA.

Observation 5**: The CA recognizes the limitations of the characterization data currently in the CA, as evidenced by the CA Maintenance Items listed and summarized in Section 11.1, as well as the future work items listed in Table 11-2. These include revisions of the properties and geochemical data packages, sorption behavior of key PA radionuclides, fate of Carbon 14 (C-14) and I-129 at the seeplines.

Recommendation: Add the above items to the CA maintenance plan.

SRNL Response:

As outlined in Section 11.1 of the CA, material properties data package revision, geochemical data package revision, sorption behavior of key PA radionuclides, fate of C-14, and phenomenon of I-129 at the seepline were all incorporated within the 2009 SRNS PA/CA maintenance plan (SRNS 2009) within Sections 5.1.1, 5.1.2, 5.1.9, 5.2.1, and 5.2.3 of that plan, respectively. Investigation of the geochemistry and environmental fate of C-14 in the SRS environment has been completed with issuance of the following two reports:

- *Carbon-14 Geochemistry at Savannah River Site (Roberts and Kaplan 2008), and*
- *Systems Model of Carbon Dynamics in Four Mile Branch on the Savannah River Site (Hinton et al. 2009)*

Material properties data package revision, geochemical data package revision, sorption behavior of key PA radionuclides, and phenomenon of I-129 at the seepline have all been maintained within the 2010 SRNS PA/CA maintenance plan (SRNS 2010) within Sections 5.1.1, 5.1.2, 5.1.8, and 5.2.1, and of that plan, respectively, because they were not completed in 2009. The fate of C-14 was dropped from the 2010 SRNS PA/CA maintenance plan (SRNS 2010) due to its completion within 2009.

Observation 6**: Radionuclide streambed inventories are minimally described

Recommendation: Radionuclide inventory of the streambed sediments should be better quantified as described in Table 11-2, item 7 and added to the CA maintenance plan.

SRNL Response:

Previous Section 11.2, Potential Future Work Under Consideration, has been updated and renamed and previous Table 11-2, Potential Future Work Items, Potential Future Work Items, has been split and modified to address the future work identified as priority items by the LFRG SRS CA Review Team. The previous Table 11-2, Potential Future Work Items, which were categorized as priority future work items within Observations 2, 4, 6, 7, 13, 14, 16, and 21, were incorporated into a new Table 11-2, Priority Future Work Items. These future work items within the new Table 11-2 have been or will be incorporated into the SRNS PA/CA maintenance plan to be addressed as priority items.

Item 7, from the previous Table 11-2, has been incorporated into the revised Table 11-2, Priority Future Work Items, as item 6. Item 6 will be incorporated into the future 2011 SRNS PA/CA maintenance plan as outlined within the 2010 SRNS PA/CA maintenance plan (SRNS 2010) Section 3.2.2, Address Issues Assigned to Maintenance from the LFRG Review of the SRS CA.

Observation 7**: No discussion is provided in the CA regarding the streambed sediment scour, deposition, and transport characteristics.

Recommendation: Item 8 of Table 11-2 should be expanded to include these streambed transport characteristics and added to the CA maintenance plan.

SRNL Response:

Previous Section 11.2, Potential Future Work Under Consideration, has been updated and renamed and previous Table 11-2, Potential Future Work Items, has been split and modified to address the future work identified as priority items by the LFRG SRS CA Review Team. The previous Table 11-2, Potential Future Work Items, which were categorized as priority future work items within Observations 2, 4, 6, 7, 13, 14, 16, and 21, were incorporated into a new Table 11-2, Priority Future Work Items. These future work items within the new Table 11-2 have been or will be incorporated into the SRNS PA/CA maintenance plan to be addressed as priority items.

Item 8, from the previous Table 11-2, has been incorporated into the revised Table 11-2, Priority Future Work Items, as item 13. It has been rewritten to include streambed sediment scour, deposition, and transport. Item 13 will be incorporated into the future 2011 SRNS PA/CA maintenance plan as outlined within the 2010 SRNS PA/CA maintenance plan (SRNS 2010) Section 3.2.2, Address Issues Assigned to Maintenance from the LFRG Review of the SRS CA.

Observation 8: Radionuclide mass fluxes among groundwater, streambed sediments, and surface water are greatly simplified. Spatial heterogeneities of streambed sediments and underlying aquifers can cause highly variable patterns of exchange fluxes and groundwater discharge. Current CA does not characterize groundwater discharge and radionuclide fluxes along the streams, and the release of radionuclides from the streambed sediments, both of which remain highly uncertain.

Recommendation: CA maintenance tasks should be added to better quantify radionuclide mass fluxes into stream bed sediments. This concern should be addressed as part of the water balance and SRS-wide groundwater model developments proposed in Items 5 and 18 of Table 11-2.

SRNL Response:

The following item has been added to Table 11-3, Potential Future Work Items, which will be evaluated during production of the 2011 SRNS PA/CA maintenance plan and added to the plan as warranted:

> *"Radionuclide mass fluxes into stream bed sediments should be better quantified as part of the Table 11-2 Item 11 water balance study and Table 11-2 Item 12 SRS regional groundwater model development."*

Additionally a footnote to Table 11-3 has been added denoting that this item was identified by the LFRG SRS CA Review Team as an item that should be considered for addition to the SRNS PA/CA maintenance plan, in order to provide it additional priority during evaluation of its addition to the maintenance plan.

Observation 9*: Section 7.2.2.5.5 and Appendix A-4 states 'due to the absence of dosimetry data for these radionuclides from extensive lists (National Council on Radiation Protection and Measurements (NCRP) 1996, International Commission on Radiological Protection (ICRP) 72, Federal Guidance Report (FGR) 12), it is concluded that these radionuclides would rarely be present…'. To screen out a radionuclide because it has not yet been considered by NCRP or ICRP, is not technically justified.

Recommendation: Sentence should be deleted or modified to reflect the technical justification for the exclusion of Argon 42 (Ar-42), Bismuth 208 (Bi-208), Gadolinium 150 (Gd-150), Niobium 92 (Nb-92), and Polonium 209 (Po-209).

SRNL Response:

This observation relates to Appendix A-4 of the radionuclide screening report (SRNS-STI-2008-00117). However, the problematic text also appears in the CA in Section 7.2.2.5.5 on page 7-16. The offending sentence has been deleted.

Observation 10: Any change in the boundaries of SRS during the period of assessment will require reevaluation of suitability of the selected points of assessment.

Recommendation: If the boundaries of SRS change or are projected to change during the period of assessment, the location of the POAs must be re-evaluated and changed if necessary. Then the CA must be reviewed and revised accordingly.

SRNL Response:

The following sentence has been added to the end of the first operations assumption in Appendix F (i.e. SRS land use plan assumption) to address this observation:

> *"Any change in the SRS land use plan that eases the prohibition of unrestricted public access and residential land use over any portion of the existing SRS site, will require a reevaluation of the CA POAs and potentially a revision of the CA."*

Observation 11*: The discussion of assumptions used to evaluate the transport of radionuclides in the main document lacked sufficient detail.

Recommendation: The process for the evaluation of the transport of radionuclides is described and justified in SRNS-STI-2008-00117, Revision 0. It is recommended that elements of Chapters 1-6 in Appendix A in this reference be included in greater detail in the CA document.

SRNL Response:

Table 2-1 from SRNS-STI-2008-00117 (Element specific K_ds used in the CA screening model) has been inserted at the end of section 7.2.2.5.4. Sections 4, 5, and 6, including Tables 4-1, 5-1, 6-1, 6-2, 6-3, and 6-4, and the text from sections 7.1 and 7.2 from SRNS-STI-2008-00117 have been inserted before 7.2.2.6.3.

Observation 12*: Discussion of the dose conversion factors (DCF) should clearly indicate that FGR 12 and ICRP Publication 72 DCFs were used for dose calculations.

Recommendation: Include a statement in the description of dose calculations used that FGR 12 and ICRP Publication 72 DCFs were used for all dose calculations.

SRNL Response:

Statements were added to Section 9.3.2.2, Dose Assessment Calculations and 9.3.4.10, Dose Module Input Parameters, indicating the source of the dose conversion factors were ICRP Publication 72 for internal doses, and EPA's FGR 12 for external doses.

Observation 13**: The SRS CA acknowledges the limitations of the current CA methodology: (1) lack of SRS-wide watershed model and groundwater flow and transport model; (2) need to reduce uncertainty of the abstractions from the existing groundwater flow and transport models into the GoldSim™ CA model, (3) limitations of the version of the GoldSim™ code used in the CA, and (4) limited capability built into the current GoldSim™ CA model to perform probabilistic analysis. Development of an SRS-wide watershed model and groundwater model are proposed in items 5 and 18 of Table 11-2.

Recommendation: Add the above specified items to the CA maintenance plan.

SRNL Response:

Previous Section 11.2, Potential Future Work Under Consideration, has been updated and renamed and previous Table 11-2, Potential Future Work Items, has been split and modified to address the future work identified as priority items by the LFRG SRS CA Review Team. The previous Table 11-2, Potential Future Work Items, which were categorized as priority future work items within Observations 2, 4, 6, 7, 13, 14, 16, and 21, were incorporated into a new Table 11-2, Priority Future Work Items. These future work items within the new Table 11-2 have been or will be incorporated into the SRNS PA/CA maintenance plan to be addressed as priority items.

Items 5 and 18, from the previous Table 11-2, have been incorporated into the revised Table 11-2, Priority Future Work Items, as items 11 and 12, respectively. Items 11 and 12 will be incorporated into the future 2011 SRNS PA/CA maintenance plan as outlined within the 2010 SRNS PA/CA maintenance plan (SRNS 2010) Section 3.2.2, Address Issues Assigned to Maintenance from the LFRG Review of the SRS CA.

Observation 14**: To reduce uncertainty, the GoldSim™ CA model should be improved, by better abstraction of groundwater flow paths, flow rates, and discharges to the streams. Water balance and SRS-wide groundwater models (proposed in Table 11-2), and studies of streambed sediment characterization and groundwater-surface water interactions should be completed to provide the basis for these improvements.

Recommendation: Add the above specified items to the CA maintenance plan.

SRNL Response:

Previous Section 11.2, Potential Future Work Under Consideration, has been updated and renamed and previous Table 11-2, Potential Future Work Items, has been split and modified to address the future work identified as priority items by the LFRG SRS CA Review Team. The previous Table 11-2, Potential Future Work Items, which were categorized as priority future work items within Observations 2, 4, 6, 7, 13, 14, 16, and 21, were incorporated into a new Table 11-2, Priority Future Work Items. These future work items within the new Table 11-2 have been or will be incorporated into the SRNS PA/CA maintenance plan to be addressed as priority items.

Items 5, 8, and 18, from the previous Table 11-2, dealing with water balance, SRS-wide groundwater models, and studies of streambed sediment characterization and groundwater-surface water interactions, have been incorporated into the revised Table 11-2, Priority Future Work Items, as items 11, 13, and 12, respectively. Item 13 has been revised to specifically address groundwater-surface water interactions. Items 11, 12, and 13 will be incorporated into the future 2011 SRNS PA/CA maintenance plan as outlined within the 2010 SRNS PA/CA maintenance plan (SRNS 2010) Section 3.2.2, Address Issues Assigned to Maintenance from the LFRG Review of the SRS CA.

Observation 15: Future work item 1 of Table 11-2 proposes to utilize the distributed processing capability of GoldSim™ in order to perform probabilistic simulations more efficiently. The GoldSim™ model version should be updated to the latest version, which now has capability to handle distributed inputs to pipe elements. To cut run time, the SRS CA team should consider using these pipe elements instead of the huge number of cell elements used in the current analysis.

Recommendation: Add the above specified items to the CA maintenance plan.

SRNL Response:

The following item has been added to Table 11-3, Potential Future Work Items, which will be evaluated during production of the 2011 SRNS PA/CA maintenance plan and added to the plan as warranted:

> *"In order to cut run times, future updates of the SRS CA GoldSim™ model should consider updating to the latest GoldSim™ version so that the new pipe elements, which can handle distributed inputs, could be used rather than the huge number of cell elements used in the current SRS CA GoldSim™ model."*

Additionally a footnote to Table 11-3 has been added denoting that this item was identified by the LFRG SRS CA Review Team as an item that should be considered for addition to the SRNS PA/CA maintenance plan, in order to provide it additional priority during evaluation of its addition to the maintenance plan.

Observation 16**: The CA acknowledges the uncertainty of the radionuclide releases from streambed sediments. Items 1, 5, 7, 8, and 9 (Chapter 11) in Table 11-2 address work proposed to further investigate releases of radionuclides from streambed sediments. Observations of LFRG M Review Criteria 3.3.2.3 should be considered to address this uncertainty.

Recommendation: Add the above specified items to the CA maintenance plan.

SRNL Response:

Previous Section 11.2, Potential Future Work Under Consideration, has been updated and renamed and previous Table 11-2, Potential Future Work Items, has been split and modified to address the future work identified as priority items by the LFRG SRS CA Review Team. The previous Table 11-2, Potential Future Work Items, which were categorized as priority future work items within Observations 2, 4, 6, 7, 13, 14, 16, and 21, were incorporated into a new Table 11-2, Priority Future Work Items. These future work items within the new Table 11-2 have been or will be incorporated into the SRNS PA/CA maintenance plan to be addressed as priority items.

Items 1, 5, 7, 8, and 9, from the previous Table 11-2 have been incorporated into the revised Table 11-2, Priority Future Work Items, as items 5, 11, 6, 13, and 14, respectively. Item 5 has already been specifically added to the 2010 SRNS PA/CA maintenance plan, and work on this item has been initiated. Items 6, 11, 13, and 14 will be incorporated into the future 2011 SRNS PA/CA maintenance plan as outlined within the 2010 SRNS PA/CA maintenance plan (SRNS 2010) Section 3.2.2, Address Issues Assigned to Maintenance from the LFRG Review of the SRS CA.

Additionally the following note has been added to Table 11-2 to address LFRG Manual Review Criteria 3.3.2.3 considerations:

> *"Note to Table 11-2: Items 5, 6, 11, 13, and 14, associated with the streambed radionuclide inventory and transport, should consider the physical and chemical characteristics of the radionuclides within the streambed, the streambed characteristics, and the effects of any Comprehensive Environmental Responses, Compensation, and Liability Act (CERCLA) actions prescribed in the Record of Decisions (RODs) or similar binding agreements for the contaminated streambeds. Additionally any extrapolation made from known data to estimate the streambed radionuclide inventory should be adequately justified."*

Observation 17: If SRS-wide watershed modeling and groundwater flow and transport modeling tasks are undertaken, consistency between PA models and the CA model would be further improved

Recommendation: The site may want to include the above in the CA Maintenance Plan.

SRNL Response:

The following item has been added to Table 11-3, Potential Future Work Items, which will be evaluated during production of the 2011 SRNS PA/CA maintenance plan and added to the plan as warranted:

> *"Once the Table 11-2 Item 11, water balance study and Table 11-2 Item 12, SRS regional groundwater model development have been completed, this information and model should be used for future SRS Performance Assessment (PA) and future CA models to improve consistency between them."*

Additionally a footnote to Table 11-3 has been added denoting that this item was identified by the LFRG SRS CA Review Team as an item that should be considered for addition to the SRNS PA/CA maintenance plan, in order to provide it additional priority during evaluation of its addition to the maintenance plan.

Observation 18*: Section 10.3 does not reflect all drinking water pathways. Additional water treatment plants have been built downstream of the site along the Savannah River in the past few years.

Recommendation: Include the newly identified water treatment plants along the Savannah River, downstream of the site.

SRNL Response:

The last paragraph of Section 3.2.9.1.1 has been revised to read:

> *"The Savannah River downstream from Augusta, GA, is classified by the State of South Carolina as a Class B waterway, which is suitable for agricultural and industrial use, the propagation of fish, and after treatment, domestic use. Below the mouth of UTR, raw water is pumped from the Savannah River for drinking water supplies. The City of Savannah Industrial and Domestic Water Supply Plant (Savannah I&D), near Port Wentworth, GA (Figure 3-19) withdraws water to supply an estimated consumer population of about 26,300. The Beaufort-Jasper Water and Sewer Authority (BJWSA) has two treatment plants near Hardeeville, SC (Figure 3-19), which supply consumer populations of approximately 58,000 (Purrysburg facility) and 77,000 (Chelsea facility). Use of the Savannah River downstream of the SRS, including water-contact recreation, is less extensive than it is upstream of SRS (Mamatey 2008)."*

Observation 19*: The discussion of exposure pathways was found to be complete and easily understood. The use of flow charts greatly enhanced the pathways discussion. However, the dosimetry discussion was found to be lacking. While the lack of discussion by no means diminishes the excellent work performed, it caused the document to lack transparency and made review difficult.

In general, the dosimetry models were difficult to decipher. The discussion of the dosimetry was found in bits and pieces but no details were presented in a structured and orderly manner. The process had been used in previous PAs and therefore much of the discussion was incorporated by reference, which made it difficult for reviewers to find the information expected to be discussed in exposure scenarios. General discussions were found in several chapters, appendices or references. For instance:

- Chapter 6 contained one sentence that indicated the final dose calculations were coupled to the transport modeling,
- Chapter 7 discussed screening which was not coupled to the final site specific analyses,
- Chapter 9, presented equations and dose conversion factors and other references contained the rational for parameter selection or more complete descriptions of the exposure scenarios.

Recommendation: Pulling all this information together in one place would greatly improve the transparency of the CA document.

This entire dose methodology and screening would benefit from a flow chart that illustrates the various steps taken to get to the final analytical runs in terms of scenario and radionuclide screening as well as selection of generic and detailed model criteria and how specific radionuclides were carried through the analyses. An example of such a flow chart is included in Appendix C of this report.

SRNL Response:

A flow chart was developed and inserted into Section 9.3.2 Dose Module, illustrating the various steps taken to get to the final dose results. A statement referring to this flow chart was added at the beginning of this section.

Observation 20*: The report emphasized the selection of pathways, transport and modeling but the dosimetry discussion should be expanded to include the methodology and rationale.

Recommendation: Expand the dosimetry discussion to include methods, rationale behind the scenario, and dose model and parameter selection.

SRNL Response:

Section 9.3.2, Dose Module, was revised as follows. First a flow chart was developed and inserted into the section, illustrating the various steps taken to get to the final dose results. A statement referring to this flow chart was added at the beginning of this section.

Second, additional information regarding selection of exposure pathways used in developing exposure scenarios was added to Section 9.3.2.1, Receptors and Scenarios. This information is summarized from Section 7.1, and reference is made back to that section as a source of specific details of the rationale for excluding pathways from consideration. The format of Section 9.3.2.1 is altered somewhat to more clearly list the exposure pathways relevant to the two exposure scenarios considered, for which doses are calculated according to equations provided in Section 9.3.2.2.

Third, the presentation of the dose assessment calculations (Section 9.3.2.2) was revised to more clearly provide the dose model (equation) for each exposure pathway in each scenario, followed by any supporting equations (in a more top-down approach). Underlying assumptions, and citation of the main resource document for the equations (Jannik and Dixon 2006), are provided. A reference to Section 7.2 as the source of the description of parameters (this information was added in response to Observation 11) found in the equations is added. The specific tables in Appendix C where parameter values used in implementing the equations are provided are also identified in Section 9.3.2.2. To clarify the relationship between exposure parameters in the equations of Section 9.3.2.2 with their values in Table C-18, an additional column was added to that table, which provides the corresponding parameter name from the equations in Section 9.3.2.2 (not just the GoldSim™ parameter name).

Fourth, the last paragraph of Section 9.3.4.10, Dose Module Input Parameters, was also revised to more clearly identify the input data for the dose module, the source of the data, and where that data can be found in the Appendix C tables.

Observation 21**: There is agreement with previous comments (Appendix G) identifying the incompleteness of the sensitivity and uncertainty analysis. In particular, the omission of inventory variability in the uncertainty analysis needs to be remedied. However, because all dose results are well below levels of concern, qualitative arguments used to address these limitations provide enough confidence to accept the primary conclusions of the sensitivity and uncertainty analysis. Furthermore, the future CA maintenance activities (particularly items 17, 27 and 28 in Table 11-2) identify these limitations and the need to address them. Given improved input data distributions and an improved GoldSim™ model, additive and multiplicative effects of factors affecting the CA results could be better assessed through a global sensitivity analysis. This effort would also better streamline future maintenance task priorities.

SRNL Response:

Previous Section 11.2, Potential Future Work Under Consideration, has been updated and renamed and previous Table 11-2, Potential Future Work Items, has been split and modified to address the future work identified as priority items by the LFRG SRS CA Review Team. The previous Table 11-2, Potential Future Work Items, which were categorized as priority future work items within Observations 2, 4, 6, 7, 13, 14, 16, and 21, were incorporated into a new Table 11-2, Priority Future Work Items. These future work items within the new Table 11-2 have been or will be incorporated into the SRNS PA/CA maintenance plan to be addressed as priority items.

Items 2, 17, 27, and 28, from the previous Table 11-2, addressing improvements to sensitivity and uncertainty analyses, have been incorporated into the revised Table 11-2, Priority Future Work Items, as items 15, 4, 16, and 17, respectively. Item 15 has already been specifically added to the 2010 SRNS PA/CA maintenance plan, and work on this item has been initiated. Items 4, 16, and 17 will be incorporated into the future 2011 SRNS PA/CA maintenance plan as outlined within the 2010 SRNS PA/CA maintenance plan (SRNS 2010) Section 3.2.2, Address Issues Assigned to Maintenance from the LFRG Review of the SRS CA.

Observation 22: The CA archive should be reviewed for adequacy, relevancy, and correctness. SRS should ensure GoldSim™ 9.6 Service Pack 4 and any other executable programs needed to reproduce the CA results are included in the archive.

SRNL Response:

The SRNL CA technical lead and the SRNL CA modeling lead will review the files in the CA Archive holding area for adequacy, relevancy, and correctness prior to placement in the permanent CA Archive.

Copies of the installation programs for GoldSim™ version 9.60 SP4 and the GoldSim™ player version 9.60 SP4 have been placed in the CA Archive holding area for eventual placement in the permanent CA Archive. The actual executable GoldSim™ program will not be archived, because as archived it would not be able to be run. This is because the dynamic link libraries (DLLs) and registry settings would not be able to be used because the Windows registry would be incorrect. However because the GoldSim™ version 9.60 SP4 installation programs are archived, SRNL will be able to install and run GoldSim™ if it becomes necessary.

Observation 23*: An outdated NCRP Report is referenced in the discussion of average background dose to members of the public (Section 3.3.2).

Recommendation: The CA should be updated to reflect the more current estimates of background exposure in NCRP Report No. 160.

SRNL Response:

Section 3.3.2, Natural Background and Anthropogenic Source of Radiation, text and Figure 3-81 have been replaced with text and a figure, based upon the more current estimates of background exposure provided in NCRP Report No. 160 (NCRP 2009).

Reference

Carilli, J. T. and Golian, S. 2010. Savannah River Site Composite Analysis Review Report, Department of Energy Low-Level Waste Disposal Facility Federal Review Group Review Team, April 20, 2010.

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