BOEING

Team Product Document

GO Number	S/A Number	Page 1 of	Total Pages	Rev. Ltr/Chg. No. See Summary of Chg.	Number
97055	12120	78	78	F	RS-00011

Program Title: ETEC (R21-RS)

Document Title : Procedures for Surveys of Radioactive Shipments

Document Type: Operating Procedure

Related Documents:

Original Issue Date: 11/20/99 Release Date: RELEASE 05-17-01 CV

Prepared By/Date	Dept.	Mail/Addr
James G. Barnes, CHP 11/20/99	641	055-T038

Approvals	Date
P. D. Rutherford	11/19/99
M. Lee	11/19/99
S. Reeder	11/19/99

IR&D Program? Yes ☐ No ☒

If Yes, Enter Authorization No.

Distribution

*	Name	Mail Addr.
*	Barnes, J. G.	T038
*	Rutherford, P. D.	T038
*	Radiation Safety Staff (12)	T038
*	Radiation Safety Files (6)	T038
*	SHEA Files	T487
*	Bassat, I. B.	T038
	Lee, M.	T038
	Lafflam, S. R.	T487

Abstract

This document describes the procedures and methods to be used to perform radiological monitoring and characterization of radioactive materials in preparation for shipment and disposition as radioactive waste.

This procedure supersedes N001OP000030 of the same title.

* Complete Document
No Asterisk, Title Page/Summary or Change Page Only.

Reserved for Proprietary/Legal Notice

Team Product Document Summary of Change

Rev.	Summary of Change	Approvals and Date
A	Adds Section 9.0 which describes process for verification of calculation process, software and output. Makes minor editorial changes to other sections of document.	P. D. Rutherford M. Lee S. Reeder R. Amar RELEASE 05-17-01CV
B	Minor Updates to Reference Section Change designator R21-WM to R21-RS. Section 1.0: Appendix A was created to list the Waste Site references. This is to remove them from the body of the procedure and make future referencing easier. All appendices were re-numbered and re-referenced throughout the procedure. Section 4.2: Minor editorial changes. Section 5.2: Clarification of scope of 49 CFR 173.435 Section 5.3: Minor editorial changes. Section 5.7: Minor editorial changes. Adds discussion of surveying of Surface Contaminated Objects (SCO), and move discussion of smear sampling of contamination to this section. This move was made to put sampling guidance into one section of the procedure. Section 5.8: Role of the Lot Follower in the characterization process was clarified. Section 5.9 and 5.10. These sections were combined as they deal predominantly with limits. All tables of limits were moved into a new Appendix D. All other appendices were renumbered and re-referenced. Section 5.11 was renumbered to Section 5.10. A note regarding site specific limits was added. Section 6.0: A note regarding site specific limits was added. Section 7.0: The limit table was relocated to Appendix D Section 8.0: Minor editorial changes. Requirements for release of materials to the end user were clarified. Appendix A: A table of waste site criteria documents was added. Appendix C: Pu-238 added as SNM. Clarification regarding transuranic concentrations was added. An incorrect gross weight in an example was corrected. Adds ISOCS. Scaling factors described for use. Appendix D: All limit tables and references were consolidated into this Appendix.	P. D. Rutherford Ravnesh Amar M. Lee D. Koncel

Team Product Document
Summary of Change

Rev.	Summary of Change	Approvals and Date
C	Updates forms referenced in the procedure. Appendices renumbered. Forms appendix moved to end of procedure. App F to B; App G to C; App H to D; App B to E Procedure repaginated from 56 to 57 pages.	P. D. Rutherford R. Amar D. Koncel
D	Updates references to revisions of 49 CFR. Minor changes in language as noted by change bars. A "wiping efficiency" of a smear has been defined. Clarify role of SHEA in DOT shipping of radioactive materials. Requires incident report be prepared if DOT receipt limits exceeded. Incorporates new requirements defining what constitutes a radioactive shipment under 49 CFR. Updates 708R Tag Format	P. D. Rutherford R. Amar D. Koncel
E	Adds requirement for direct alpha survey of incoming vehicles (Sect. 7) Adds specimen incoming survey form	P.D. Rutherford R. Amar J. Vargo
F	Adds description of redesigned characterization spreadsheet tool (App F). Defines the "Radiation Safety" function of SHEA Appendices re-ordered. Minor clarifications. (Changes are highlighted in yellow) Sections of former Appedix C have been excised and put in new App G.	P. D. Rutherford R. Amar J. Vargo

[INTENTIONALLY LEFT BLANK]

TABLE OF CONTENTS

1.0 PURPOSE AND SCOPE 7

2.0 APPLICABLE DOCUMENTS AND REFERENCES 8

2.1 Regulatory documents 8

2.2 Boeing procedures / Supporting documents 9

2.3 Other references 9

3.0 DEFINITIONS 10

4.0 RESPONSIBILITIES 11

4.1 GENERAL 11

4.2 Radiation Safety 11

4.3 Shipping Departments 11

5.0 SHIPMENT CHARACTERIZATION 12

5.1 General 12

5.2 IDENTIFICATION OF RADIOISOTOPES 12

5.3 ESTIMATED ACTIVITY 12

5.4 SURFACE CONTAMINATED ITEMS 13

5.5 DISTRIBUTED CONTAMINATION 13

5.6 WASTE SAMPLING PROCEDURES 13

5.7 Sampling Methods: 14

5.8 Radioactive Waste Lot Follower 19

5.9 RADIATION INTENSITIES AND CONTAMINATION LEVELS FOR PACKAGES AND CONTAINERS 20

5.10 TYPE OF ACTIVITY 20

6.0 SURVEYS OF INCOMING AND DEPARTING LOADED, EXCLUSIVE USE VEHICLES 21

7.0 SURVEYS OF EMPTY EXCLUSIVE USE VEHICLES 22

8.0 SURVEYS OF INCOMING PACKAGES CONTAINING RADIOACTIVE MATERIALS 23

9.0 DATA QUALIFICATION AND REVIEW 24

10.0 FORMS AND RECORDS 25

APPENDIX A. WASTE SITE ACCEPTANCE CRITERIA DOCUMENTS 27

APPENDIX B. FORMULAS, SPECIFIC ACTIVITIES, AND UNIT CONVERSTIONS[5] 28

APPENDIX C. LIMITS AND CONDITIONS APPLICABLE TO SHIPMENTS 33

APPENDIX D. SPECIMEN FORMS. 38

APPENDIX E. DISCUSSION AND EXAMPLES OF ACTIVITY ESTIMATION FOR CONTAMINATED MATERIALS 44

APPENDIX F; INSTRUCTIONS FOR USE OF CHARACTERIZATION SOFTWARE TOOL 48

APPENDIX G. INSTRUCTIONS FOR USE OF ALTERNATE SOFTWARE TOOLS, WITH EXAMPLES 67

1.0 PURPOSE AND SCOPE

These procedures apply to the generation of radiological characterization data for radioactive material shipments and to the check-in and check-out surveys for vehicles which have or are intended to transport radioactive material on the public roads or highways. These procedures conform to the requirements of the state and federal agencies and Boeing policy in SOP C-404, *Shipping Radioactive Materials*, which details the policies to carry out the state and federal requirements. For this kind of shipment, the major federal agency requirements originate from the Department of Transportation (DOT), Nuclear Regulatory Commission (NRC), and the Department of Energy (DOE). The most relevant requirements and regulations are listed in Section 2, Applicable Documents and References.

Several departments have responsibilities for performing functions that assure radioactive material shipments conform to federal and state requirements. Radiation Safety personnel interface and cooperate with these other units and perform a vital role in the shipment preparation process. The scope of this Radiation Safety procedure is limited to include:

1. The tasks of defining the isotopes and radioactivity for each package in a shipment

2. The tasks of performing surveys to assess and document the radiation and non-fixed (removable) surface contamination status of packages and transport vehicles, and

3. The documentation tasks associated with defining the radiological status of the shipment preparation

These procedures emphasize the acquisition, development, and preparation of data for radioactive waste shipments originating from SSFL decommissioning projects. These procedures meet the requirements of the Waste Disposal Sites listed in Appendix A. In addition, data generated by the processes described in this procedure are used to prepare shipments for transportation under DOT regulations.

A number of appendices are included to provide reference material and perspective information pertinent to radiological characterization tasks.

2.0 APPLICABLE DOCUMENTS AND REFERENCES

2.1 REGULATORY DOCUMENTS

2.1.1 UNITED STATES DEPARTMENT OF ENERGY (USDOE)

10 CFR 835, *Occupational Radiation Protection.*

2.1.2 UNITED STATES NUCLEAR REGULATORY COMMISSION (USNRC)

10 CFR 20, *Standards for Protection Against Radiation.*

10 CFR 61, *Management and Disposal of Low Level Wastes by Shallow Land Burial and Other Alternative Methods*.

10 CFR 71, *Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions*.

USNRC IEN 90-35: *Transportation of Type A Quantities of Nonfissile Radioactive Materials.*

USNRC IEN 91-35: *Labeling Requirements for Transporting Multihazard Radioactive Materials*.

USNRC IEN 80-32: *Clarification of Certain Requirements for Exclusive-Use Shipments of Radioactive Materials*.

USNRC, Reg. Guide 7.3, *Procedures for Picking Up and Receiving Packages of Radioactive Material*.

2.1.3 STATE OF CALIFORNIA

California Administrative Code, Title 17, Subchapter 4, Group 4, *Transportation of Radioactive Material.*

2.1.4 UNITED STATES DEPARTMENT OF TRANSPORTATION (USDOT)

49 CFR 173, Subpart I, *Radioactive Materials*.

44 CFR 38690, *Transportation of Radioactive Materials: Memorandum of Understanding*.

2.2 BOEING PROCEDURES / SUPPORTING DOCUMENTS

[Online procedures are designated with a "✓"]

✓SOP C-404, *Shipping Radioactive Materials*.

✓SOP C-405, *Instructions for Receipt of Shipments Containing Radioactive Materials.*

✓Supporting Document, (R21 WM) EID-04492, *Procedure for Measuring, Packaging, and Shipping Non-SNM Radioactively Contaminated Protective Clothing*

✓Supporting Document, (R21 RS) RS-00012, *Methods and Procedures for Radiological Monitoring*.

✓Supporting Document (R21 WM) EID-04493, *Procedure for Packaging and Shipping of Radioactive Materials*.

✓Supporting Document (R21 RS) RS-00017, *Quality Control And General Operating Procedure Using Canberra Multichannel Analyzers.*

Supporting Document, N001SRR140119, *Analysis of Hazardous Waste for Radioactivity*.

Supporting Document N001OP160003, *Packaging and Shipment of Radioactive Waste Low Specific Activity - to Westinghouse Hanford Disposal Site*.

Supporting Document 1730P000007, *Packaging and Shipment of Radioactive Waste Low Specific Activity - to DOE Nevada Test Site*.

Supporting Document N001OP240042, *Procedure for Measuring, Packaging, and Shipping Non-SNM Radioactively Contaminated Protective Clothing*.

✓RS-00033, *Boeing Policies Regarding SCO and LSA Wastes.*

2.3 OTHER REFERENCES

NTSWAC, *Nevada Test Site Defense Waste Acceptance Criteria.*

WHC-EP-00063-4, *Hanford Site Solid Waste Acceptance Criteria*.

Envirocare of Utah document, *Material Acceptance Process Manual*.

3.0 DEFINITIONS

HEPA - High Efficiency Particulate Air filtration unit (99.99% efficient).

HP - Health Physicist; A representative of D641, Radiation Safety .

Materials - Radioactive waste or radioactive items that constitute the contents of a package.

Package - The packaging, together with it's contents, as presented for transport.

PPE - Personnel Protective Equipment.

QA - The Quality Assurance department or one of it's representatives.

RMHF - The Radioactive Materials Handling Facility, located in Area IV, Santa Susana Field Lab.

Radiation Safety – The function of the SHEA Health, Safety and Radiation Services group that deals with ionizing radiation and radioactive materials.

RS - Radiation Safety

RSO - The Boeing-SSFL Radiation Safety Officer.

SHEA Transportation Specialist - The SHEA individual responsible for assuring that off-site shipments of radioactive materials meet Department of Transportation regulations.

4.0 RESPONSIBILITIES

4.1 GENERAL

Several departments share the process necessary to receive or ship radioactive materials. SOP C-404 describes the various departmental responsibilities.

4.2 RADIATION SAFETY

Radiation Safety personnel are responsible for:

4.2.1 Performing necessary surveys and analyses to define isotopes present and curie content of each package for outgoing shipments (“characterization”)

4.2.2 Performing surveys of materials, packages/containers and vehicles to define radiation levels

4.2.3 Surveying shipments for removable (non-fixed) contamination.

4.2.4 Entering data, information, and signatures on the Radioactive Waste Packaging Lot Follower and Procedure Verification Form (Form 652-A)

4.2.5 Reviewing survey data, and assuring that all radiation safety requirements are met prior to shipping and approving shipment by signing Form 710-S-1, “Radioactive Material Shipping Instructions”

4.3 SHIPPING DEPARTMENTS

The department originating or receiving the shipment is responsible to notify Radiation Safety staff of an incoming empty transport vehicle or of an incoming or outgoing radioactive waste or radioactive materials shipment.

5.0 SHIPMENT CHARACTERIZATION

5.1 GENERAL

The DOT regulation specifies packaging and shipment mode alternatives and requirements based upon the radioisotopes to be shipped, their form and radioactivity. Consequently, radiological characterization data are necessary for preparation of a shipment containing radioactive materials.

Radiation Safety personnel are responsible for providing information on the Radioactive Material Tag (Form 708-R-1) regarding contents, estimated activity in curie units, radiation intensities, surface contamination levels and type of activity. This tag must be completed for each package of a multipackage shipment.

5.2 IDENTIFICATION OF RADIOISOTOPES

The principal radioisotopes in the package shall be identified and the information entered on the tag. The isotope information must be reliable and may be obtained from the user, from sampling and subsequent isotopic analyses, from sampling and gamma spectroscopy, or from reliable process knowledge. When process knowledge is used, the sources of that information must be reliable and well documented. As discussed in Appendix E.1, information more than 10 years old should be verified by current sampling.

Characterization can be accomplished by ratioing the activity of certain isotopes to an isotope that is more easily characterized (e.g., Cs-137 or Co-60). When this approach is used, the basis for the ratios should be documented. Determination of the ratio of alpha to beta activity and reference to isotope decay tables may be helpful to confirm or suggest the isotope(s) present.

Where reliable identification of the principal isotope(s) is not feasible, then in consultation with the shipper, Radiation Safety should perform an evaluation to devise satisfactory alternative approaches.

For purposes of transportation, 49 CFR 173.433, presents a listing of radionuclides and permissible activities per package as well as instructions regarding treatment of mixtures of isotopes and cases where either isotope or activity cannot be determined

5.3 ESTIMATED ACTIVITY

The total radioactivity of the isotopes in each package shall be estimated based on the best information reasonably obtainable. The activity is expressed in terms of curies, millicuries or microcuries. Alpha and beta activities are added to arrive at the total activity. Where the isotopic concentration and the masses of materials in the package are known, the total activity may be derived by multiplying the activity concentration (mCi/g) of each isotope by the mass of the associated material in the package to obtain the total activity of each isotope. The mass of the container and any non-radioactive material added to the container to provide shielding shall not be used to calculate specific activities.

5.4 SURFACE CONTAMINATED ITEMS

The decision regarding whether radioactive contamination is confined to the outer surfaces of items or distributed below the surface shall be made by Radiation Safety personnel, assisted by the operations organization. The decision may be based on the nature of the material, history of use, reliable process information or subsurface sampling.

In cases of surface contaminated items, reasonable engineering estimates shall be made. In those cases, an approximation must be made of the total area of affected surfaces in the package in terms of square centimeters. This area is then divided by 100 cm^2 and that result is multiplied by the typical dpm per 100 cm^2 contamination level to obtain a total activity in dpm for the contents. The total dpm is then divided by 2.22×10^9 to obtain millilcuries.

5.5 DISTRIBUTED CONTAMINATION

For items containing neutron activation products or radioactive contamination distributed in a material, two general approaches may be used to estimate the total activity.

- For a homogeneous mixture, a representative sample may be obtained and analyzed by radiochemical or radiation spectroscopy methods to obtain the concentrations and isotopic ratios to be extrapolated to the total mass of the shipment[1].

- Where homogeneity is not present, activity can be determined by measuring the penetrating radiation dose rate at a distance from the waste containers and, using shielding calculations and isotopic ratio data, back-calculate the total activity. (This method is described in Appendix E.3.1)

Further data, discussion, rules-of-thumb, examples of methods for use in estimating the activity of a waste package, and instructions for a software tool for these calculations are to be found in appendices B, E, and F.

5.6 WASTE SAMPLING PROCEDURES

NOTE: The following sampling procedures apply only to samples to be analyzed by Radiation Safety personnel, or by a subcontract laboratory. Other departments (e.g., DOE Site Restoration) who take samples not intended for Radiation Safety analysis shall implement their own sampling procedures.

NOTE: It is recognized that many samples are obtained by Radiation Safety personnel, who may in turn do their own analyses. These persons also must follow these waste sampling procedures.

NOTE: Appendix B of Reference 3 provides useful information and guidance for establishing sampling strategies.

---

1 Analytical procedures may be found in (R21 RS) RS-00017, "Quality Control and General Operating, Procedure for Gamma Spectroscopy Using Canberra Multichannel Analyzers.

Samples taken for chemical or other non-radiological constituents may have special requirements. If samples submitted for radiometric analysis are subsequently to be analyzed for chemical or hazardous material constituents, then special requirements should be stipulated by the submitting organizations or by Environmental Protection; e.g., a 14-day holding time limit for volatile organic compounds. Special requirements shall be communicated to Radiation Safety upon transfer of the sample.

When radioactive materials are dispersed within a waste matrix, the facility Radiation Safety representative shall decide whether or not sampling of the material is necessary in order to identify and/or quantify the constituents. If new or unusual forms of waste are under consideration, the Radiation Safety representative shall consult with or, at least, inform Radiation Safety management regarding the sampling decision. The following procedures and methods shall be followed if Radiation Safety technical staff deem that waste sampling is necessary.

5.7 SAMPLING METHODS:

The method of sampling shall depend upon the physical state of the waste as follows:

- <u>Liquid Mixtures</u>
    - For homogeneous liquid mixtures, take at least one sample per container of the liquid. If the container size exceeds 55 gallons, more than one sample should be taken; the number will be dependent upon the sampler's judgment. The size of the sample should be about 1 liter if practical.
    - For heterogeneous liquid mixtures (i.e., the contents readily separate into different constituents), two approaches may be utilized:
        - Permit the constituents to separate, then sample each constituent separately (obtaining at least one sample per layer); or,
        - Thoroughly mix the material, and obtain a sample before phase separation begins.
- <u>Solid Materials</u>
    - Sampling regimens should be based on statistical models to ensure the sample set reasonably represents the radiological condition of the material.
    - Unless sampling methods have previously been developed for the specific waste form, a sampling plan should be developed in consultation with Radiation Safety management and technical staff.
    - Samples should be at least 500 grams if practical.

- Surface Contaminated Materials

  - A standard smear technique shall be used with each smear covering an approximate area of 300 cm^2 and sufficient smears taken in the most appropriate locations to yield a representative assessment of the removable contamination levels.

  - The Radiation Safety representative will determine the number of smears to be made, based on the nature of the container, where it was loaded and stored, and how it was handled.

  - Sufficient smears will be made in order to sample the entire package or material surface area and specifically the areas most likely to be contaminated. As a minimum, packages of 55-gallon drum size or larger should be smeared in at least three (3) locations; 1 for sides, 1 for top, and 1 for bottom. Smears should be made such that handling and sealing surfaces and the bottom of the package or container are always smeared.

  - Waste materials characterization may be based upon smear surveys of the point of origin (e.g., building surveys prior to decontamination and decommissioning) or by direct smear survey of the materials.

  - Smears may be characterized for isotopes as described for water and solid samples.

  - The wiping efficiency of a smear shall be assumed to be 0.1, unless efficiencies have been empirically determined. (See 49 CFR 173.443)

5.7.1 SAMPLE CONTAINERS, LABELS AND SEALS

Use plastic bottles with screw caps to contain liquid samples. Use rigid containers to hold solid samples (plastic bags may be used only if the sample is to be stored for a short period of time or when placed in rigid outer containers to prevent bag breakage).

Each sample container shall be labeled in the field by the originator using a label which contains as a minimum, the information shown in Figure 1.

Labels shall be obtained from Radiation Safety representatives or from radiometric analysis laboratory personnel.

_____
SAMPLE NUMBER

_____
NAME OF COLLECTOR (PRINT)

_____
DATE AND TIME OF COLLECTION

_____
SAMPLE DESCRIPTION

_____
SAMPLE LOCATION

Figure 1. Radioactive Waste Sample Label (Specimen)

After filling, a self-adhesive label, bearing only the sample number, shall also be applied to the sample container to serve as a seal. The seal shall be applied so that it will be torn when the container is opened.

Radiation Safety personnel shall use the following prenumbering system for issue of labels:

Category	**Year**	**Sequence Number**
024	01	0001
059	01	0001
RMF	01	0001
ENV	01	0001
OTHER _____ (fill in blank)	01	0001

As an example, an identification number could be 024-01-0007. Sample collectors shall use the same numbering system for field log book notation. Radiation Safety shall use the same numbering for laboratory records.

Radiation Safety representatives or laboratory personnel shall refuse to accept unlabeled or unsealed samples, or samples with a broken seal.

After analysis, the samples that were found free of radioactive or hazardous materials may be discarded and the containers cleaned for reuse. Samples found to be radioactive above the release levels shall be properly labeled and returned to the originator for proper disposition.

5.7.2 RADIATION SAFETY FIELD SAMPLE LOGBOOKS

Bound, page-numbered logbooks shall be used by Radiation Safety personnel to record sampling information and data for samples collected by Radiation Safety personnel. Facility logbooks may be used, or dedicated logbooks established. (Other organizations will have their own logbook procedures.)

The following procedures shall be followed by Radiation Safety for logbook sampling entries:

- Make entries in ink, date, and sign
- Describe item sampled, location of sample(s), and purpose of sampling
- Describe sampling method
- Describe number and approximate volume of samples
- Enter sample identification number(s) (same number as on sample label)
- Describe to whom (facility, person, or organization) sample will be submitted for analysis

Enter any other significant information or comments pertinent to the sampling.

Logbooks shall be clearly identified on the cover as to their purpose. They should be kept in a secure (locked, if possible) location when not in use. It is strongly suggested that Engineering logbooks be procured from the Technical Library to serve as field logbooks.

When logbooks are filled, they shall be submitted to Radiation Safety management for review and for permanent storage.

5.7.3 SAMPLE STORAGE AND TRANSPORT

Sample storage in the field ordinarily should not exceed a few days (less than a week). Transport to the Radiation Safety laboratory or counting station shall be such that integrity of the package, label, and contents is preserved. The method of transport is not critical unless Radiation Safety personnel have imposed contamination control requirements.

5.7.4 ANALYSIS REQUEST AND CHAIN OF CUSTODY

The collector or sample deliverer shall initiate a combination Radiometric Analysis Request and Chain-of-Custody Record form to accompany the sample to the Radiation Safety laboratory. The form serves to transmit information from the sample collector's field log book and to-document the persons who have had custody of the sample and were responsible for assuring sample identity and integrity.

The form shall be provided by Radiation Safety and is shown in Figure 2. The form content is explained by the items on the form.

Figure 2. Combination Analysis Request and Chain-of-Custody Form (Form X31849)

BOEING

The Boeing Company
Santa Susana Field Laboratory
5800 Woolsey Canyon Road
Canoga Park, CA 91304-1148

SHEA
Radiation Safety Department

RADIOMETRIC ANALYSIS REQUEST AND CHAIN-OF-CUSTODY RECORD

Sample		Collection	Analysis Requested								
			Gross		γ			Isotopic			
Number	Type	Date	α	β	Spec	^{3}H	^{90}Sr	U	Th	Pu	Other
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	
			☐	☐	☐	☐	☐	☐	☐	☐	

Description of Item Sampled, Location, and Purpose of Sample: _____

Sample Collector: _____

(Print and Sign) Date: _____

RECORD OF SAMPLE CUSTODY

Names of Custodians (Print and Sign)	Received		Transferred	
	Date	Time	Date	Time

Results Reported (To and How): _____

Sample Disposition (Describe): _____

X31849 REV (06OCT05)

[Current Form: http://leos.web.boeing.com/cgi-bin/formtrackleos/track3.pl?x31849.doc]

Radiation Safety representatives or laboratory personnel shall refuse to accept samples for analysis if the origional of this form does not accompany the sample(s).

The completed form shall be kept in the Radiation Safety laboratory files.

5.8 RADIOACTIVE WASTE LOT FOLLOWER

The Radiation Safety representative responsible for the waste radioactivity estimates for specific items going into a waste container is responsible for entering radiometric data and analysis information on the waste package lot follower (Form 652-A). The radioisotopes are to be identified as well as the analysis method.

For analysis method, enter the number of this document (i.e., RS-00011), and the appropriate appendix method used for the estimate. If another method is used, identify the applicable method. When appropriate, enter the sample number.

With the exception of naturally occurring isotopes other than K-40, any isotope identified or considered to be present in a waste container should be listed on the lot follower. Explanatory information regarding characterization of waste containers (e.g., identification of daughter products, etc.) may also be listed on the lot follower. Such explanatory information may be needed for the correct interpretation of the characterization data during the evaluation process as the package is cleared for shipment to the waste site.

A summary of the appendix methods follows:

Appendix Number	**Method of Estimate**
E-1	Isotopes by historical or process knowledge
E-2	Possible isotopes by facility identity
E-3.1	Activity by MicroShield computer program
E-3.2	Activity by rule-of-thumb for laundry and filters
E-3.3	Activity by gamma spectrometry of samples
E-3.4	Activity by surface contamination measurements
E-3.5	Activity by gamma spectrometry for constituents in a package
F	Instructions for Software Calculation Tool for Waste Container Characterization

5.9 RADIATION INTENSITIES AND CONTAMINATION LEVELS FOR PACKAGES AND CONTAINERS

Measurements of radiation intensities (levels) shall be made with the RO-2 ion chamber, a micro-R meter or equivalent instruments. DOT regulations require determinations of radiation levels for packages. The regulation specifies these limits in terms of mSv per hour and mrem per hour in order to be inclusive of neutron exposure when present. In most cases, Boeing does not possess materials that emit neutrons, therefore, for application and documentation under this procedure, instrument measurements in terms of mR(oentgen)/h are used and are equivalent to mrem/h.

NOTE: Site specific contamination and radiation limits may apply to waste shipments bound for radioactive waste disposal sites (see EID-04482, section 27.15, *Packaging and Shipment of Radioactive Waste*).

If a receiving facility's contamination limits are more restrictive than the DOT requirements, the Boeing department coordinating the shipment will advise Radiation Safety of the receiver's requirements and the receiver's limits will be applied.

Several sites utilize the contamination limits of 10 CFR 835, Appendix D. For convenience, these limits are reproduced in Appendix C.

For the purposes of commercial transportation, the maximum values for removable contamination and for radiation dose rates shall be less than the limits provided in Appendix C.

For transporation between Canoga Park sites (i.e., Canoga Park, DeSoto, and SSFL), the exterior packaging surfaces of items transported shall be surveyed and limited to removable contamination levels less than 100 dpm/100 cm^2 for beta/gamma activity and 20 dpm/100 cm^2 for alpha activity.

5.10 TYPE OF ACTIVITY

The Radioactivity Material Tag (708R; Appendix D) also requires entry of information regarding the type of radiations emitted by the radioactive material. Based on the isotopes involved, an entry shall be made for either alpha, beta, gamma, or combination of these.

The removable contamination levels and the radiation dose rates determined in step 5.9 for the outside of each completed shipping package shall be entered on the Radioactive Material Tag.

6.0 SURVEYS OF INCOMING AND DEPARTING LOADED, EXCLUSIVE USE VEHICLES

Radiation Safety (HP) personnel shall survey loaded radioactive material shipments on exclusive use vehicles, both incoming and before their departure from Boeing. The Radiation Safety representative for the facility responsible for the shipment shall obtain the appropriate survey data per techniques detailed in paragraphs 4.3 and 4.4. These data shall be entered on the Radiation Survey Report (shown in Appendix D) and, as appropriate, on the "Radioactive Material Transportation Departure Survey Form," (shown in Appendix D)

Incoming exclusive use vehicle shipments shall be surveyed before unloading with the same requirements that are applied to departing shipments. The purpose is to verify compliance with requirements.

Vehicle surveys are not required for normal or non-exclusive use vehicles.

Dose rate limits for exclusive use vehicles are provided in Appendix C.

NOTE: Site specific contamination and radiation limits may apply to waste shipments bound for radioactive waste disposal sites (see EID-04482, section 27.16, *Packaging and Shipment of Radioactive Waste*).

If a receiving facility's contamination limits are more restrictive than the DOT requirements, the Boeing department coordinating the shipment will advise Radiation Safety of the receiver's requirements and the receiver's limits will be applied.

Several sites utilize the contamination limits of 10 CFR 835, Appendix D. For convenience, these limits are reproduced in Appendix C.

7.0 SURVEYS OF EMPTY EXCLUSIVE USE VEHICLES

Radiological surveys are required for empty “exclusive use” vehicles coming into Boeing for loading of radioactive material shipments or for “exclusive use” vehicles that have delivered such shipments. The departments responsible for receiving (or coordinating) such shipments are required to notify Radiation Safety of the need and schedule for the survey activity.

The DOT regulation requires that “exclusive use” vehicles shall be surveyed after each use: (For empty vehicles coming into Boeing to load shipments, the Boeing survey before loading is intended to confirm the vehicle still meets requirements.)

Dose rate and contamination limits for EMPTY exclusive use vehicles are provided in Appendix C. Surveys should consist of direct measurements for beta and alpha emitters, and a representative smear survey for removable radioactive materials. If no activity is detected, then the survey result may be recorded as less than the applicable limit for the associated isotope and activity.

8.0 SURVEYS OF INCOMING PACKAGES CONTAINING RADIOACTIVE MATERIALS

The flowpath for incoming packages of radioactive materials is described in SOP C-405, *Instructions for Receipt of Shipments Containing Radioactive Materials.*

Incoming packages labeled as containing radioactive materials must be surveyed by Radiation Safety personnel as soon as practical after receipt. These packages are received at Boeing by Receiving. Whichever group receives the package will notify the Radiation Safety Officer (RSO) or a designated alternate in Radiation Safety. As soon as practical, Radiation Safety personnel will perform a radiation and contamination survey of the package and record other information that may be pertinent to a Use Authorization or the sealed source inventory[4].

Radiation Safety will obtain dose rate and contamination data to confirm the package met DOT requirements and to establish precautionary measures if necessary.

NOTE: Levels exceeding

- 3.6 Bq/100 cm^2 (220 dpm/100 cm^2) alpha; or,
- 36 Bq/100 cm^2 (2,200 dpm/100 cm^2) beta,

may require official notifications to the delivery service and regulatory agencies. Notification shall be made to Traffic, the Radiation Safety manager and the prospective user. These persons will prepare an incident report and make other notifications based upon the levels found.

NOTE: If contamination levels exceed:

- 20 dpm/100 cm^2 of alpha activity; or,
- 100 dpm/100 cm^2) of beta activity

An evaluation shall be made to determine if contamination control efforts (e.g., posting and access control to contaminated areas, recall of the delivery vehicle, etc.) are required. If required, area decontamination should be performed in a timely manner. Guidance regarding such posting and cleanup is provided in SM-40.402, *Radiological Standards*, Table 2-2.

If all administrative requirements have been satisfied (e.g., Use Authorization has been issued, etc.), and if the radiation and contamination levels are found to be within the DOT and Boeing Radiation Safety requirements, the material may be released to the user.

---

4 The NRC requires this survey be performed within 3 hours of receipt on a normal workday or within 13 hours of receipt if delivery is after normal work hours.

9.0 DATA QUALIFICATION AND REVIEW

9.1 Spreadsheets, computer programs or other calculation aids may be developed to assist in the calculation and documentation of the waste characterization process.

9.2 Spreadsheets, programs, or other calculational aids shall be verified to accurately calculate waste parameter data prior to first use for waste characterization. The verification shall be done using a standardized test data and outcome set developed for the purpose using an independent calculation technique (such as pen and paper, alternate spreadsheet, etc.).

The independent verification shall be documented in writing describing the verification in enough detail that the verification calculations could be reproduced at a later date having only the verification report at hand. A copy of this verification and a copy of the software program or aid (e.g., a CD-ROM with a copy of the program) shall be entered into the SHEA filing system for future reference.

A Validation and Verification ("V&V") report prepared by a commercial entitity in support of a software application may be utilized for this purpose. The acceptance of the V&V report should be documented, and a copy of the program and report placed into the SHEA filing system.

9.3 Spreadsheet, programs, or other calculation aids should be protected in some way to prevent inadvertant changes being introduced into a verified calculation program. This may be accomplished by protected fields, "frozen" versions of compiled code, or other routine software control techniques.

9.3 A verification shall be conducted by an independent reviewer of 1 out of every 20 waste characterization calculations performed. The verification shall be documented in a log maintained for that purpose. The verification log may be electronic in nature, as long as a hardcopy of the log can be produced upon request.

10.0 FORMS AND RECORDS

The data obtained from surveys required by this procedure shall be entered on the appropriate forms and copies distributed as shown below. (Specimens of the forms are shown in the appendices.)

Form	Data	Distribution
Radioactive Material Tag (Form 708-R)	Contents (Isotopes or TBD if determination is in progress) Activity (TBD if determination is in progress) Radiation Levels Contamination Activity Type	Attached to item or container
Radiation Survey Report (Form 732-A)	Survey Results	Original to Radiation Safety Copies: SHEA Trans. Specialist RMHF ETEC QA Facility HP
Radioactive Material Transportation Departure (Form X31848)	Summarizes radiological survey results for transport vehicle and lists applicable DOT limits.	Original to Radiation Safety Copies: SHEA Trans. Specialist RMHF ETEC QA Facility HP
Radioactive Container Survey (Form X31850)	Documents radiological surveys of transportation containers for radioactive shipments	Original to Radiation Safety Copies: SHEA Trans. Specialist RMHF ETEC QA Facility HP
Radioactive Material Shipping Instructions (Form 710-S-1)	Requires Radiation Safety approval signature	Original to Shipping Data File

Form	Data	Distribution
Calculational Aid Verification	Documents the verification of software and calculation aids used in waste characterization calculations	Electronic file, with hardcopy produced upon request. Link To Logsheet: Waste Characterization Review Logsheet.xls

APPENDIX A. WASTE SITE ACCEPTANCE CRITERIA DOCUMENTS

SITE	DOCUMENTS
Nevada Test Site	DOE/NV-325, *Nevada Test Site Waste Acceptance Criteria* (NTSWACS)
Envirocare of Utah	*Bulk Waste Disposal and Treatment Facilities Waste Acceptance Criteria (Document not numbered)* *Containerized Waste Facility Waste Acceptance Criteria (TSC-2.0)*
Hanford Site	WHC-EP-00063-4, *Hanford Site Solid Waste Acceptance Criteria*.

APPENDIX B. FORMULAS, SPECIFIC ACTIVITIES, AND UNIT CONVERSTIONS[5]

Useful Formulas

- Inverse square formula for point source:

$$I_2 = I_1 \frac{(D_1)^2}{(D_2)^2}$$

$$I_1 = I_2 \frac{(D_2)^2}{(D_1)^2}$$

I_1 = Radiation exposure at D_1

I_2 = Radiation exposure at D_2

D_1 = Distance at point closer to source

D_2 = Distance further from source

- Approximate exposure rate in AIR from a gamma emitting point source:

R / h at 1 foot $\cong$ 6CEn

mr / h / mCi at 1 meter $\cong$ 0.5nE

$$\text{mCi} \cong \frac{\text{mR / h at 1 meter}}{0.5\text{nE}}$$

C = curies

E = photon energy in MeV

n = number of photons emitted for each nuclear transformation (disintegration)

Reminder: Co-60 emits two photons per transformation; one of 1.33 MeV and one of 1.17 MeV.

- Specific activity of a pure radioisotope:

General: $$\text{SpA} = \frac{(\ln 2)(N)}{T_{1/2}}$$

N = number of atoms per unit mass

$T_{1/2}$ = half life

For $T_{1/2}$ in minutes:

$$\text{SpA in curies per gram} = \frac{1.88 \times 10^{11}}{(T_{1/2}\ \text{minutes})\ (\text{atomic mass})}$$

---

[5] Data source is "Radiological Health Handbook," U.S. Public Health Service, 10/70

For $T_{1/2}$ in years:

$$\text{SpA in curies per gram} = \frac{3.578 \times 10^5}{(T_{1/2}\ \text{years})\ (\text{atomic mass})}$$

Example: SpA for Co-60

$$\text{SpA} = \frac{3.578 \times 10^5}{(5.26\ \text{y})(60)} = 1.134 \times 10^3\ \text{Ci / g}$$

Specific Activities of Some Radioisotopes

Radionuclide	Symbol	Half Life	Curies per gram
Hydrogen - 3	H-3	12.3 y	9.64×10^3
Carbon - 14	C-14	5.73×10^3 y	4.46
Sodium - 24	Na-24	15.0 h	8.71×10^6
Phosphorus - 32	P-32	14.3 d	2.85×10^5
Potassium - 40	K-40	1.28×10^9 y	6.99×10^{-6}
Iron - 55	Fe-55	2.73 y	2.38×10^3
Nickel - 59	Ni-59	7.6×10^4 y	7.98×10^{-2}
Cobalt - 60	Co-60	5.27 y	1.13×10^3
Nickel - 63	Ni-63	100 y	56.8
Strontium - 90	Sr-90	29.1 y	136
Technetium - 99 m	Tc-99 m	6.01 h	5.27×10^6
Ruthenium - 106	Ru-106	1.02 y	3.31×10^3
Iodine - 131	I-131	8.04 d	1.24×10^3
Cesium - 137	Cs-137	30.17 y	86.5
Promethium - 147	Pm-147	2.62 y	929
Europium - 152	Eu-152	13.5 y	174
Europium - 154	Eu-154	8.59 y	270
Iridium - 192	Ir-192	73.8 d	9.22×10^3
Polonium - 210	Po-210	138.4 d	4.49×10^3
Radium - 226	Ra-226	1602 y	0.988
Uranium - 233	U-233	1.592×10^5y	9.65×10^{-3}
Thorium - 232	Th-232	1.41×10^{10} y	1.09×10^{-7}
Uranium - 234	U-234	2.46×10^5 y	6.21×10^{-3}
Uranium - 235	U-235	7.04×10^8 y	2.16 x 10-6
Uranium - 238	U-238	4.47×10^9 y	3.36×10^{-7}
Plutonium - 239	Pu-23 9	2.41×10^4 y	6.21×10^{-2}
Plutonium - 240	Pu-240	6.56×10^3 y	0.228
Plutonium - 241	Pu-241	14.4 y	103
Americium - 241	Am-241	432.7 y	3.42

Unit Conversions

Radiological Activity:

Ci	=	2.22×10^{12} dis/m
mCi	=	2:22 x 10^{9} dis/m
μCi	=	2.22×10^{6} dis/m
pCi	=	2.22 dis/m
Bq	=	27pCi
Sv	=	100 mR

Volume:

1 m^3	=	10^6 cm^3
1 m^3	=	10^3 liters
1m^3	=	35.314 ft^3
1 m^3	=	2.642×10^2 gallons
1 ft^3	=	28.32 liters
1 ft^3	=	7.481 gallons

Density:

1 g/cm^3	=	62.43 lb/ft^3

Mass

1 g	=	2.205×10^{-3} lb
1 lb	=	453.6g
1 oz	=	28.35 g
1 kg	=	2.205 lb

Area:

1 ft^2	=	929 cm^2
1 ft^2	=	9.29×10^{-2} m^2
1 m^2	=	1×10^4 cm^2
1 in^2	=	6.45 cm^2
100 cm^2	=	15.5 in^2 or 0.11 ft^2

Transmission Factors for Attenuation by Concrete: Multiply the dose rate before or without shielding by the transmission factor to determine the resultant dose rate after accounting for shielding.

$$D_{shielded} = D_0 B$$

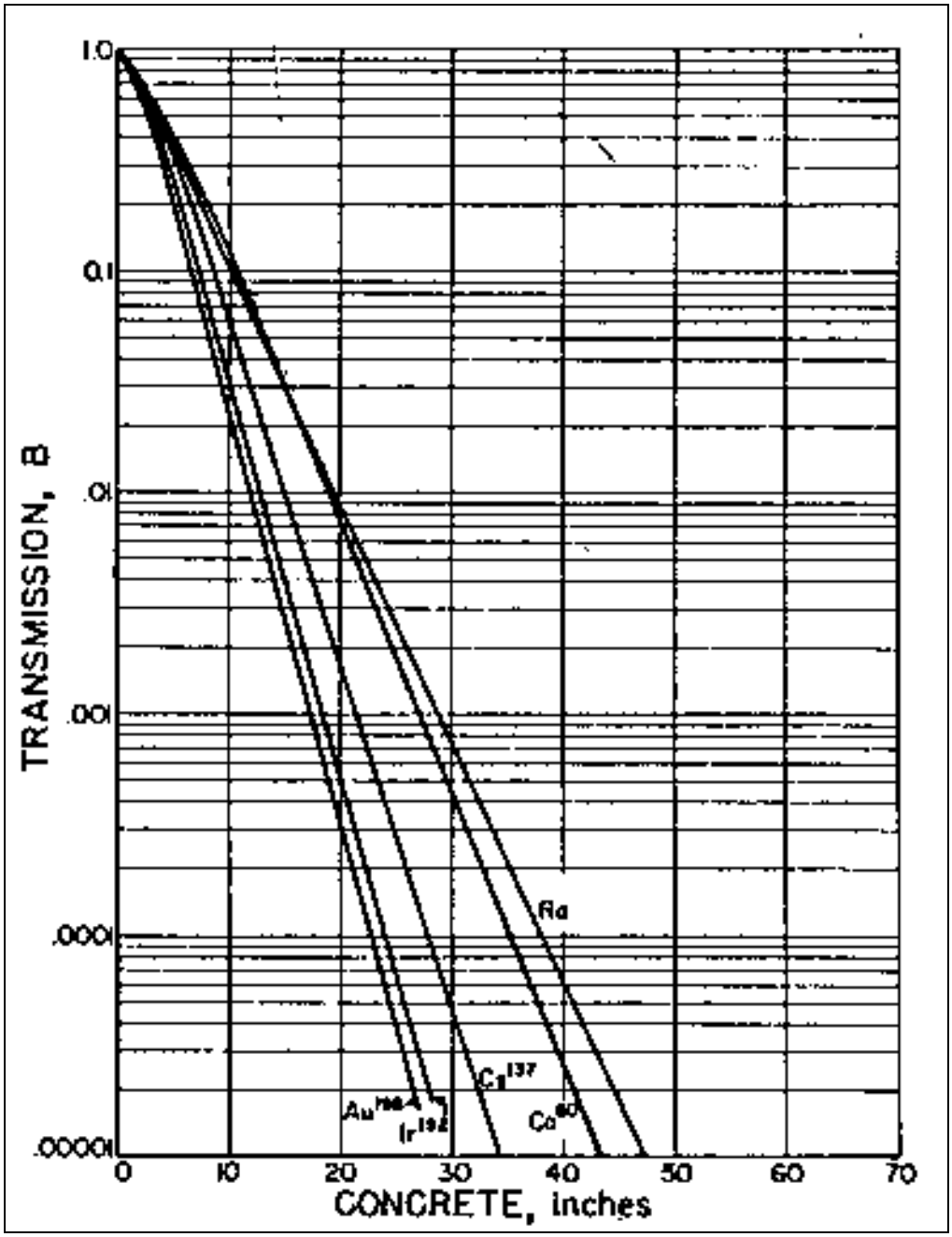


Figure B-1. Transmission Through Concrete (Density 147 lb/ft^2) of Gamma Rays from Radium; Cobalt 60, Cesium 137, Gold 198; Iridium 192

From Rad Health Handbook (1970): p. 149

<u>Transmission Factors for Attenuation by Iron:</u> Multiply the dose rate before or without shielding by the transmission factor to determine the resultant dose rate after accounting for shielding.

$$D_{shielded} = D_0 B$$

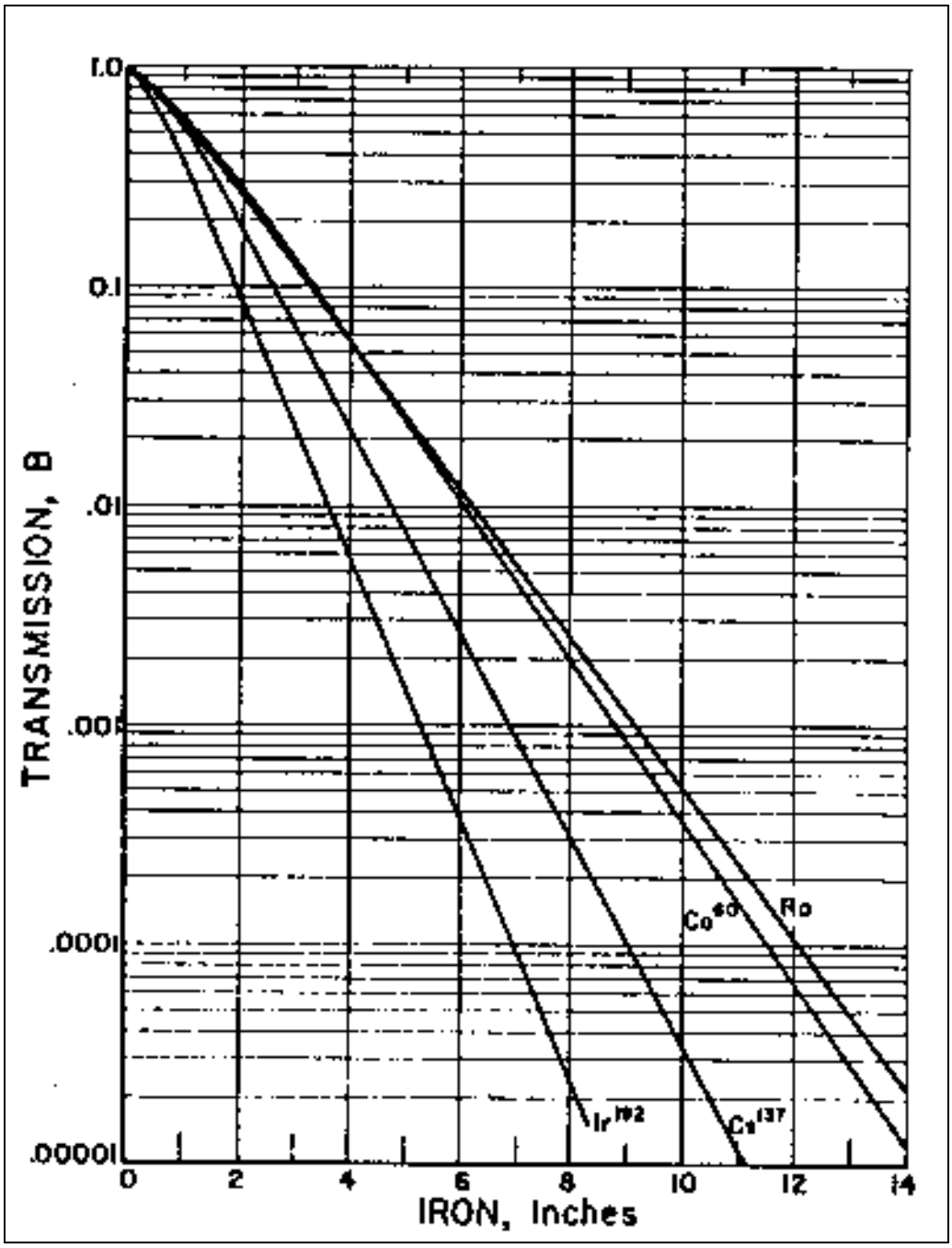


Figure B-2. Transmission Through Iron of Gamma Rays from Radium; Cobalt 60, Cesium 137; Iridium 192

From Rad Health Handbook (1970): p. 149

APPENDIX C. LIMITS AND CONDITIONS APPLICABLE TO SHIPMENTS

The following table lists radiation **dose rate limits** applicable to shipment of radioactive materials. The definitive source for these limits is 49 CFR 173.441.

Transportation Dose Rate Limits		
Type of Shipment	**Measurement Location**	**Limit**
Individual Package in an Exclusive Use Shipment		
• Shipment with exclusive use of closed vehicle • Package in fixed position in vehicle during travel • No unloading or loading of vehicle between beginning and end of transport	External surface of package at any point	10 mSv/h (1000 mrem/h)
Standard Shipments		
• Package conforms to activity limits and DOT container specifications (see Reference 6, paragraphs 173.411 through 173.435)	External surface of package at any point	2 mSv/h (200 mrem/h)
• Transport index[2] less than 10	At one meter from surface of package	0.1 mSv/h (10 mrem/h)
"Limited Quantity" of Radioactive Material		
• Limited[3] quantities of radioactive material as defined in the DOT regulation (Reference 6, 173.421) • Inner package marked RADIOACTIVE • Exempt from specification packaging and labeling	External surface of package at any point	0.005 mSv/h (0.5 mrem/h)

Transportation Dose Rate Limits		
Type of Shipment	**Measurement Location**	**Limit**
Empty Package Which Previously Contained Radioactive Material, and		
• Is in good condition, securely closed • Internal contamination does not exceed 100 times the limits shown in Section 4.4 • Former outside labels removed, if removable, replaced with DOT EMPTY label • External surface contamination below limits in Section 4.4	External surface of package, at any point	0.005 mSv/h (0.5 mrem/h)
Quantity Exempt From DOT Radioactive Material Shipping Requirements		
• Exempt from DOT radioactive material requirements when the activity concentration *and* the total activity in the consignment are equal to or less than the values specified in the table in 49 CFR 173.436 or values derived according to the instructions in 49 CFR 173.433.	N/A	N/A

[2] Transport index = mSv/h x 100 (mrem/h x 1) at 1 meter or for fissile material, can be a number based on allowable number of packages (Reference 6)

[3] Limited quantities are defined in 49 CFR 173.421 for specific forms and isotopes by reference to the A_1, A_2 table in 173.435. The definition varies for manufactured articles and special material forms and by radioisotope. The limit for typical Boeing isotopes are (for solids): Cs137 0.005 TBq (135 mCi), Co60 0.004 TBq (108 mCi), Sr90 0.001 TBq (27 mCi).

The following table lists limits for **removable contamination** applicable to shipment of radioactive materials. The definitive source for these limits is 49 CFR 173.443.

Transportation Removable Contamination Limits		
Type of Radioactive Emission	**Limit**	**Derived Limit**
Beta-gamma	0.36[2] Bq/cm^2 (22 dpm/cm^2)	2,200 dpm/100 cm^2
Alpha	0.036 Bq/cm^2 (2.2 dpm/cm^2)	220 dpm/100 cm^2

For an exclusive use vehicle shipment, the removable contamination levels at the beginning of the shipment shall be equal to or less than those shown above. At the receiver's end of the shipment, the removable contamination levels due to any mobilization of fixed contamination must be less than 10 times the limits shown above.

Wipes shall cover a 300 cm^2 physical area each and be averaged and reported to an area of 100 cm^2.

[2] DOT regulations specify a limit of 0.4 Bq/cm2 or 22 dpm/cm^2. Boeing utilizes the dpm/cm^2 unit; the equivalent in Bq/cm^2 is approximately 0.36 Bq/cm^2.

The **dose rate limits for exclusive use vehicles** are as follows. The definitive source for these limits is 49 CFR 173.441.

Dose Rate Limits for Exclusive Use Vehicles	
	Limit and Location
Exclusive use vehicle	• 2 mSv/h (200 mrem/h) at any surface of the vehicle, including top, side, underneath
	• 0.1 mSv/h (10 mrem/h) at any point 2 meters from outer lateral surfaces of vehicle
	• 0.02 mSv/h (2 mrem/h) in any normally occupied position unless persons are program-covered radiation workers
Exclusive use flatbed vehicle	• 2 mSv/h (200 mrem/h) at any point on the vertical planes projected from the outer edges of the vehicle, the upper surface of the load or load enclosure and on the lower external surface of the vehicle
	• 0.1 mSv/h (10 mrem/h) at any point 2 meters from outer lateral surfaces of the vehicle
	• 0.02 mSv/h (2 mrem/h) in any normally occupied position unless persons are program-covered radiation workers

The **dose rate and contamination limits for EMPTY exclusive use vehicles** are as follows. The definitive source for these limits is 49 CFR 173.443.

Limits for EMPTY Exclusive Use Vehicles		
Radiation dose rate	< 0.005 mSv/hr (0.5 mrem/hr) at each accessible surface	
Removable contamination		
$\beta\gamma$ emitters	< 36 Bq/100 cm^2	2200 dpm/100 cm^2
α emitters	< 3.6 Bq/100 cm^2	220 dpm/100 cm^2

The following are the surface contamination limits from 10 CFR 835, Appendix D (downloaded 3-29-2004 from gpo website). This is provided for convenience; any questions regarding these limits should be verified by consulting 10 CFR 835 directly.:

APPENDIX D TO PART 835—SURFACE CONTAMINATION VALUES

The data presented in appendix D are to be used in identifying the need for posting of contamination and high contamination areas in accordance with §835.603(e) and (f) and identifying the need for surface contamination monitoring and control in accordance with §§835.1101 and 835.1102.

SURFACE CONTAMINATION VALUES[1] IN DPM/100 CM[2]

Radionuclide	Removable[2 4]	Total (Fixed + Removable)[2, 3]
U-nat, U-235, U-238, and associated decay products	[7]1,000	[7]5,000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	20	500
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	200	1,000
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above[5]	1,000	5,000
Tritium and tritiated compounds[6]	10,000	N/A

[1] The values in this appendix, with the exception noted in footnote 5, apply to radioactive contamination deposited on, but not incorporated into the interior or matrix of, the contaminated item. Where surface contamination by both alpha-and beta-gamma-emitting nuclides exists, the limits established for alpha-and beta-gamma-emitting nuclides apply independently.

[2] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[3] The levels may be averaged over one square meter provided the maximum surface activity in any area of 100 cm^2 is less than three times the value specified. For purposes of averaging, any square meter of surface shall be considered to be above the surface contamination value if: (1) From measurements of a representative number of sections it is determined that the average contamination level exceeds the applicable value; or (2) it is determined that the sum of the activity of all isolated spots or particles in any 100 cm^2 area exceeds three times the applicable value.

[4] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by swiping the area with dry filter or soft absorbent paper, applying moderate pressure, and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. (Note—The use of dry material may not be appropriate for tritium.) When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area shall be based on the actual area and the entire surface shall be wiped. It is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination levels are within the limits for removable contamination.

[5] This category of radionuclides includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.

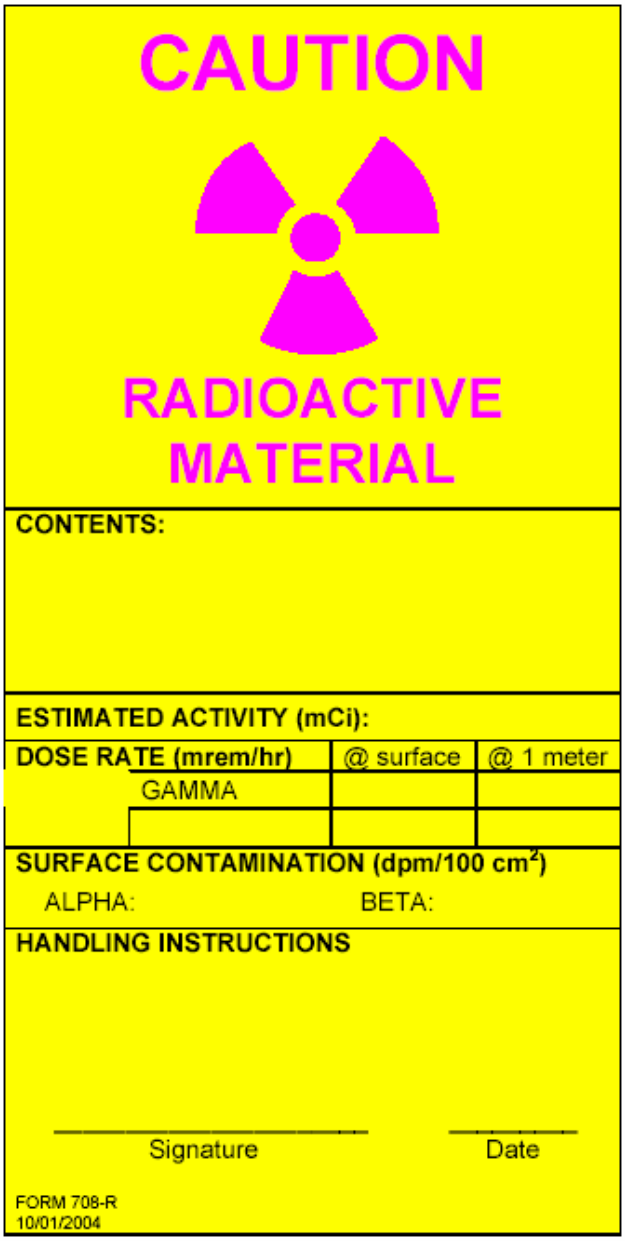
[6] Tritium contamination may diffuse into the volume or matrix of materials. Evaluation of surface contamination shall consider the extent to which such contamination may migrate to the surface in order to ensure the surface contamination value provided in this appendix is not exceeded. Once this contamination migrates to the surface, it may be removable, not fixed; therefore, a "Total" value does not apply.

[7] (alpha)

[58 FR 65485, Dec. 14, 1993, as amended at 63 FR 59688, Nov. 4, 1998]

APPENDIX D. SPECIMEN FORMS[3].

RADIOACTIVE MATERIAL TAG
(FORM 708-R)


CAUTION

RADIOACTIVE MATERIAL

CONTENTS:

ESTIMATED ACTIVITY (mCi):

DOSE RATE (mrem/hr)	@ surface	@ 1 meter
GAMMA		

SURFACE CONTAMINATION (dpm/100 cm^2)

ALPHA: BETA:

HANDLING INSTRUCTIONS

Signature Date

FORM 708-R
10/01/2004

Alternate layouts and designs may be utilized as long as the tag contains substantially the same information as this specimen.

[Current Form: http://leos.web.boeing.com/cgi-bin/formtrackleos/track3.pl?708-r.doc]

[3] The forms listed in the procedure are intended to be specimen forms. They reflect the general design and data requirements of the forms described in the procedure. Minor form revisions or updates are NOT necessarily reflected in the procedure. Current forms can be obtained from the Boeing form library or from current departmental form stocks. Sources of current forms are noted beneath each form.

RADIATION SURVEY REPORT
(FORM 732-A)

BOEING

RADIATION SURVEY REPORT

FACILITY:	
LOCATION:	

SAMPLE NUMBER	DATE SAMPLED	DATE ANALYZED	PURPOSE: DESCRIPTION	UNITS LIMIT				

COMMENTS:	INSTRUMENT				
	IDENTIFICATION				
	CALIBRATION DUE				
	BACKGROUND				
	EFFICIENCY				
SAMPLED BY: ANALYZED BY:	COUNT TIME				
REVIEWED BY: DATE:	Page of				

FORM 732-A REV 10-2005

[Current Form: http://leos.web.boeing.com/pol-proc/pc-forms/732-a.xls]

VEHICLE DEPARTURE SURVEY
(FORM X31848)

RADIOACTIVE MATERIAL TRANSPORTATION DEPARTURE SURVEY FORM

Page of

Carrier: _____ Trailer _____ Truck #: _____

Survey Date: _____ Departure Date & Time: _____ _____

Instruments

Model: _____ S/N: _____ Cal. Due Date: _____

Model: _____ S/N: _____ Cal. Due Date: _____

Contents: _____

Remarks: _____

LOCATION		CONTAMINATION Bq/100 cm^2 (2)		RADIATION mSv/h			
		Beta-Gamma	Alpha	Surface	1 Meter	2 Meter	Cab
Trailer Surface (Avg.)							
Trailer Surface (Max.)							
Top							
Bottom							
Front							
Back							
Driver Side (Trailer)							
Passenger Side (Trailer)							
Occupied Area (Cab)							
Contents (Packages) Max.							
Vehicle Limits:	Normal	N/A	N/A	N/A	0.5	N/A	N/A
	Exclusive Use	40	4	2	N/A	0.1	0.02
Package Limits:	Normal	40	4	2	0.1	N/A	N/A
	Exclusive Use	40 (4)	4 (4)	2 (1)	N/A	N/A	N/A

Notes:
(1) 10 mSv/hr package limit if ALL OTHER 49 CFR 173.441 (b) "Exclusive Use" conditions are met.
(2) Vehicle contamination surveys are performed after vehicle arrival and prior to loading.
(3) Radiation Safety abbreviated notations: NDA = No Detectable Activity (contamination).
Bkg = No radiation detected above background.
Background = _____ mSv/h
(4) Removable contamination not to exceed 10 times the limit upon delivery to receiver site.

Radiation Safety Surveyor: Signature Date:

1 dpm = 0.017 Bq
1 mrem/h = 0.01 mSv/h

Radiation Safety/D641

X31848 Rev (13 SEP 04)

[Current Form: http://leos.web.boeing.com/cgi-bin/formtrackleos/track3.pl?x31848.doc]

RADIOACTIVE MATERIALS SHIPPING INSTRUCTIONS
(Form 710-S-1)

The Boeing Company

RADIOACTIVE MATERIAL SHIPPING INSTRUCTIONS

REQUESTER	DEPT. AND GROUP	MAIL ADDRESS	PHONE	MANAGER'S APPROVAL	DATE

DATE TO BE SHIPPED	CONTRACT NO.	GO NO.	SUB ACCT. NO.	P.O. NO.	FREIGHT CHARGES
					☐ TO BE PREPAID ☐ COLLECT FROM CONSIGNEE

SECURITY CLASSIFICATION OF ITEMS TO BE SHIPPED:
☐ UNCLASSIFIED ☐ CONFIDENTIAL ☐ SECRET

METHOD OF TRANSPORTATION DESIRED OR SPECIFY OTHER:
☐ RAIL ☐ RAIL EXPRESS ☐ TRUCK ☐ AIR FREIGHT ☐ PARCEL POST ☐ OTHER _____

ITEMS DESCRIBED ON THIS FORM ARE TO BE SHIPPED TO:

NAME: _____

ADDRESS: _____

CITY, STATE, ZIP CODE: _____

HAZARDS
☐ RADIOACTIVE ☐ FISSILE
☐ FLAMMABLE ☐ PYROPHORIC
☐ EXPLOSIVE ☐ TOXIC
☐ CORROSIVE
☐ OTHER _____

SPECIAL SHIPPING INSTRUCTIONS (REQUESTER)

REASON FOR SHIPMENT:
☐ TO DOE-APPROVED SITE FOR BURIAL
☐ TO SUPPLIER/CO. DIVISION FOR SERVICES
☐ EXCESS PROPERTY DISPOSITION
☐ _____

PACKAGING INSTRUCTIONS

PACKAGE AND TRANSPORT AS NOTED BELOW

SHEA EH&S Support _____ DATE _____

SHIPPING INSPECTION

OPERATION	DOE SITE RESTRT.	QA INSP.	DATE
MATERIAL VERIFICATION			
PACKAGE*			
LOADING			

*INSPECTION INCLUDES CHECKS FOR: (1) PROPER CLOSURE AND SEALS; (2) CONTAINER DAMAGE; AND (3) PROPER IDENTIFICATION OF PACKAGE.

RADIATION SAFETY APPROVAL

NUCLEAR MATERIALS MANAGEMENT (IF SNM OR SOURCE MATERIAL IS PRESENT)

ITEM NO.	QUANTITY	PACKAGE IDENT. NO.	DESCRIPTION	RAD MAT'L TRANSP. UNITS

DOE SITE RESTORATION APPROVAL _____ DATE _____

SHEA EH&S Support APPROVAL _____ DATE _____

PACKAGING APPROVAL

DATE PACKAGED:	CARTONS:	BOXES:

CRATES:	WEIGHT:	DRUMS:	DIMENSIONS:

FOR SHEA USE ONLY

PACKING SHEET NO.:	B/L NO.:	DOT ID NO.:

SHIPPED:	CARRIER:

Form 710-S-1 REV. 9/28/05

[Current Form: http://leos.web.boeing.com/cgi-bin/formtrackleos/track3.pl?710-s-1.doc]

RADIOACTIVE CONTAINER SURVEY
(Form X31850)

RADIOACTIVE CONTAINER SURVEY

Page ____ of ____

Container ID No.: _____ Type: Box: _____ Drum: ____ Other: _____

Survey Location: RMHF: _____ Generator Facility: _____

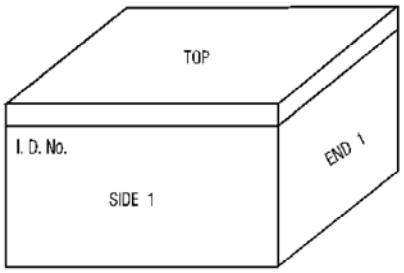
Instruments

Model: _____ S/N: _____ Cal. Due Date: _____

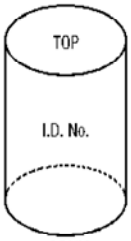
Model: _____ S/N: _____ Cal. Due Date: _____

Signature: _____ Date: _____

Remarks: _____



BOX



DRUM

Radiation Survey Summary (mR/h)

Location	Contact	30 cm *	1 Meter	2 Meter
Top				
Bottom				
Side 1 (Dia.)				
Side 2				
End 1				
End 2				

TRANSPORT LIMITS

1. Normal: < 200 mR/h contact and transport index < or = 10
2. Exclusive Use: < 200 mR/h OR < 1000 mR/h contact IF ALL OTHER 49 CFR 173.441(b) EXCLUSIVE USE CONDITIONS ARE MET.

NOTIFY D642 Management in writing if contact reading > 200 mR/h.

Person notified: _____

Signature: _____ Date: _____

* NTS CRITERIA:

1. >100 mR/h (1mSv/h) @ 30 cm shall be considered for remote handling.

X31850 NEW (13 AUG 04)

[Current Form: http://leos.web.boeing.com/cgi-bin/formtrackleos/track3.pl?x31850.xls]

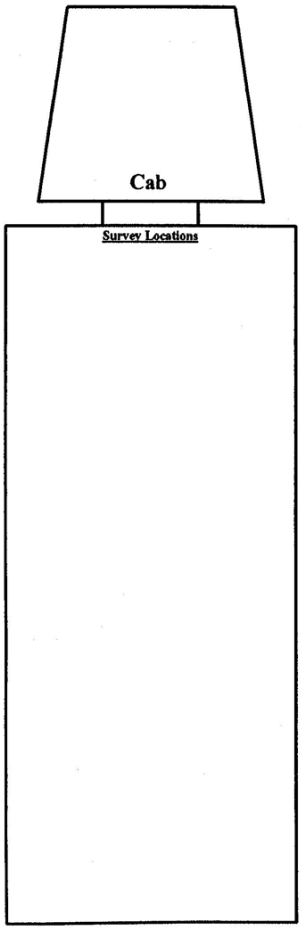
INCOMING VEHICLE SURVEY FORM
(Used In Conjunction With A Standard Survey Form (732A))

BOEING

RADIOACTIVE WASTE SHIPMENT INCOMING VEHICLE SURVEY

Carrier	
Truck #	
Trailer #	
Trailer Type	
Shipment #	

Use Survey Form (732A) for recording of survey results



1 Bq = 60 dpm	1.7 Bq = 100 dpm	0.3 Bq = 20 dpm

[Current Form: Radiation Safety Department]

APPENDIX E. DISCUSSION AND EXAMPLES OF ACTIVITY ESTIMATION FOR CONTAMINATED MATERIALS

E.1 DISCUSSION

The estimation of total radioactivity for a container of contaminated material is just that, an "estimation". It is our objective to perform the estimation as rigorously as feasible to result in an accuracy within an order of magnitude (factor of 10) of the actual value. Efforts should be made to achieve an estimate believed to be within a factor of two. The purpose of the activity estimate is to permit the shipper and transporter to manage the shipment within regulatory constraints, to permit the receiver to account for inventory, and, in case of a spill or other emergency, allow emergency response persons to estimate the degree of potential hazard involved.

The radioisotopes in the package for shipment must be identified if not known and the activity calculated for each based on chemical or radiometric analysis. In many cases historical sampling data may be available to identify the plausible mix of radioisotopes in materials contaminated or neutron activated in past years. Data more than 10 years old should be confirmed by new sampling. For each container, measurements, sampling or other source information must be obtained before the materials are loaded into the container.

Smears or direct instrument readings for both alpha and beta activity of materials before packaging may be used to identify radioisotopes present. If alpha activity is observed then isotopes possibly present are:

Plutonium - 239

Uranium - 234, 235, 238

Thorium - 230, 232

Americium - 241

The presence of beta activity could indicate the presence of:

Strontium - 90

Cesium - 137

Cobalt - 60

Uranium and thorium decay products

Samples which indicate gamma activity are easily analyzed by gamma spectroscopy for those emitters. However, pure beta and alpha emitters may also be present and their activity contributions must be included. For example, in cases where cesium-137 is found, strontium-90 usually will be present. The amounts of each may be assumed to be equal unless the waste material has been exposed to water (Cesium is much more soluble than strontium.) Once chemical analysis has shown the current ratio of strontium to cesium for a particular waste material, that ratio may be applied in the future to a gamma spectroscopy cesium value to obtain

the strontium quantity as long as other conditions remain the same. Similarly, other isotopic ratios, once established, may be used to estimate isotope activities not identifiable by gamma spectroscopy.

E.2 POSSIBLE RADIOISOTOPES

Current nuclear operations at the Santa Susana Field Laboratory (SSFL) consist of decontamination and decommissioning operations and processing or disposition of the resultant waste. Active work with nuclear materials stopped in the reactor test facilities many years ago and the short-lived decay products are no longer present.

The current listing of possible radioisotopes in waste originating from the various SSFL facilities is as shown in Table F-1.

TABLE E-1. POSSIBLE RADIOISOTOPES IN WASTE BY FACILITY

	RMHF		**059**	**024**	
H-3	X				
Fe-55	X		X		
Co-60	X		X	X	
Ni-63	X		X	X	
Sr-90	X				
Cs-134	X				
Cs-137	X				
Eu-152	X		X	X	
Eu-154	X		X	X	
Th-232	X				
U-234,235,238	X				
Pu-238, 239, 240, 241	X				
Am-241	X				

Among all isotopes considered for transuranic determination[4], only ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Am have been found to be present at SSFL. The concentrations of these isotopes in any waste stream destined for any low level waste site (e.g., Nevada Test Site, Hanford Site) will not exceed 100 nCi/g.

E.3 APPROACHES AND EXAMPLES OF TYPICAL ESTIMATES FOR WASTE CHARACTERIZATION

The examples that follow show some approaches that may be used to estimate isotope activity in a waste shipment. For purposes of documentation, copies of all data sheets and supporting calculations used to determined activity estimates should be attached to the Lot Follower for storage in RMHF files. Another copy will be stored in Radiation Safety .

E.3.1 Estimating Total Activity for a Waste Box Using the Computer Program MicroShield

The computer program MicroShield, designed to calculate radiation levels to be expected from a distributed and shielded source, also may be used to calculate the exposure rate from a standard source activity and use this relationship to convert radiation readings to activity estimates. Currently, MicroShield is used to estimate curie value for most of the boxes of contaminated or radioactive waste generated by Boeing's remediation projects. In order to use MicroShield for this application, it is necessary that the relationships of the box content's radioisotopes be known.

This method involves using radiation readings taken on each surface of the box, both at contact, and at one meter and two meters from each surface (eighteen total measurements).

The shielding program is menu driven and requires the following inputs:

- Geometry configuration. A number of cases are stored in the program. Generally, Geometry 11, which is a rectangular solid source with slab shielding, is used for a waste box. Geometry 7, which is a cylindrical solid source with side shields, is used for drums.
- Source dimensions (source is the contents of the waste box). Height, width, and length of the box (or height and radius for a drum) is entered from the perspective of the dose measurer. Thus, the values used for these parameters will depend on the side of the box being modeled (e.g., top/bottom, side or end). Dimensions may be expressed in centimeters, meters, inches, or feet. The standard waste box dimensions for each of the three characteristic surfaces may be placed in program files for repeated use.
- Distance(s) from the nearest outside surface of the container to the point of the dose rate measurement.
- Activity basis for the calculation. In past calculations (see Appendix G), one (1) millicurie is used for the basis for proportional adjustments. The software tool described in

[4] These isotopes are ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{242}Pu, ^{244}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{243}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm, ^{248}Cm, ^{250}Cm, ^{247}Bk, ^{249}Cf, and ^{251}Cf.

Appendix F utilizes a standard 1 curie total activity to establish “mr/hour” per curie reference rates.

- Density(ies) of the principal source materials in g/cc. The separate densities must sum up to equal the overall density of the waste box contents as determined from the box volume and net weight. To aid in these calculations, a spreadsheet (in Excel) has been developed to determine the separate component densities. The spreadsheet calculations are based on the net weight of the contents of the box (less the mass of the container and any non-radioactive materials added for shielding), the box dimensions, and estimated volumes for each separate component (e.g., steel, concrete, etc.) in the box. A section of the spreadsheet tool described in Appendix F simplifies these calculations.

- Thicknesses of other external intervening attenuators; e.g., waste box sidewall, and air between the box surface and the point of measurement. These distances must sum to equal the distance between the source surface and the-point of measurement. The spreadsheet tool in Appendix F simplifies these calculations.

- Radionuclide identification for the emitter(s) of the penetrating radiation.

E.3.2 Determination of Content from Calibrated, Direct Gamma Spectroscopy.

The In-Situ Object Counting System (ISOCS) can be utilized to determine isotopic content of a waste package. Refer to the Users Manual for the system for detailed instructions in its use.

Use of this approach directly determines a reference concentration of Cs-137 which can then be used to ratio other amounts of radioactive isotopes in the container.

APPENDIX F; INSTRUCTIONS FOR USE OF CHARACTERIZATION SOFTWARE TOOL

F-1 General

1.1 The process involves the use of laboratory radiological data scaled to a container using Microshield, ISOCS gamma spectroscopy, or laboratory assessment of the waste media.

1.2 The process is intended to give a supportable, documented estimate of radiological characterization data and the resulting total activity of a container.

1.3 The key elements of the process are:

1.3.1 Description of the container and contents

1.3.2 Determining a likely mixture of radioisotopes for a particular waste stream / container using vendor laboratory services and process knowledge.

1.3.3 (For Microshield calculations) Characterizing the physical media for the purposes of calculating scaling factors from MicroShield.

1.3.4 (For Microshield calculations) Determining the radiation levels emanating from the container

1.3.5 (For Microshield calculations) Using the weighted average of three sets of readings (i.e., at 30 cm, 1 meter, and 2 meters from the package surface) as a reference point for calculation of scaling factors using Microshield.

1.3.6 Using the scaling factor from MicroShield, ISOCS readings, or laboratory assessment of waste media to adjust the total activity for all isotopes determined to be in the waste.

1.3.7 Generating a hardcopy summary document for inclusion in the waste characterization file maintained by RMHF personnel.

1.4 The process utilizes a standardized Excel template spreadsheet for different container geometries. This procedure uses a template for a rectangular container. Versions of the template also support calculations for a cylindrical container.

The template is designed to perform the calculations in a transparent manner, and then to document the results for inclusion in a characterization file for the container.

1.5 This instruction corresponds to the Excel Spreadsheet *New Template (01-05-2007).xls* or versions of a derivative spreadsheet that corresponds to the layout and “look and feel” of the original. The spreadsheet contains a number of Visual Basic routines that are referred to in the instructions.

F-2 Description of Container and Contents

2.1 The key sections of the worksheet for container characterization are illustrated below:

1) Fill in all pertinent measurement data regarding dimensions and weight of box

Date:	1/5/2007					
Package ID:	Lift Liner GG-0158					
Content:	Soil, Concrete, Asphalt from RMHF Pond					
Physical Data						
			Length	Width	Height	
Container Type:	Lift Liner		96	66	58	inches
			243.8	167.6	147.3	cm
			8.0	5.5	4.8	feet
Tare Wt:	40	lbs				
Gross Wt:	20140	lbs				
Net Wt:	20100	lbs				
	9.12E+06	grams				
Volume:	212.7	cu. ft.				
	6.02E+06	cm3				
Density:	1.51E+00	g/cm3				

Tab: “1) Box Data”

2.2 Entries are self-explanatory. Bold, black lettering with yellow highlight are fields requiring entry from the user. Red lettered fields are calculations that require no intervention from the user.

The field of most interest in the Density field. This value is used in calculating the total activity in the container

2.3 Standard containers used are rectangular boxes and cylindrical containers (e.g., drums).

F-3 Determination of Waste Stream / Container Isotopic Concentrations

3.1 Spreadsheet Tab "2) Isotopic Data" provides calculations for the ratioing of isotopes in the waste stream / container.

3.2 Over the life of the operation, SSFL dealt with a number of alpha, beta, and gamma emitting radioisotopes. Typical alpha emitters generally came from the Thorium (Th), Uranium (U), and Plutonium (Pu) families. Strontium (Sr) was commonly present, as were Cobalt (Co) and Cesium (Cs).

3.3 Operations generated waste requiring disposal as radioactive materials. Generally, types of operations generated relatively generic waste streams, thus a generic waste stream could be characterized and applied to specific containers.

3.4 Waste streams were generally characterized by obtaining isotope-specific analyses from third party laboratories. SSFL was also able to perform gamma spectroscopy, and to a limited extent, could perform gross beta and alpha analyses on materials.

3.5 Since the start of decontamination and decommissioning, the site has obtained a number of samples and characterized them. These samples may be specific to certain wastes (i.e., contents of a specific waste stream), or more generic (i.e., consolidated air sample or smear media for a particular facility).

3.6 Characterization results are documented and data summaries are prepared listing the results of the various samples that have been conducted over time. This procedure references a spreadsheet used to summarize the characterization data (Characterization Data Summary File); previous approaches used a number of "ad hoc" summary spreadsheets of varying layouts (which are still acceptable for use).

The summary spreadsheet provides a consolidated source for radiological data. The referenced spreadsheet also provides a data table where isotopes have been normalized to the total activity in a sample. This permits sample data of various media types and sources to be utilized together to determine isotopic ratios.

The primary characteristics of the summary spreadsheet are illustrated below:

- Each column summarizes a vendor or in-house characterization. The column header provides identification data for the original source of the data, and the last row of the data table provides a link to the original spreadsheet which was used as the source of the data.

Numbers in these columns are LINKED to the original spreadsheet and should not be manually changed or adjusted. The ORIGINAL spreadsheet should be reviewed and corrected if errors are noted. Note that in this table, the activity units (pCi/sample, pCi/g, etc.) do not have to be the same for different columns and data sources.

Isotopes	RMHF Evaporator Sludge (pre-1998) Waste Characterization Vendor Analysis pCi/sample	RMHF Evaporator Sludge (after 1998) Waste Characterization Vendor Analysis pCi/g	RMHF Evaporator Sludge (~ 2003) Waste Characterization Vendor Analysis pCi/sample
H-3			
Na-22			
Ar-39			
Ca-41			
Fe-55			
Ni-59			
Co-57			
Co-60	5.25E+03	2.82E+03	3.47E+03
Ni-63			
Kr-85			
Sr-90	5.78E+05	6.60E+04	2.98E+05
Nb-94			
Ba-133			
Cs-134			
Cs-137	2.48E+05	2.70E+05	1.67E+05
Pm-145			7.97E+01
Eu-152			5.70E+01
Eu-154			3.43E+01
Eu-155		6.30E+02	7.31E+01
Pb-210			
Ra-226			2.56E+02
Th-228			1.91E+01
Th-230		3.20E+00	

The data summary is hyperlinked to the original spreadsheet or source of data at the bottom of the results columns.

Link:	RMHF_Evap_Sludge_BOX_CHAR.XLS	RMHFEvapH2O98.xls	2003-RMHF Evaporator Sludge ISOCS 2003.XLS

- A visual basic routine attached to each datasheet is used to calculate a rationalized value for each isotope to the total activity in the sample. The RATIO of activities, not the absolute values, is utilized in the waste characterization process. Use of ratios permits the combination of a number of different data sources in the characterization process.

Calc Ratios	RMHF Evaporator Sludge (pre-1998)	RMHF Evaporator Sludge (after 1998)	RMHF Evaporator Sludge (~ 2003)
Activity Ratios to Total Activity	Waste Characterization	Waste Characterization	Waste Characterization
	Vendor Analysis	Vendor Analysis	Vendor Analysis
	Ratio to Total Activity	Ratio to Total Activity	Ratio to Total Activity
H-3			
Na-22			
Ar-39			
Ca-41			
Fe-55			
Ni-59			
Co-57			
Co-60	6.08E-03	7.98E-03	7.13E-03
Ni-63			
Kr-85			
Sr-90	6.69E-01	1.87E-01	6.11E-01
Nb-94			
Ba-133			
Cs-134			
Cs-137	2.87E-01	7.64E-01	3.43E-01
Pm-145			1.64E-04
Eu-152			1.17E-04
Eu-154			7.04E-05
Eu-155		1.78E-03	1.50E-04
Pb-210			
Ra-226			5.25E-04
Th-228			3.92E-05
Th-230		8.95E-06	

One or more of these ratio columns are then manually cut-and-pasted into the container spreadsheet template to permit summary values to be calculated.

3.7 Starting at column "B" of the Characterization worksheet, columns may be copied and pasted from the characterization summary spreadsheet RATIOS table (NOT the data tables) to the container template (starting at Column J). Up to ten data columns may be selected from any source determined to be pertinent by the individual performing characterization.

Mixture 1		Mixture 2	
RMHF RMHF (2002) Smear Composite Vendor Analysis Original Ratio to Total Activity	RMHF RMHF (2002) Smear Composite Vendor Analysis Modified Ratio to Total Activity	RMHF Composite Smears (2004) Smears Vendor Analysis Original Ratio to Total Activity	RMHF Composite Smears (2004) Smears Vendor Analysis Modified Ratio to Total Activity
1.01E-02	1.01E-02	1.78E-03	1.78E-03
3.64E-01	3.64E-01	2.51E-02	2.51E-02
5.60E-01	5.60E-01	9.54E-01	9.54E-01
1.17E-04	1.17E-04	8.42E-05	8.42E-05
	1.61E-05		2.74E-05
7.82E-05	7.82E-05	7.50E-05	7.50E-05
4.44E-03	4.44E-03	5.19E-03	5.19E-03
2.77E-04	2.77E-04	2.30E-04	2.30E-04
1.39E-03	1.39E-03	4.11E-04	4.11E-04
8.88E-04	8.88E-04	4.51E-04	4.51E-04
2.74E-02	2.74E-02	3.59E-03	3.59E-03
1.61E-02	1.61E-02	2.11E-03	2.11E-03
1.47E-02	1.47E-02	5.94E-03	5.94E-03
7.40E-05	7.40E-05	8.64E-04	8.64E-04
1.00E+00	1.00E+00	1.00E+00	1.00E+00

Columns may be stacked out, to a maximum of 10 entries (Col AB).

Raw *ratio* data from the Characterization Spreadsheet is *Edit/Cop*(ied) from the Characterization Sheet, then *Edit/Paste Special/Value* entered into the gray column of the Container Sheet. The top of the pasted column should be positioned to start at Row 7.

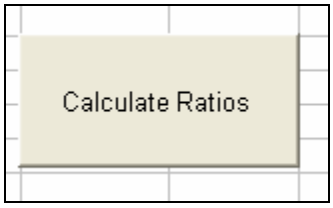
Make NO entries into the unhighlighted (white) columns. These will be populated by calculated values.

Row and column heights may be adjusted to permit acceptable display formats.

A hyperlink is provided to the Characterization Summary File for convenience.

3.8 Because different analyses may have measured different isotopes, not all sample data may contain information about all isotopes. It is necessary, however, when comparing different datasets, that all isotopes be provided with either a value from direct measurement, or with a value determined by estimation (for example to a cesium-137 measurement).

This estimation process is performed by a Visual Basic routine started by pressing this button after the data columns have been added to the spreadsheet.



This activates a Visual Basic routine that performs the following functions:

- All data ratio columns are scanned for each isotope. If a positive (non-blank) field is encountered, then the isotopic and Cs-137 values are noted.

- At the end of the row for an isotope, if any positive values are found, a ratio of all positive isotope values and the correlated cesium values for those sample sets only are used to calculate a ratio of the isotope to the cesium values. This is recorded in Column I.

F	G	H	I
	Ratio to Adjusted Total Activity for All Samples		Initial Ratio to Cesium-137 for Calculating "missing" isotope data
H-3			
Na-22			
Ar-39			
Ca-41			
Fe-55			
Ni-59			
Co-57			
Co-60			5.46E-03
Ni-63			
Kr-85			
Sr-90			1.69E-01
Nb-94			
Ba-133			
Cs-134			
Cs-137			1.00E+00
Pm-145			

Note that if no isotopic measurement value or no cesium measurement value is found, the reading is IGNORED. There must be an isotope AND a cesium value present for the data to be used for ratio calculation. The end result of the VB routine is a calculated average ratio to cesium using only those ratio tables where both have values.

- Once a cesium ratio is calculated, the VB routine calculates a projected value for any isotope which does not have a value, but where a cesium value occurs. This is determined by multiplying the cesium value in the column by the average ratio value in Column I for that isotope. Each blank in the spreadsheet is filled in using this method.

  Note that the *original* data (grey field) is not changed. The calculated values are placed in the associated calculation field (unhighlighted) to the entry data column's immediate right. This is to ensure that the calculation process is transparent from original data to calculated data.

  Data that is based upon a measured ratio from the characterization file is written directly to the processed column (black font on white background). The calculated values are highlighted using red font on yellow background.

3.9 Special adjustments are made in the isotope ratios for the following situations:

- Pu-239/240 mixed measurements are partitioned into separate Pu-239 and Pu-240 measurements. This is based upon a ratio assuming that 63% of any combined report is Pu-239, and 37% is Pu-240.

- If no Pu-241 value is reported in direct measurement, but an Am-241 value is provided, a Pu-241 value is derived by multiplying the Am-241 value by a factor of ten.

3.10 Once the "missing" values are calculated, a ratio to total activity across all samples is calculated by the Visual Basic routine.

F	G
	Ratio to Adjusted Total Activity for All Samples
H-3	
Na-22	
Ar-39	
Ca-41	
Fe-55	
Ni-59	
Co-57	
Co-60	4.48E-03
Ni-63	
Kr-85	
Sr-90	1.39E-01
Nb-94	
Ba-133	
Cs-134	
Cs-137	8.13E-01
Pm-145	
Eu-152	
Eu-154	

All values (either from direct measurement or determined from ratio to cesium levels) are included in this calculation.

Note that once calculated, these ratios to total activity are used for all additional calculations. The "ratio to Cesium" values are *not* used, as they were calculated using table values that did not include the special adjustments described above.

A ratio to cesium can be derived by the following formula:

$$\text{Ratio}_{\text{isotope to Cs-137}} = \frac{\text{Ratio}_{\text{isotope to Total}}}{\text{Ratio}_{\text{Cs-137 to Total}}}$$

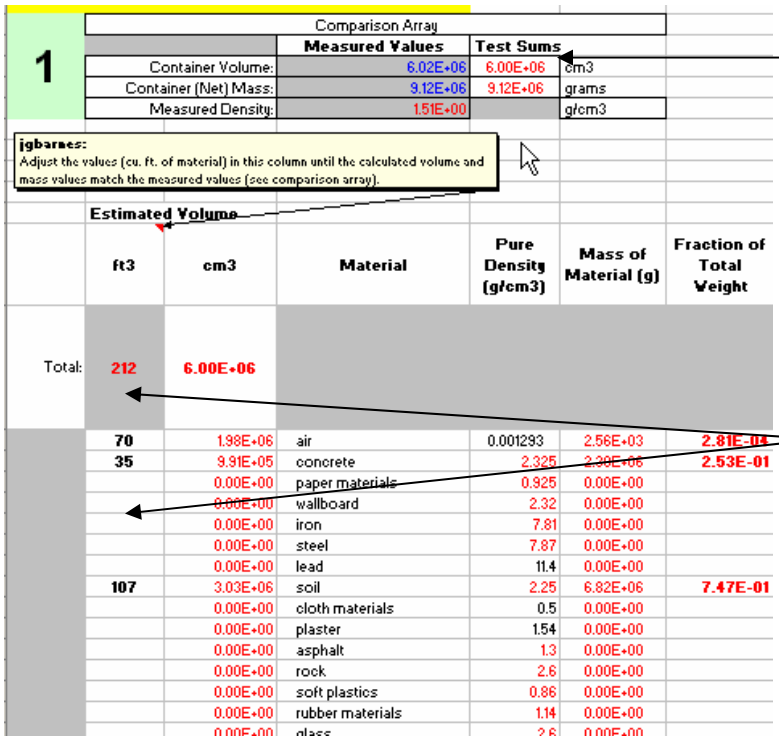
F-4 Characterization of Waste Media for Shielding and Buildup Factor determination

NOTE: This step is not required if MicroShield calculations are *not* being used for waste characterization.

4.1 Radioactive materials are generally evenly distributed through the volume of the waste media. This media can serve as a shielding material and can affect the buildup factor of the propagating radiation. This section provides a methodology to estimate this effect in the waste container.

4.2 The Worksheet "3) Contents Density Data" contains the data and Visual Basic routines used to determine these values for the container. The sheet is divided into three sections.

4.3 Section 1 provides a methodology for the description of the mixture of common waste materials in the SSFL waste streams.



1	Comparison Array			
		Measured Values	Test Sums	
	Container Volume:	6.02E+06	6.00E+06	cm3
	Container (Net) Mass:	9.12E+06	9.12E+06	grams
	Measured Density:	1.51E+00		g/cm3

jgbarnes:
Adjust the values (cu. ft. of material) in this column until the calculated volume and mass values match the measured values (see comparison array).

	Estimated Volume					
	ft3	cm3	Material	Pure Density (g/cm3)	Mass of Material (g)	Fraction of Total Weight
Total:	212	6.00E+06				
	70	1.98E+06	air	0.001293	2.56E+03	2.81E-04
	35	9.91E+05	concrete	2.325	2.30E+06	2.53E-01
		0.00E+00	paper materials	0.925	0.00E+00	
		0.00E+00	wallboard	2.32	0.00E+00	
		0.00E+00	iron	7.81	0.00E+00	
		0.00E+00	steel	7.87	0.00E+00	
		0.00E+00	lead	11.4	0.00E+00	
	107	3.03E+06	soil	2.25	6.82E+06	7.47E-01
		0.00E+00	cloth materials	0.5	0.00E+00	
		0.00E+00	plaster	1.54	0.00E+00	
		0.00E+00	asphalt	1.3	0.00E+00	
		0.00E+00	rock	2.6	0.00E+00	
		0.00E+00	soft plastics	0.86	0.00E+00	
		0.00E+00	rubber materials	1.14	0.00E+00	
		0.00E+00	glass	2.6	0.00E+00	

This section is used to match the density of the selected media components to both volume and mass of the container.

Volumes of material are designated in this column. Density values are used to estimate the associated mass of material.

The user adjusts the values in various mixtures until the calculated mass and volume values match those determined by direct measurement.

Both volumes (and mass) should be adjusted until the projected totals agree with the measurement values (as shown in the summary table above).

- When the volume/mass values have been determined, the spreadsheet calculates a "Fraction of Total Weight" value in Column G for each constituent.

4.4 Once the Weight Fractions have been calculated, the materials are parsed into their constituent elemental parts using standard chemical descriptions.

- A constituent spreadsheet is hyperlinked to the section. Constituent tables are entered into the container spreadsheet by cut and paste from this spreadsheet. *Every* material selected in Section 1 must have an parsing table entered into the Section 2 table:

2 — **Constituent Chart for Microshield**

Constituent Material	Element	Elemental amu	Relative Elemental Mass in Molecule / Mixture (Wt. Ratio)	Number of Atoms in Molecule	Weight Fraction in Waste Mixture	Relative Weight Ratio in Waste Mixture	Elemental Weight in Container (grams)	Gram-Moles in Container
air	Nitrogen	14.0	0.81	239	2.81E-04	2.27E-04	2.07E+03	1.48E+02
	Oxygen	16.0	0.19	56	2.81E-04	5.32E-05	4.85E+02	3.03E+01
	Argon	39.9	0.00	1	2.81E-04	9.51E-07	8.67E+00	2.17E-01
concrete	Hydrogen	1.0	0.006	148	2.53E-01	1.56E-03	1.42E+04	1.41E+04
	Oxygen	16.0	0.551	832	2.53E-01	1.39E-01	1.27E+06	7.93E+04
	Sodium	23.0	0.018	19	2.53E-01	4.56E-03	4.16E+04	1.81E+03
	Magnesium	24.3	0.002	2	2.53E-01	5.08E-04	4.63E+03	1.91E+02
	Aluminum	27.0	0.050	45	2.53E-01	1.27E-02	1.16E+05	4.29E+03
	Silicon	28.1	0.351	302	2.53E-01	8.86E-02	8.08E+05	2.88E+04
	Sulfer	32.1	0.001	1	2.53E-01	3.35E-04	3.05E+03	9.53E+01
	Potassium	39.1	0.003	2	2.53E-01	8.17E-04	7.45E+03	1.91E+02
	Calcium	40.1	0.015	9	2.53E-01	3.77E-03	3.44E+04	8.57E+02
	Iron	55.8	0.002	1	2.53E-01	5.84E-04	5.32E+03	9.53E+01
Soil	Silicon	28.1	0.467	1	7.47E-01	3.49E-01	3.18E+06	1.13E+05
	Oxygen	16.0	0.533	2	7.47E-01	3.98E-01	3.63E+06	2.27E+05
					Checksums:	1.00E+00	9.12E+06	

Material parsing tables are extended down in these columns.

These are linked to the values in Section 1

Cut and Paste formulas into new entries

Table adjustments are done by hand editing, as shown above. Each component block should be separated by one empty line (as shown).

Adjust the checksum formulas, as needed. The "Relative Weight" checksum should equal a value of 1, and the "Elemental Weight" checksum should equal the mass of the container (based on the volume of materials selected in Section 1)

4.5 When the constituent table is completed, press the "Create Elements Listing" button. This will activate a Visual Basic program that will generate a description of the waste media at an elemental level. Conceptually, this is best conceived as the waste being composed of "molecules" with the characteristics described in the constitutent table.

Create Elements Listing

3		Custom Element Listing	Gram Moles in Container	Mixture Molecule Element Ratios	Element AMU (i.e., g / mole)	Mass by Element in Mixture Molecule (grams)	Wt. Ratio in Mixture Molecule		Microshield Wt. Ratio (%)	Microshield Number of Atoms
	1	Oxygen	3.06E+05	1410369	16.0	3.75E-17	5.37E-01		53.707	1410369
	2	Silicon	1.42E+05	655065	28.1	3.06E-17	4.38E-01		43.788	655065
	3	Aluminum	4.29E+03	19756	27.0	8.85E-19	1.27E-02		1.269	19756
	4	Sodium	1.81E+03	8342	23.0	3.19E-19	4.56E-03		0.456	8342
	5	Calcium	8.57E+02	3951	40.1	2.63E-19	3.77E-03		0.377	3951
	6	Hydrogen	1.41E+04	64977	1.0	1.09E-19	1.56E-03		0.156	64977
	7	Potassium	1.91E+02	878	39.1	5.70E-20	8.17E-04		0.082	878
	8	Iron	9.53E+01	439	55.8	4.07E-20	5.84E-04		0.058	439
	9	Magnesium	1.91E+02	878	24.3	3.55E-20	5.08E-04		0.051	878
	10	Sulfer	9.53E+01	439	32.1	2.34E-20	3.35E-04		0.034	439
	11	Nitrogen	1.48E+02	680	14.0	1.58E-20	2.27E-04		0.023	680
	12	Argon	2.17E-01	1	39.9	6.64E-23	9.51E-07		0.000	1

- Each element in the mixture is identified and the amount of that element (in *gram-moles*) in the container is calculated.
- The lowest value in the above calculation is then used as the lowest common denominator to rationalize the table into a ratio ("Mixture Molecule Element Ratios")
- Using the elemental atomic mass unit values, the "Mass by Element in Mixture" is calculated (converting from gram-moles to grams).
- A weight ratio for the mixture is then calculated.
- The values are then summarized into an entry table for the Microshield materials creation table (highlighted in green).

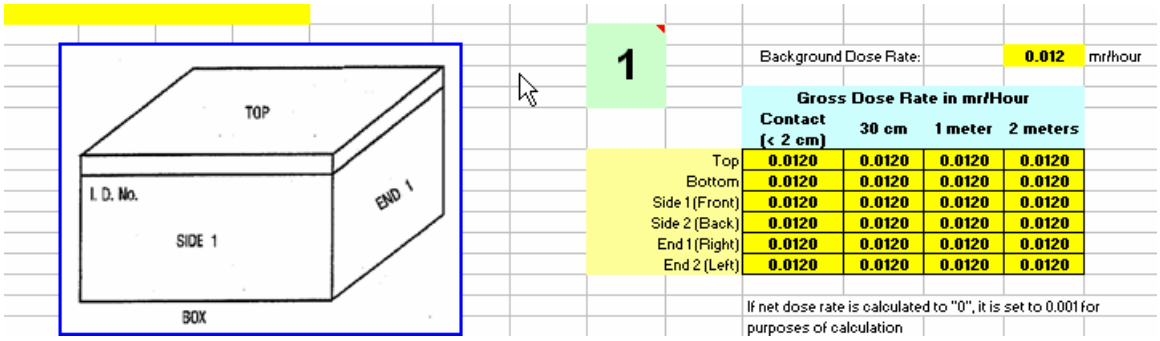
F-5 Use of Microshield to Calculate Conversion Factors for Dose Rate to Isotopic Activity

This section refers to Worksheet "4) Microshield Setup." It is *not* a required step if Microshield values are not being used for determination of isotopic activity.

5.1 Standardized survey data is obtained and recorded.

A standardized survey matrix is used. Readings are taken on each surface of the container (e.g., front, back, top, bottom, right, left for boxes; and top, bottom and side for drums). The Visual Basic code and survey matrices may vary for different container geometries.

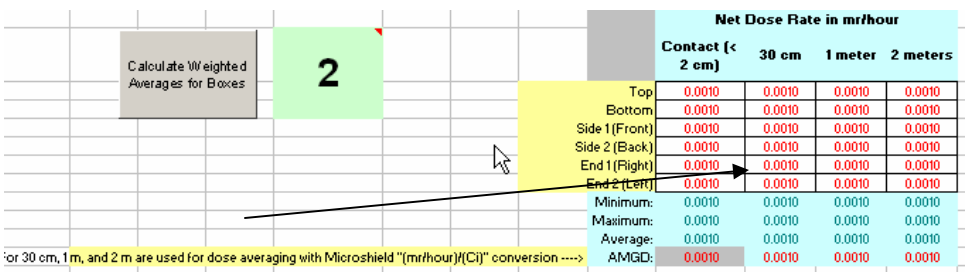
Section 1 of the worksheet provides a matrix for these entries. The matrix is filled in with the maximum reading obtained on each surface of the container at contact (< 2cm), 30 cm, 1 meter, and 2 meters from the box.



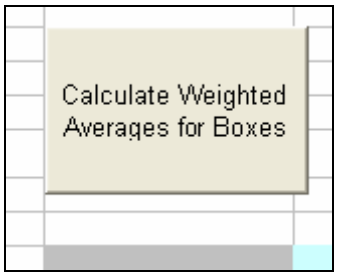
In the first section, the actual readings are recorded in the matrix *without* background subtraction. A background reference value is entered in the cell provided.

5.2 Net readings are calculated, and an Arithmetic Mean of Grouped Data (AMGD) is calculated.

The worksheet calculates a net dose rate for each standardized location. These values are displayed in Section 2 of the worksheet. If the net dose is zero (i.e., background), then a value of 0.001 mr/hour is arbitrarily assigned.



Several values for each distance and surface are summarized below the data matrix (i.e., minimum, maximum, average). The Arithmetic Mean of Grouped Data (AMGD) is used to smooth the variability in the radiation readings (see discussion following). This calculation is embedded in the spreadsheet in a VisualBasic routine activated by pressing this button on the spreadsheet.



There is generally great variability in radiation readings emanating from a container due to shelf-shielding, variations in activity concentrations, etc. This variability is numerically smoothed by calculating a weighted average of the data for each surface to obtain an average value for each surface. This is determined by using an Average Mean for Group Data matrix. An example of such a matrix is illustrated below; all matrices are listed to the right of the main readings matrix shown above.

Sample Arithmetic Mean of Grouped Data					
Contact Reading					
Class	Low	High	Class Mark (CM)	frequency (f)	CM * f
1	0.04	0.086	0.063	5	0.315
2	0.086	0.132	0.109	0	0
3	0.132	0.178	0.155	0	0
4	0.178	0.224	0.201	0	0
5	0.224	0.27	0.247	0	0
6	0.27	0.316	0.293	0	0
7	0.316	0.362	0.339	0	0
8	0.362	0.408	0.385	0	0
9	0.408	0.454	0.431	0	0
10	0.454	0.5	0.477	1	0.477
				Sum:	0.792
			Arithmetic Mean of Group Data:		0.132

An Arithmetic Mean of Grouped Data (AMGD)[5] permits each reading to be weighted so that the overall effect of outliers is reduced. As the radiation readings are used to adjust isotopic concentration data to the specific container, the use of these weighted averages gives a better reference level for these adjustments than a simple arithmetic mean.

The general process is:

- The data column in the matrix is scanned for the lowest value.
- The data column is then scanned for the highest value.
- A ten-element array is generated using the lowest and highest values as the start point for the low element of the table and the high value as the maximum point in the table. A series of ten “bins” are then defined to break the range into equal slices.

[5] See Levin; Statistics for Management. p. 48. The approach gives more weight to the more common measurements, and prevents one outlier from skewing the overall average dose on the container.

- For each bin, a “class mark” (CM) is calculated as the average between the highest and lowest value for that category.
- Each entry in the original dose matrix is then categorized into each bin (frequency (f)).
- A “CM*f” value is generated by multiplying the frequency and class mark for each bin.
- The “CM*f” value is then summed at the bottom of the table.
- The “CM*f” value is then divided by the total number of data elements found in the bins (in this case, that number is six (for boxes) or three (for drums)).
- The “CM*f” value is then linked to the AMGD field in the original data table.

If all values in the original dose table are equal, the AMGD is undefined. In this case, the AMGD is set to be equivalent to the arithmetic mean.

AMGD values are listed on the Net Dose matrix and are used later in the Microshield calculations.

5.3 Set-up the Microshield case file for the container, and calculate reference doses.

Worksheet Sections 3 – 7 provide instructions for setting up the Microshield file and calculating representative dose rates for the physical characteristics of this container, assuming a 1 Curie total activity in the continer. This will generate a mR/hour per Curie reference dose that will be used to estimate the activity of the container based upon the AMGD dose rates.

This procedure assumes that the user is familiar with the basic operations of the Microshield program. Instructions for the software are provided by the manufacturer and are beyond the scope of this instruction.

Refer to the worksheet for instructions in how to load and run the case file in Microshield.

Enter the resulting scaling factors (“mr/hour per Curie”) values in the cells in the Worksheet. These values are transferred to several other locations in the Spreadsheet for later calculations.

F-6 Activity Estimates

6.1 The following table in Worksheet "5) New Output" is automatically populated by a Visual Basic routine. This matrix is used to determine a best estimate of the container activity using MicroShield scaling factors. If Microshield values are *not* used, this table should be ignored (*but not altered)*.

Waste Volume:	3.99E+07	cm³			Net Weight:	1.39E+07	g				
Density:	3.49E-01	g/cm³									
Average Measured Dose Rates				Microshield (mr/hr) / (Ci)					Microshield Calculated Activity		
					(with Buildup)	(w.o. Buildup)	Average			mCi	
30 cm	0.001	mr/hour		(30 cm)	2.92E+01	1.36E+01	2.14E+01			4.68E-02	
1 m	0.001	mr/hour		(1 m)	1.83E+01	8.97E+00	1.36E+01			7.33E-02	
2 m	0.001	mr/hour		(2 m)	1.04E+01	5.16E+00	7.75E+00			1.29E-01	
									Average Microshield TOTAL Activity:	8.31E-02	mCi

Scaling factors which include buildup and which do not have buildup are averaged for a best estimate scaling factor.

A total activity is estimated from each of the 30 cm, 1 meter, and 2 meter measurement sets by scaling the AMGD dose for each distance to the scaling factors determined from Microshield. This results in three separate estimates of activity, which are then averaged in order to derive a total activity (which represents a best estimate).

6.2 Activity estimates can also be determined from ISOCS measurements of cesium-137 activity or by laboratory measurement of cesium-137 concentration. These values are entered in the specified cells on the "New Output" Worksheet, as shown:

Average Microshield TOTAL Activity:	5.42E-01	mCi
Cs-137 ISOCS value:	1.03E-01	mCi
Cs-137 Lab Value:	3.20	pci/g

The Microshield value is automatically inserted into this small table. The ISOCS and Laboratory value must be manually entered. These values are then linked into other areas of the calculation worksheet.

F-7 Calculation of Total Isotopic Activity, Decay Heat, and TRU Concentrations for the Container

7.1 Package Activity is then calculated in the next portion of the Worksheet.

The method of calculation should be selected using the appropriate "radio button" above the table. This will determine the activity value to be parsed into "per isotope" constituents. The selected value is displayed, as indicated.

Selecting these values (or changing values in the total activity entry cells) will erase the table in order to ensure that erroneous values are not left in the table accidently. Message boxes inform the user of the needed actions for recalculation.

Microshield Value
Lab
ISOCS Value
Lab Results

Compile Lot Follower Listing from Microshield

Microshield Value (mCi) 5.42E-01
ISOCS Value (mCi) 1.03E-01
Cs-137 Lab Value (pCi/g) 3.20

Calculate Calcuate Decay Heat &

Isotope	Ratio to Total Activity	Calculated Activities (mCi): Microshield Calculation	ISOCS (from Cs-137)	Laboratory Results (from Cs-137)	Decay Heat Factor (Watts / mCi)	Package Decay Heat (Watts)
Co-60	4.48E-03			1.61E-04	1.54E-05	2.48E-09
Sr-90	1.39E-01	NOT USED	NOT USED	4.97E-03	6.70E-06	3.33E-08
Cs-137	8.13E-01			2.92E-02	4.82E-06	1.41E-07
Th-228	1.08E-04			3.88E-06	3.27E-05	1.27E-10
Th-230	2.33E-05			8.37E-07	2.82E-05	2.36E-11
Th-232	8.23E-05			2.95E-06	2.43E-05	7.16E-11
U-234	3.59E-03			1.29E-04	2.88E-05	3.71E-09
U-235	1.93E-04			6.94E-06	2.77E-05	1.92E-10
U-238	7.11E-04			2.55E-05	2.53E-05	6.45E-10
Pu-238	7.19E-04			2.58E-05	3.32E-05	8.55E-10
Pu-239	1.67E-02			5.97E-04	3.11E-05	1.86E-08
Pu-240	9.78E-03			3.51E-04	3.12E-05	1.09E-08
Pu-241	1.11E-02			3.98E-04	3.18E-08	1.26E-11
Am-241	5.04E-04			1.81E-05	3.34E-05	6.04E-10

The calculations are relatively straightforward and summarized below:

When the "Compile Lot Follower Listing . . . " button is pressed, a Visual Basic routine does the following actions:

- The *Isotope* and *Ratio to Total Activity* columns are directly copied from the Isotopic Data Worksheet
- Depending upon which value has been selected as the basis for the activity estimate, a table of isotopic content (in millicuries) is calculated for each isotope in the table. The other two estimations are blanked and labeled as "not used" to avoid confusion.
- These millicuries values are then entered by hand into the "Isotopic (mCi)" section of the Radioactive Material Lot Follower (Form 652-A).

7.2 When the Lot Follower values are calculated, pressing the "Calculate Decay Heat . . . ." button activates a Visual Basic process that calculates the associated decay heat for each isotope in the package. Decay Heat factors are derived from the Hanford Waste Acceptance Criteria (WAC) document (the original table is included in the Spreadsheet package as Worksheet "REF) Decay Heat").

Total:	9.97E-10	Watts / cu ft.
	Transuranics (nCi/g)	
		Microshield (nCi/g)
	Pu-239	6.55E-05
	Pu-240	3.85E-05
	Pu-239/240	
	Am-241	1.98E-06
	TOTAL:	**1.06E-04**

The total value for decay heat is transferred to the Lot Follower.

7.3 Transuranic Isotopes (TRU) consists of Pu-239, Pu-240, and Am-241. The decay heat Visual Basic routine also calculates the specific activity (nCi/g) for transuranics. This total value is also entered on the Lot Follower.

Concentrations exceeding 100 nCi/gram must meet additional packaging and shipping requirements.

F-8 Documentation

8.1 The data generated by the spreadsheet is used to update the Lot Follower for the associated package.

8.2 The Spreadsheet is saved to the Container computerized file repository named with the container ID (e.g., “B-1007.xls”). This repository is periodically backed up to the SHEA server for safekeeping.

8.3 A printed copy of the Worksheet “5) New Output” is printed and filed in the Lot Follower data package.

APPENDIX G. INSTRUCTIONS FOR USE OF ALTERNATE SOFTWARE TOOLS, WITH EXAMPLES

Prior to the release of Revision F to this procedure, waste calculations were conducted using the techniques described below. These approaches are still valid, however, Revision F introduced a software tool which simplifies and standardizes the process to a great degree. Current calculations are generally done using the software tool.

After the necessary parameters are entered, the MicroShield program is run once for each of the nine symmetric dose determinations made on the waste box (for this method, the opposite box surfaces are assumed to be symmetric, i.e., top and bottom, opposite sides, and opposite ends, and thus only one MicroShield run is made for each pair of surfaces). For each run, MicroShield calculates a value for the dose rate that would be delivered for our input basis, which is one (1) millicurie of the source material. The resultant terms can then be combined with the eighteen observed mR/hr values for the box to obtain an estimate of the activity (mCi) of penetrating radiation emitter in the box.

In addition, the ratios of non-penetrating radiation emitters present to the amount of penetrating radiation emitter must be known to permit an estimate of activity from all the radioisotopes present. These ratios may be derived from process knowledge or isotopic analyses of appropriate samples.

To facilitate the final activity calculations, a second spreadsheet was developed (again in Excel) to combine the various MicroShield and dose rate data. An example of this calculational method, using the two spreadsheets, is given below.

<u>Example</u>

Assume that a Type 3 waste box (B-767) contains bags of soft trash, aluminum channel, concrete rubble, miscellaneous steel, and electrical pumps from RIHL. From the Lot Follower, the following information is known:

Gross weight = 1,773 lb. Tare weight = 640 lb. Net weight = 1133 lb.

Volume estimates are: Aluminum - 0.2 gal., concrete - 15 gal., iron - 36 gal., trash - 32.8 gal, plastic - 20.11 gal., and copper - 0.15 gal. Note: for these calculations, soft trash is represented by wood at a density of 0.6 g/cc and plexiglass at a density of 1.18 g/cc.

With these inputs, the density calculation spreadsheet is shown in Figure H-1.

Density Calculations for RMDF Waste Box:				**B-767**		**(Type 3)**			
8/6/97				**6-Aug**					
Gross Weight (lbs):			**1773**		**Container Volume (cu. ft):**			**51**	
Tare Weight (lbs):			**640**		**Container Volume (gal)):**			**385**	
Container Length (in):			**84**		**Container Net Weight (lbs):**			**1133**	
Container Width (in):			**46**		**Material Density (g/cm^3):**			**0.353**	
Container Height (in):			**23**						
Materials in Box:					1 cuft = 7.48 gal				
			Estimated						
			Relative	**Calculated Actual**					
		Material	**Volume**	**Volume**	**Density**				
Item	**Material**	**Density**	**(gal)**	**(gal)**	**(g/cm^3)**				
Void Space	Air	0.00122	--	308	0.0010				
Aluminum	Aluminum	2.70	0.2	0.2	0.0015				
Concrete	Concrete	2.35	15	15	0.0947				
Steel	Iron	7.86	6.45	6.7	0.1361				
Lead	Lead	11.35		--	--				
Lead Glass	Lead glass	4.36		--	--				
Trash	Wood	0.6	32.8	34	0.0529				
Plastic	Plexiglas	1.18	20.11	21	0.0637				
Copper	Copper	8.93	0.15	0	0.0036				
	Volume Correction:		**1.03**	**Total:**	**0.353**				
				Δρ:	**0.000**				

Figure G-1. Example Spreadsheet Calculation of Waste Box Component Densities

From the lot follower information, the spreadsheet has calculated final air, aluminum, concrete, iron, wood, plexiglass, and copper densities of 0.001, 0.0015, 0.0947, 0.1361, 0.0529, 0.0637, and 0.0036 g/cc respectively. The final values were determined from the initial volume estimates entered in Column 4, adjusted by a volume correction factor, to yield agreement between the "actual" box density (0.353 g/cc), and the sum of density components in Column 6. The volume correction factor can be adjusted by hand, or use can be made of the built-in "Solver" routine in Excel to adjust the factor to yield the required agreement (i.e., a $\Delta\rho = 0$).

Once the density values have been determined, these densities are entered into the MicroShield code along with the other required data.

Interpretation of rad readings on Box:				**B-767**		**(Type 3)**					
(per N001OP000030)				**6-Aug**							
8/6/97	**E. R. McGinnis**			**Geometric mean (mCi):**			**0.679**				
Now includes heat generation				**Arithmetic mean (mCi):**			**1.307**				
Background (mR/h):		**0.01300**		**Combined mean (mCi):**			**0.993**				
	Net Readings (mR/h)				**Gross Readings (mR/h)**						
	contact	**1 meter**	**2 meters**		**contact**	**1 meter**	**2 meters**				
top	2.887	0.187	0.067		2.900	0.200	0.080				
bottom	0.187	0.077	0.027		0.200	0.090	0.040				
side 1	0.187	0.017	0.009		0.200	0.030	0.022				
side 2	0.127	0.027	0.009		0.140	0.040	0.022				
end 1	0.107	0.017	0.009		0.120	0.030	0.022				
end 2	1.987	0.187	0.087		2.000	0.200	0.100				
MicroShield factors (mR/hr per mCi)											
	contact	**1 meter**	**2 meters**								
top	0.6351	0.1249	0.0455								
bottom	0.6351	0.1249	0.0455								
side 1	0.5370	0.0825	0.0308								
side 2	0.5370	0.0825	0.0308								
end 1	0.5160	0.0545	0.0192								
end 2	0.5160	0.0545	0.0192								
	Activity (mCi):										
	contact	**1 meter**	**2 meters**								
top	4.546	1.497	1.473								
bottom	0.294	0.616	0.593								
side 1	0.348	0.206	0.292								
side 2	0.236	0.327	0.292			minimize					
end 1	0.207	0.312	0.469			this					
end 2	3.851	3.431	4.531		1.307	1.2190					
						relative					
Geometric mean (mCi):			**0.6785**			stddev					
Arithmetic mean (mCi):			**1.3069**								
Combined mean (mCi):			**0.9927**								
	Activity estimates (mCi)				**Heat generation, W/cuft**						
ratio	**nuclide**		**daughters**								
0.05	Co-60	**0.0496**	Ni-60m	**0.0496**		**1.49E-08**					
0.3871	Sr-90	**0.3651**	Y-90	**0.3651**		**4.75E-08**					
0.95	Cs-137	**0.9431**	Ba-137m	**0.7981**		**8.66E-08**					
					total	**1.5E-07**					
TOTALS (mCi):		**1.3578**	**(w/o daughters)**								
		2.5706	**(with daughters)**								
(per N001OP000030)											

Enter recorded exposure rate readings in "gross" block. Enter background, and MicroShield conversion factors (mR/hr per mCi) at various distances. Adjust background, within reasonable bounds, to improve results (No net=0 results are acceptable), by minimizing the stddev. Print out and use results as necessary to complete Lot Follower.

Figure G-2. Spreadsheet Calculation of Waste Box Activity

The source of penetrating radiation is known to be 95% Cs-137 and 5% Co-60 by gamma spectroscopy of waste from the RIHL. Thus, 9.5×10^{-4} Ci of Cs-137 and 5×10^{-5} Ci of Co-60 should be entered as the Source (1 mCi total).

Start MicroShield and load the appropriate case for the Top/Bottom surfaces of a Type 3 waste box. Edit the case title, the source densities and materials, and the source isotopes as given above. Finally, enter secondary dose point distances in the Sensitivity Analysis menu corresponding to 1 meter and 2 meters from the appropriate box surface.

After the inputs are entered, run the MicroShield code. Upon completion, the code will return three calculated dose rate values (mR/hr), one for the reference dose point (contact) and one each for 1 meter and 2 meters. The values returned should be:

Contact = 0.6351 mR/hr

1 meter = 0.01249 mR/hr

2 meters = 0.004547 mR/hr

Repeat this process for the side and end surfaces of the Type 3 box to obtain six additional dose values.

Input the background, the eighteen measured dose values (mR/hr) for the waste box, and the nine calculated MicroShield dose values (mR/hr/mCi) into the activity calculation spreadsheet to determine the final average activity of the waste box. With these inputs, the density calculation spreadsheet is given in Figure G-2.

The spreadsheet calculates the gamma activity based on the assumed isotopic mix. The eighteen individual activities (from the eighteen measured dose values) are averaged both arithmetically and geometrically, and the final total activity is the average of these two values. Two averages are used so that the results are not overly dominated by unusually high or low values. Individual nuclide activities (including daughter products) are determined from the determined total and the assumed isotopic mix. In this example case, the total estimated activity in the waste box is determined to be 1.3578 mCi (without daughter products) and 2.5706 mCi (with daughter products).

A review of existing data for the waste activity determinations was conducted to estimate likely ranges of uncertainty or bias in the above method. Previous methods for estimating waste activity were often based on use of the “6CEn” rule-of-thumb (see section G.3.5(b)), to determine the equivalent unshielded point-source activity, based on a single reading of the exposure rate. (This rule relates exposure rate to activity, radionuclide, and distance. Knowing the exposure rate and assuming a value for the distance and the identity and fractions of the radionuclides, the activity can be estimated.) The use of the present MicroShield method, with 18 formalized measurements of the exposure rate from a waste box, is intended to reduce or eliminate the effects of two major sources of error (bias) in estimating waste activity from the rule-of-thumb method. The exposure rate from a box is considerably attenuated below the value that would be produced by an unshielded point source, and this leads to an underestimate of the activity. Further, individual high-activity pieces of waste may be near one or another face of the box, and give an exaggerated reading, which will imply a higher activity than is correct. These two effects tend to compensate each other, but by varying and unknown amounts.

While the ideal approach to validating the MicroShield method would be to apply it to a waste box that has been well-characterized by an independent method, it appears that such a box is not readily available. Therefore, results from the present method were studied to estimate the likely uncertainty in activity estimates. Since 18 measurements from each box are used in calculating an average reference activity, the variations in these measurements can be used to estimate the variability of the average. This variability can be used to estimate the uncertainty in the

reference activity, due to random variations in the contents of the boxes. Comparison of the reference activities with estimates made using the 6CEn rule can give an indication of the general accuracy of the MicroShield method.

In the MicroShield method, the measurements are made at 3 distances from each of the 6 faces of the box. The presence of a point source ("hot spot") near a face will lead to a high reading from that face, but a compensating lower reading from the opposite face. Thus, this method should reduce the bias effects resulting from distributions of material and activity that differ from the total and uniform distributions assumed in the calculation.

The performance of the MicroShield method was investigated by calculating the standard error of the mean for each of 85 previously characterized boxes, based on the variability observed in the 18 point estimates of activity. The waste activity estimates by MicroShield are compared to those using the 6CEn rule in Figure C-3. This figure shows that the MicroShield values are generally greater than the 6CEn estimates, as indicated by the preponderance of estimates lying above the diagonal line representing equal values. This results from the adjustment for attenuation of the gamma radiation by the waste material which is not accounted for in the 6CEn method. Those cases where the MicroShield estimate is below the line, i.e., less than the 6CEn estimate, occur when the point source hot spot effect is strong. On the average, the MicroShield method produces estimates that are 3.5 times greater than the 6CEn method. This relation holds over 34 orders of magnitude variation in activity, with no strong evidence of a trend.

The standard error of the mean for each MicroShield value is shown by the error bars on the points. In most cases, these uncertainties are reasonably small. The overall average standard error is 36%. Those cases where this uncertainty is large, on the order of 100%, are those in which the point source hot spot effect is unusually large.

In summary, this review shows that (1) the MicroShield method is superior (and more conservative) than the simple 6CEn rule because of its ability to average a greater amount of data per waste box, and to account for attenuation in the waste box itself, and (2) that the bias and uncertainty in waste activity estimates by the MicroShield method are not large.

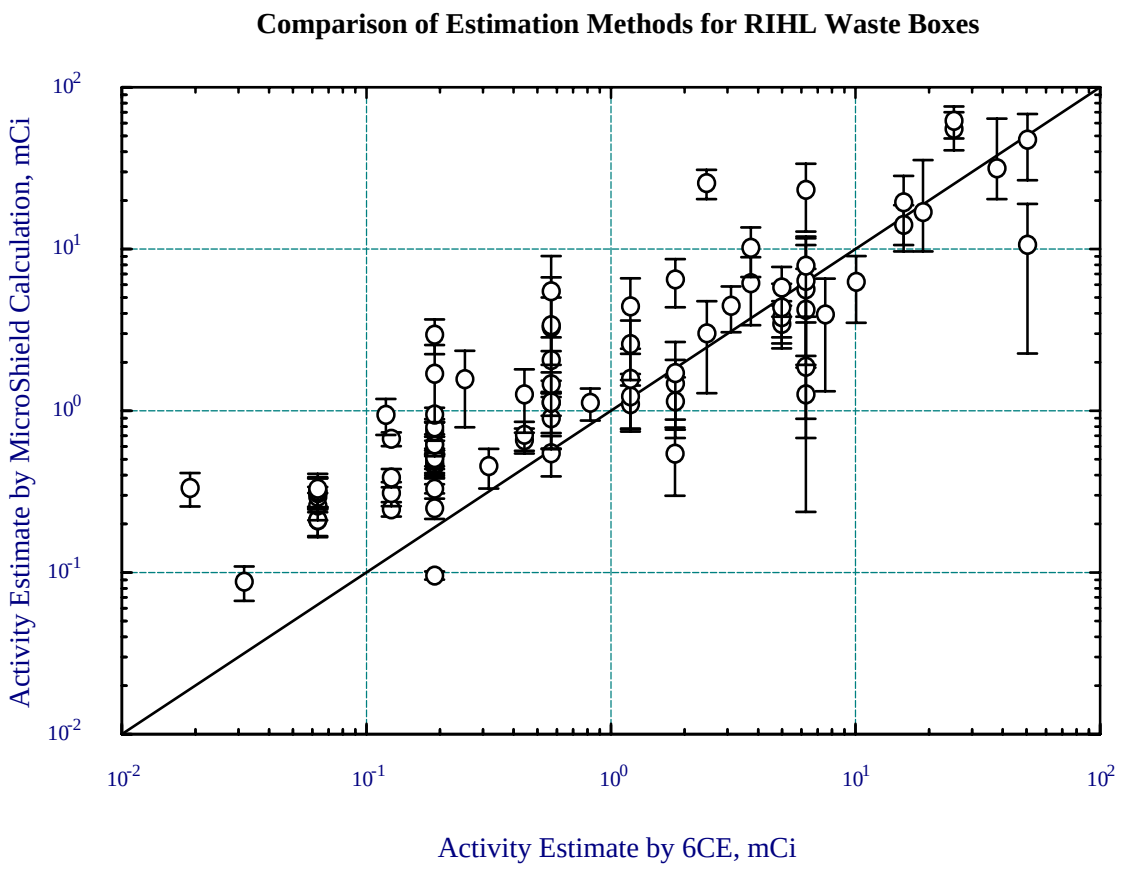


Figure G-3. Comparison of 6CEn and MicroShield Estimation Methods

G.3.2 Estimating Total Activity for Laundry and Filters

A rule-of-thumb relationship was developed by R. J. Tuttle for activity and mR readings on packages containing laundry or filters. The relationship is as follows:

Representing a laundry bag as a sphere 18 inches in diameter, containing 45 lb. of uniformly contaminated clothing, and calculating the exposure rate at an effective distance of 2 inches from the surface (to represent the geometry of the detector) gives the following conversion factors:

For MFP (Cs-137 with an equal activity of Sr-90)

mCi = 0.32 (mR/hr at contact)

For AP (Co-60)

mCi = 0.048 (mR/hr at surface)

A compromise rule-of-thumb has been adopted: divide the surface exposure rate (mR/hr) by 10 to estimate total unspecified activity (mCi).

This is a compromise between “divide by 3” for MFP, and “divide by 20” for AP. If the composition is known, these more accurate factors may be used.

For a hot spot, assuming that a point source is located approximately 1-1/2 in. in from the surface where the maximum reading is found, results in another factor of 10, due to the inverse distances squared. This gives the following conversions for a bag with a hot spot:

MFP mCi = mR/hr/30

Generic mCi = mR/hr/100

Co-60 mCi = mR/hr/200

The same factors are used for boxed filters, which are similar to the bagged laundry.

G.3.3 Estimating Activity in Evaporator Sludge

Assume waste sludge from RMHF HEPA filtered Evaporator is to be dried, packaged and shipped for offsite disposal.

A sludge sample is obtained and counted for alpha and beta activities. Alpha is not detected, therefore thorium, uranium, plutonium and americium isotopes are not likely to be present.

Samples are obtained for gamma spectroscopy. Spectroscopy results show 0.5 μCi of Co60 and 0.1 μCi of Cs-137 in the 500 gram sample. By inference, 0.1 Ci of Sr-90 can be assumed in the sample based on Cs-137. The concentration in the sludge is then:

0.5 μCi Co-60, 0.1 μCi Cs-137 and 0.1 μCi Sr-90 in 500 grams = 0.7 μCi/500 g

The weight of sludge placed in the disposal drum is 500 lb. The total activity in this package is then:

$$\text{Concentration} = \frac{0.7\mu\text{Ci}}{500\text{g}(1 / 453.6\text{ g / lb})} = \frac{0.7\mu\text{Ci}}{1.1\text{ lb}} = 0.64\mu\text{Ci / lb}$$

Total Activity = (0.64 μCi/lb) (500 lb) = 318 μCi = 0.3 mCi

G.3.4 Estimating Activity on Surface Contaminated Material

In Building 059, assume a contaminated plastic containment tenting is removed and placed in a waste box for transfer to the RMHF. The surface area of the plastic is about 1000 ft^2. Operations involved were activated concrete chipping and activated rebar cutting.

An instrument reading of the contaminated surface shows a typical value to be 25,000 dpm/100 cm^2 of beta activity. Converting this value gives:

$$\frac{25\ x\ 10^3\ dpm/100cm^2}{2.2\ x\ 10^6\ dpm/\ \mu Ci} = 11.2\ x\ 10^{-3}\ \mu Ci/100\ cm^2$$

The total observed activity is then:

$$((11.2\ x\ 10^{-3}\ \mu Ci/100cm^2)/100)*(9.29\ x\ 10^2\ cm^2\ /\ ft^2)(1\ x\ 10^3\ ft^2) = 104\ \mu Ci$$

The question at this point is whether there are other isotopes on the contaminated surface that were not observed but must be accounted for in the total activity estimate.

By referring to the Building 059 decommissioning project plan (ETEC 059-AN-0002, Rev. A), the major activation products can be identified. These are shown in Table G-1 of this document. Calculations based on data in the 059 project plan, including corrections for any significant isotope decay since 1989, show the following activity ratios in the significant isotopes present:

Approximate Ratios According to Source
(Based on Co-60 as 1)

Isotope	Energy β (Max.)	Gravel	Rebar	Coils	Vac Tank
Co-60	0.318 MeV	1	1	1	1
Ni-63	0.067 MeV no gamma	-	-	1	1
Fe-55	e capture 6 KeV X-ray	0.9	1.8	0.3	0.3
Eu-152	e.c. 72% β 28%, 1.48 MeV	2.4	-	-	-

Assuming concrete and rebar contributed equally to the amount of contamination deposited upon the plastic, the beta activity detected will have originated from Co-60 and Eu-152 and the activity values for each are:

let X = β from Co-60 which came from concrete

also X = β from Co-60 coming from rebar,
then 2.4 X (28%) from Eu- 152

Total β activity = X $Co\text{-}60_{concrete}$ + X $Co\text{-}60_{rebar}$ + 0.67 X_{Eu}
104 Ci of = 2.67 X

X = 39 μCi of Co-60 from concrete

X = 39 μCi of Co-60 from rebar

Total Co-60	=	78 μCi
Total Fe-55	=	$Fe\text{-}55_{concrete}$ + $Fe\text{-}55_{rebar}$
	=	0.9(39) + 1.8(39)
	=	35 + 70 = 105 μCi
Total Eu-152	=	$Eu\text{-}152_{concrete}$
	=	2.4 (X) = 2.4(39)
	=	93.6 μCi

G.3.5 Estimating Activity from Isotopes Distributed in Materials

(a) Sampling Data Are Available

Assume activated concrete and reinforcing bars from Building 059 are broken up and placed in a waste box for transfer to the RMHF. From there, final packaging and shipment preparations will be made for off-site disposal. Further assume representative samples have been taken of both materials for gamma spectroscopy. Sampling results indicate

Concrete (700 g)		Rebar (50 g)	
Co-60	4.2 μCi	Co-60	5 μCi
Eu-152 and 154	10 μCi		

Assume the concrete in total load weighs 5,000 lb.

From the sampling information, the activity from principal radioisotopes in the concrete can be calculated as follows:

$Ci_{concrete}$ = Co-60 + Eu-152 and 154 + Fe-55

Mass = (5,000 lb) (453.6 g/lb) = 2.27×10^6 g

$$Ci_{concrete} = (2.27 \times 10^6)\ \frac{(4.2 \times 10^{-3}\ \text{mCi})}{700\ g} + (2.27 \times 10^6\ \text{g})\ \frac{(10\ x\ 10^{-3}\ \text{mCi})}{700\ g} + \text{Fe-55}$$

Fe-55 = 0.9 (Co-60) = 0.9 (13.6) = 12.2 mCi

Fe-55 is based on 0.9 ratio of Fe-55/Co-60 shown on previous page.

$Ci_{concrete}$ = 13.6 + 32.4 + 12.2 = 58.2 mCi
Co Eu Fe

In order to calculate the radioisotope activity in the rebar portion of the package, it is necessary to estimate the relative weight fractions of rebar and concrete in the container. Assume this fraction has been estimated from the construction drawings to be 1 lb of rebar to 10 lb of concrete. Then, the rebar in the waste container is estimated to be:

$$(5{,}000\ \text{lb concrete})\left(\frac{1\ \text{lb rebar}}{10\ \text{lb concrete}}\right) = 500\ \text{lb of rebar}$$

The activity estimate for the rebar is:

$$\mu Ci_{rebar} = (500\ \text{lb})(453.6\ \text{g/lb})\ \frac{5\ \mu\text{Ci}}{50\ \text{g}} + \text{Fe-55 estimate}$$

= 22,680 μCi + Fe-55

22.7 mCi + Fe-55, From the preceding page, the ratio of Fe-55 to Co-60 is 1.8.

mCi_{rebar} = 22.7 + 1.8 (22.7) = 63.6 mCi

The activity by isotope for the <u>mixture</u> (concrete + rebar) then is:

Co-60 = 13.6 + 22.7 = <u>36.3</u> mCi
Fe-55 = 12.2 + 40.9 = <u>53.1</u> mCi
Eu-152, 154 = <u>32.4</u> mCi

(b) Sampling Data Are Not Available

Another less rigorous approach that could be taken for confirmation of an estimate based on sampling is to work from dose rate measurements made outside the waste container and make a number of assumptions regarding packing fraction, location of radioisotopes and shielding effects.

With the same conditions as above, but without sampling information, assume a portable gamma spectrometer has been used and has identified Co-60 as the only significant gamma emitter in the waste container. Assume the maximum radiation reading is 2 mR/h at the surface of the side of the container at mid-level.

The assumptions made are:

- The source of the radiation is a point source located 2 feet into the container.
- The only shielding material is concrete.
- The only radioisotope responsible for the gamma radiation is Co-60
- The concrete packing fraction is 75%, so 25% of the volume is air. The exposure rate at 1 foot is approximated by:

  mR/h at 1-foot = 6 CEn C = mCi
  E = energy (MeV)
  n = photons per transformation

  With those assumptions,

  $$\text{Without shielding, mR/h} = \frac{\text{surface reading}}{\text{transmission factor}}$$

  (Distance) (packing fraction) = effective shielding distance

  (2 feet of concrete) (0.75) = 1.5 feet effective shielding distance

  From Figure E-1 on page 32, factor at 18 inches ≅ 0.015

  $$\text{then, mR/h} = \frac{2\ \text{mR / h}}{0.015} = 133\ \text{mR/h at surface if no shielding were present}$$

  Correcting exposure rate from surface (which is 2 feet from source) to 1 foot behind surface,

  $$I_2 = I_1 \frac{(D_1)^2}{(D_2)^2} = 133 \frac{(2)^2}{(1)^2} = 532\ \text{mR / h at 1 foot}$$

From the exposure rate approximation,

mR/h at 1-foot = 6 CEn

for Co-60, mR/h = 6 (C) (1.25) (2)

solving for C,

$$C = \frac{(mR / h)}{(6)(1.25)(2)}$$

$$C = \frac{532}{15} = 35.5 \text{ mCi for Co-60 estimate}$$

(Actual conditions, using similar assumptions may not show such fortuitous agreement. The point source, source location, and packing fraction assumptions had significant bearing on the result.)

The activity ratios if available can then be used to estimate activities of the other radioisotopes present. Activity ratios can also be used to establish scaling factors for simple calculation of isotopic content using a single marker isotope (e.g., Cs-137 for RMHF materials or Co-60 for 4059 wastes). Where multiple shipments of waste may be generated from one lot of material, a computerized worktool system may be developed to simplify the calculation and documentation of the package characterization process.