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Nuclear Criticality Safety Analysis Summary Report: Concentration, Storage and Transfer Facilities and Operations (U)

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REVISION HISTORY

Revision	Author	Reason(s) for Revision	Date
0	D. A. Eghbali	Develop revision 0 to document the criticality safety of the Concentration, Storage, and Transfer Facilities using the relevant baseline NCSEs and NCSAs.	9/2008
1	M. H. Barnett	Revision 1 updates the document due to changes in several referenced NCSEs.	12/2009
2	N. P. Devine M. H. Barnett	Revision 2 adds the NCSE for Salt Dissolution in Tank 37H and the NCSA for Tank 49H Macro Batch 3 to ARP/MCU.	5/2010
3	M. H. Barnett	Revision 3 updates the document due to changes in N-NCS-H-00192 and adds two NCSAs.	12/2010
4	N. P. Devine	Revision 4 updates the NCSE for Tank 50 Valve Box Transfers and added an NCSA for Tank 49 Macro Batch 4.	7/2011
5	M. H. Barnett	Revision 5 updates the NCSASR to incorporate various NCSAs, the Chemical Cleaning NCSE, the revised ARP/MCU NCSE, and N-NCS-H-00234, and to add clarification to the N-NCS-H-00152 discussion and N-NCS-H-00132 barriers.	3/2012
6	M. H. Barnett	Revision 6 updates the NCSASR to incorporate various NCSAs and revisions to the F and H-Area Chemical Cleaning NCSEs.	5/2012
7	M. H. Barnett	Revision 7 updates the NCSASR to incorporate the ARP/MCU NCSE revision.	7/2012
8	R. L. Reed	Revision 8 updates revision numbers of several references and the ARP/MCU NCSE revision incorporating use of the Next Generation Solvent (NGS). Minor editorial updates. Revision bars indicate changes.	3/2013
9	M. H. Barnett	Added N-NCS-F-00137, N-ESR-G-00007, and Rev. 1 of the CSPDD. Incorporated changes in N-NCS-H-00132. Incorporated N-DCF-H-00054 which was for N-NCS-H-00192.	6/2013
10	M. H. Barnett	Incorporate the revision to N-NCS-H-00192 to allow for filter-only operations at ARP/MCU. Added new NCSAs.	1/2015
11	M. H. Barnett	Incorporate new revision of N-NCS-H-00192.	3/2015
12	M. H. Barnett	Incorporate new revision of N-NCS-H-00192 and N-NCS-H-00210.	2/2016

ABBREVIATIONS/ACRONYMS

AC	Administrative Control
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARP	Actinide Removal Process
BA	Bounding Assumption
CCP	Corrosion Control Program
CSC	Criticality Safety Committee
CSL	Criticality Safety Limit
CST	Concentration, Storage and Transfer
CSTF	Concentration, Storage and Transfer Facilities
CSTO	Concentrate, Storage & Transfer Operations
DCP	Double Contingency Principle
DDA	Deliquification, Draining and Adjustment
DID	Defense-in-Depth
DSA	Documented Safety Analysis
DU	Depleted Uranium
DWPF	Defense Waste Processing Facility
DWS	Dissolution Water Skid
ECP	Enrichment Control Program
EFQP	Evaporator Feed Qualification Program
EOI	Element of Incredibility
ESP	Extended Sludge Processing
ETP	Effluent Treatment Project
FGE	Fissile Gram Equivalent
FTF	F-Area Tank Farm
GPE	General Purpose Evaporator
HDB	H-Area Diversion Box
HEU	High Enriched Uranium
HTF	H-Area Tank Farm
HLW	High Level Waste
HLWD	High Level Waste Division
HPT	H-Area Pump Tank
ITP	In-Tank Precipitation
LPPP	Low Point Pump Pit
LWPT	Late Wash Precipitate Tank
MCU	Modular Caustic Side Solvent Extraction Unit
MST	Monosodium Titanate
NAS	Sodium Aluminosilicate
NaTPB	Sodium Tetraphenylborate
NCS	Nuclear Criticality Safety
NCSA	Nuclear Criticality Safety Assessment
NCSASR	Nuclear Criticality Safety Analysis Summary Report
NCSE	Nuclear Criticality Safety Evaluation
NGS	Next Generation Solvent

PDDs	Program Description Documents
PPT	Precipitate Pump Tank
PRFT	Precipitate Reactor Feed Tank
RBA	Restricted Bounding Assumption
RPM	Routine Process Monitoring
SAC	Specific Administrative Control
SCOM	Sludge Carryover Minimization
SEFT	Strip Effluent Feed Tank
SPT	Sludge Pump Tank
SRR	Savannah River Remediation LLC
SRS	Savannah River Site
SRTC	Savannah River Technology Center
TCP	Transfer Control Program
TSR	Technical Safety Requirement
TTJ	Telescoping Transfer Jet
URS SMS	URS Safety Management Solutions LLC
URS PS	URS Professional Solutions LLC
VB	Valve Box
WAC	Waste Acceptance Criteria
WSMS	Washington Safety Management Solutions, LLC
WSRC	Washington Savannah River Company

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1.0 INTRODUCTION

1.1 Objective

This Nuclear Criticality Safety Analysis Summary Report (NCSASR) summarizes the Nuclear Criticality Safety Evaluations (NCSEs) and Nuclear Criticality Safety Assessments (NCSAs) for Concentration, Storage and Transfer Facilities (CSTF). This NCSASR will be the technical criticality safety reference for the Documented Safety Analysis (DSA, Ref. 1) and the Technical Safety Requirements (TSRs, Ref. 2) that capture Nuclear Criticality Safety (NCS) barriers from applicable baseline NCSEs/NCSAs (see Table 1). This NCSASR is intended to be a "linking document" between the DSA and TSRs and the applicable NCSEs/NCSAs. This NCSASR simplifies the revision process to the DSA/TSRs when the baseline NCSEs are revised, often without a need to change the DSA/TSRs unless the revisions to the NCSEs result in a change to the barriers that are specifically included in the DSA or TSRs. Therefore, at a minimum this NCSASR must be revised along with the CSTF DSA annual update. Each time an NCSE or NCSA is originated or revised for CSTF, this NCSASR may be revised for corresponding revisions per the discretion of the CSTF Chemistry/Criticality Lead. Descriptions of processes, facilities, and equipment are provided in Chapter 2 of the CSTF DSA (Ref. 1).

1.2 Scope

The scope of this NCSASR is limited to the storage and processing of liquid radioactive wastes in the CSTF. CSTF includes the following facilities:

- F-Area Tank Farm (FTF),
- H-Area Tank Farm (HTF),
- 299-H Decontamination/Maintenance Shop,
- 242-16H Evaporator
- 241-96H Actinide Removal Process (ARP) Facility, and
- Modular Caustic Side Solvent Extraction Unit (MCU).

Note that the ETP (Effluent Treatment Project) is not a part of the CSTF; however, ensuring the criticality safety basis for ETP is part of the CSTF criticality safety responsibility.

Per the CSTF DSA (Ref. 1), the 242-1F Evaporator and the 242-1H Evaporator are inactive process areas; therefore, no criticality assessments/evaluations are needed. The addition of waste into or waste transfers through (transfers out are permitted) the inactive locations are prohibited. Criticality safety concerns for facilities other than those listed here are outside the scope of this NCSASR. The baseline NCSEs/NCSAs for the CSTF are listed in Table 1. NCSEs/NCSAs that are no longer applicable are listed in Appendix 1. The analyses of the operations in those NCSEs/NCSAs are potentially useful in future evaluations or assessments.

Table 1. Baseline NCSEs and NCSAs for this NCSASR

Document No.	Title
Generic Evaluations	
N-NCS-H-00134 (Ref. 3)	NCSE: Minimum Safe Gadolinium to Plutonium Ratio in an Infinite System
N-NCS-H-00172 (Ref. 4)	NCSE: Minimum Safe Gadolinium to Fissile Mass Ratio in an Infinite System
N-NCS-H-00152 (Ref. 5)	NCSE: H-Canyon Waste Stream Poisoning with Fe and Mn
N-NCS-H-00174 (Ref. 6)	NCSE: Safe Neutron Poison Ratios for Neutron Poisons Commonly Found in Saltcake
N-NCS-H-00034 (Ref. 7)	NCSE: Minimum Safe Ratios of Inherent Saltcake Metals to Uranium
N-ESR-G-00007 (Ref. 99)	NCSA: Sorption of Fissile Material onto Sodium Aluminosilicates
Waste Storage And Transfer	
N-NCS-H-00132 (Ref. 8)	NCSE: Tank 50 Valve Box Transfers
N-NCS-H-00165 (Ref. 9)	NCSE: Tank 50 Solids
N-ESR-G-00006 (Ref. 10)	Resolution of Actinide Loading in Tank 50 Solids and Extension to the Rest of the Tank Farm
N-ESR-H-00005 (Ref. 11)	NCSA: H Canyon HEU Transfer to Tank 50
N-ESR-H-00011 (Ref. 12)	NCSA: Transfer of Tank 23 and Tank 49 Supernate into Tank 50
WSRC-TR-2003-00169 (Ref. 13)	NCSA: Tank 48
WSRC-TR-2003-00343 (Ref. 14)	NCSA: H-Canyon Neptunium Solution Transfer to HLW Tank Farm
Salt Dissolution	
N-NCS-H-00151 (Ref. 15)	NCSE: Tank 41
N-NCS-H-00164 (Ref. 16)	NCSE: Salt Dissolution Process of F-Area Tanks 1, 2, and 3
N-NCS-H-00184 (Ref. 17)	NCSE: Tank 41 Dissolved Salt Solution Transfer to Tank 49
N-NCS-H-00234 (Ref. 18)	NCSE: Salt Dissolution in Tank 25
N-NCS-H-00141 (Ref. 54)	NCSE: Analysis of Inherent Saltcake Metals to Uranium in Tank 37H (U)
N-NCS-H-00255 (Ref. 84)	NCSE: Salt Dissolution in Tank 37H
N-NCS-H-00260 (Ref. 86)	NCSA: Tank 41H Saltcake Dissolution Using DWPF Recycle Material (U)
N-ESR-G-00008 (Ref. 19)	NCSA: Generic Saltcake Mining For Samplers
WSRC-TR-96-0336[1] (Ref. 20)	NCSA for Tank 38H Salt Dissolution
WSRC-TR-2004-00017 (Ref. 21)	NCSA: Incidental Salt Dissolution in Tank 27
N-NCS-F-00128 (Ref. 80)	NCSA: Saltcake Dissolution during Slurry Operations in Tank 4F
N-NCS-H-00275 (Ref.92)	NCSA: Saltcake Mining at Tank 41H C1 Riser for Transfer Pump Installation (U)
N-NCS-H-00281 (Ref. 95)	NCSA: Saltcake Dissolution during Slurry Operations in Tank 13H
N-NCS-H-00282 (Ref. 96)	NCSA: Bulk Saltcake Dissolution in Tank 10H (U)
N-NCS-F-00137 (Ref. 98)	NCSE: Salt Dissolution in Tank 33F (U)
N-NCS-H-00290 (Ref. 102)	NCSA: 2014 Tank 37H Saltcake Dissolution
Waste Evaporation	
N-NCS-H-00180 (Ref. 22)	NCSE: Operation of the 2H Evaporator System
N-NCS-H-00287 (Ref. 101)	NCSA: 2013 Chemical Cleaning of 2H Evaporator (U)
Actinide Removal	
N-NCS-H-00192 (Ref. 23)	NCSE: Actinide Removal Process and Modular CSSX Unit
N-NCS-H-00210 (Ref. 24)	NCSE: MST in Tank 49
N-NCS-H-00269 (Ref. 89)	NCSA: Tank 49H Macro Batch 4 to ARP/MCU
Sludge Heel Removal	

[1] Note that Revision 4 of WSRC-TR-96-0336 did not supersede Revision 3 and Revision 3 is the applicable document. Revision 4 of WSRC-TR-96-0336 is no longer applicable and it is listed in Appendix 1.

Document No.	Title
N-NCS-F-00122 (Ref. 26)	NCSE: Tanks 4, 5, 6 and 8 Chemical Heel Removal
N-NCS-H-00274 (Ref. 91)	NCSE: Tanks 9 through 15 Chemical Heel Removal
N-NCS-H-00279 (Ref. 97)	NCSA: Tank 12H Chemical Heel Removal (U)
Aluminum Dissolution	
N-NCS-H-00211 (Ref. 27)	NCSE: Aluminum Dissolution in Sludge

1.3 Definitions

Several of the definitions are not consistent with the current revision of SCD-3 (Reference 30), as only recent documents were written to the latest revision of SCD-3. The decision was made not to update the definitions as the majority of the baseline documents were not written to the current revision of SCD-3.

Administrative Control (AC) – A control that relies on the actions, judgment, and responsibility of people for its implementation. Because administrative controls are human-based, and therefore subject to error in application, they are less desirable than engineered controls. These controls are sometimes accompanied or enhanced by equipment items that alert an operator to take action (Ref. 30).

Barrier – One, or more, or a combination of: passive or active engineered criticality controls, administrative criticality control(s), criticality safety defense-in-depth feature(s), assumption(s), or nature of the process considerations that prevent exceeding a CSL. Engineered and/or administrative criticality controls are used to demonstrate compliance with the Double Contingency Principle, or to demonstrate (alone or in combination with nature of process considerations) that a scenario is BEU. (Ref. 30).

Bounding Assumptions (BA) – Conditions assumed for those physical parameters of importance to criticality safety (e.g., thick water reflection) other than the parameter upon which a Criticality Safety Limit (CSL) is based. If a BA for a parameter is "unrestricted," it means that the most reactive conditions for the parameter are assumed. If a BA for a parameter is "restricted," potential scenarios involving violation of the BA should be addressed and controls implemented as necessary. BA's include initial design constraints and physical process conditions not frequently altered, not requiring operator action, and not requiring frequent validation (Ref. 30).

Credible – The attribute of being believable based on commonly acceptable engineering judgment. A credible event is an event expected to occur with a frequency greater than once in a million years (i.e. greater than 1.0×10^{-6} per year) as determined qualitatively by engineering judgment or more formally as a result of a quantitative analysis. Assigning a numerical probability that defines credible is not required in an NCSE (Ref. 30).

Criticality Safety Control (CSC) – Passive engineered equipment, active engineered equipment, and administrative requirements which maintain parameter(s) important to criticality safety within established limits with a given reliability (i.e., failure frequency). Criticality safety controls serve as barriers to a criticality accident (Ref. 30).

Criticality Safety Limit (CSL) – A value within which a parameter important to criticality safety is controlled at or less than the subcritical limit, to ensure that the subcritical margin is maintained for credible scenarios. If the CSL is set at a value less than the subcritical limit, an additional margin of safety exists. An operating margin must account for any process-related uncertainties such that operating limits are established and maintained below the CSL to ensure that the CSL is not violated (Ref. 30).

Design Feature –Devices or hardware features such as fixed geometry, fixed spacing, fixed size, etc., which serve to maintain criticality safety.

Element(s) of Incredibility (EOI) – The identified process conditions, assumptions, routine process monitoring, and normal procedures that exist for reasons other than criticality safety, and, by their

presence, render a specific criticality accident scenario incredible (some scenarios may be judged to be incredible by a combination of elements of incredibility and one or more criticality safety controls). Elements may be passive, active, administrative, or programmatic in nature (Ref. 30). Note: The term, EOI, is neither defined nor used in revision Reference 30 beyond Revision 25; they just refer to the listed items as barriers. However, as many NCSEs were written prior to the implementation of Revision 25, and use EOIs, the definition is left in this document.

Equivalent Plutonium, Fissile Gram Equivalent (FGE) Pu-239 – Mass of fissile/fissionable materials expressed as equivalents of Pu-239 when a mixture actually contains several nuclides (such as U-233, U-235, Pu-239, Pu-241, Cm-244 and Cm-247). Equivalencies are based on American National Standards Institute (ANSI)/American Nuclear Society (ANS) standards (Refs. 31 and 32).

Equivalent Uranium, FGE U-235 – Mass of fissile/fissionable materials expressed as equivalents of U-235 when a mixture actually contains several nuclides (such as U-233, U-235, Pu-239, Pu-241, Cm-244 and Cm-247). Equivalencies are based on ANSI/ANS standards (Refs. 31 and 32).

Incredible – The attribute of NOT being believable on the basis of commonly acceptable engineering judgment. An incredible event is an event expected to occur with a frequency less than once in a million years (i.e., less than 1.0×10^{-6} per year) as determined qualitatively by engineering judgment or more formally as a result of a quantitative analysis. Assigning a numerical probability that defines incredible is not required in an NCSE (Ref. 30).

Nature of Process – Inherent process conditions (form of fissionable material, distribution of fissionable material, chemical composition of fissile material, inherent density of fissionable material, and other such conditions) that contribute to the criticality safety of a process or operation. If used to justify incredibility alone, nature of the process means there are no events that could lead to the formation of a critical mass of fissionable material, and no criticality controls are needed on any parameter of the process to prevent a criticality accident. (Ref. 30).

Unmitigated (scenario) – A scenario for which no criticality controls or programs (whose primary purpose contributes to preventing or mitigating the event) are assumed to be in place to provide a reduction in the estimated criticality event frequency or consequences, but as-designed process conditions (e.g., flowsheet and materials) and programs otherwise required or included as part of the Safety Analysis (e.g., nuclear material accountability, safeguards and security, transfer control) are assumed to be in place for normal operation (Ref. 30).

Waste Acceptance Criteria (WAC) – Formal criteria that must be met by the waste generator before they can send waste to the CSTF. The WAC Program ensures that the composition of waste streams received into the CSTF is within analyzed limits.

1.4 Elements of Incredibility

There are process conditions, normal procedures, and programs in CSTF that exist for reasons other than criticality safety and, by their presence, render a specific criticality accident incredible (some scenarios may be judged to be incredible by a combination of elements of incredibility and one or more criticality safety controls). The identified CSTF EOI's are described below. Note: Several NCSEs do not use the term Elements of Incredibility, in accordance with Reference 30, but they credit many of the programs listed below.

1.4.1 CSTF Waste Acceptance Program

The CSTF WAC (Ref. 33) is a program that provides criteria that must be met by the waste generators before waste can be sent to the CSTF. The WAC ensures that the composition of waste streams received into the CSTF is within the analyzed limits. The WAC provides the required mass ratio of neutron

poisons to equivalent U-235 and Pu-239 to ensure that the waste received is inherently safe. The WAC also has limits on the uranium enrichment in the 2H Evaporator.

1.4.2 CSTF Corrosion Control Program

The safety function of the CSTF Corrosion Control Program (CCP) is to ensure that corrosion of safety related equipment in the tank farms is managed for both stress corrosion cracking and pitting, so as to preserve the integrity of each component for its service life. According to Reference 34, carbon steel exposed to alkaline solutions has a low general corrosion rate. The CCP requires minimum concentrations of inhibitor (hydroxide and nitrite) to be present for corresponding amounts of corrosive species in waste to prevent corrosion.

The CCP requires that the actual tank chemistry is obtained through periodic sampling and the sampling schedule is based on the tank chemistry/use of the tank as specified by the program. In addition, tank-to-tank transfers are evaluated to ensure chemistry is kept within limits. For water additions to waste tanks, the corrosion control program sets limits on how much water can be added to a waste tank over certain time intervals based on the concentration of the inhibitor in the tank

Note that this program does not necessarily but can, as directed by management, apply to tanks in the "Removed from Service" mode. This program does not apply to tanks that have been declared a "Closed" Tank.

The CCP is described in the CSTF Program Description Documents (PDDs, Ref. 34).

1.4.3 Transfer Control Program

The Transfer Control Program (TCP) is established to govern all the CSTF-initiated waste transfers and certain aspects of transfers associated with Tank 50. The TCP is credited in many NCSEs to prevent inadvertent waste transfers. The program has monitoring and material balance requirements for waste transfers to detect inadvertent transfer events. Prior to a transfer, an independent verification of the correct transfer path alignment is completed and procedures identify the functional equipment needed to stop transfers and siphons. During a transfer, if a missing waste volume is greater than 5,000 gallons, the affected waste transfer will be terminated immediately. Also, in the event of the material balance discrepancies being greater than 15,000 gallons, the affected waste transfer will be terminated immediately. The TCP is described in the CSTF PDDs (Ref. 34).

1.4.4 Evaporator Feed Qualification Program

The Evaporator Feed Qualification Program (EFQP) prevents the formation of sodium aluminosilicate (NAS) scale in the 242-16F and 242-25H evaporators thus eliminating the criticality concerns caused by the adherent sodium diuranate. The EFQP ensures that the total accumulated scale in the 242-16H evaporator is less than or equal to 200 gallons and evaporator feed enrichment is maintained such that evaporator pot criticality is not credible. The EFQP is described in the CSTF PDDs (Ref. 34).

1.4.5 Sludge Carryover Minimization Program

The Sludge Carryover Minimization Program (SCOM) implements programmatic controls that minimize the amount of sludge solids carried over in the feed streams to the Tank Farm Evaporators and the program defines the criteria to designate transfers as Sludge Slurry Transfers in support of the TCP. The program minimizes the accumulation of fissile material in the evaporators. The SCOM is described in the CSTF PDDs (Ref. 34).

1.4.6 Tank 48 Prohibition

Monosodium titanate (MST) was added to Tank 48 during the 1983 In-Tank Precipitation (ITP) process demonstration and during the 1995 Batch 1 ITP processing to adsorb strontium, but it also adsorbs fissile materials such as uranium and plutonium. Sodium tetraphenylborate (NaTPB) was added to Tank 48

during both the 1983 and the 1995 ITP processing. Additional NaTPB was added during the 1996 process verification testing. The ITP process was suspended in December 1997.

Currently, Tank 48 contains large quantities of cesium/potassium tetraphenylborate (KTPB), with virtually no decomposition evident (only a small quantity of benzene produced based on years of slurry pump operation) because of chemistry controls (free hydroxide and temperature) in place. Chapter 18 of the CSTF DSA (Ref. 1) provides an overview of the methods to prevent and mitigate hazards in Tank 48. Waste cannot be transferred in or out of Tank 48 to protect Tank 48 and other CSTF storage tanks.

1.4.7 Routine Process Monitoring

Routine Process Monitoring (RPM) is the conduct of operations practices. Equipment monitoring and management responsibilities are fundamental to operating a facility which meets the Department of Energy (DOE) expectations. Routine Process Monitoring includes but is not limited to the following activities:

- Maintaining awareness of the facility status at all times and ensuring that facility equipment is operated in accordance with written and approved procedures.
- Operations personnel respond to instrument indications and alarms (level, flow, temperature, etc.) until such indications and alarms are proven to be false. Procedure Manual 2S Conduct of Operations (Ref. 35) lists the requirements for Routine Process Monitoring.

The WSRC 2S Manual is the document which lists the Conduct of Operations requirements for CSTF.

1.4.8 Sampling Methodology

Each macro batch in Tank 49 will be qualified using the sample results for the source tanks contributing to the salt solution macro batch in Tank 49 and the characteristics of salt solution left in Tank 49 from the previous macro batch. The sampling and characterization of the solutions is performed in accordance with written procedures based upon the Sampling Methodology described in Reference 28, which specifies requirements for performing sampling activities in the CST Facilities, consistent with CST Documented Safety Analysis (DSA) requirements.

2.0 REQUIREMENTS DOCUMENTATION

This NCSASR is subject to the requirements specified in Section 6.4 of the Nuclear Criticality Safety Manual (Ref. 30).

3.0 METHODOLOGY

This NCSASR summarizes the criticality safety requirements and contingency analysis for CSTF operations. The criticality safety hazards associated with these operations were identified, evaluated/ assessed, and documented in the applicable baseline NCSEs/NCSAs, listed in Table 1. Refer to the specific NCSEs/NCSAs for further details.

The methodology from Reference 36 was used in the baseline NCSEs/NCSAs to identify the necessary robust barriers so that a criticality accident was deemed incredible. Reference 36 outlines the overall process and the criteria by which engineering judgments are made and provides guidelines for determining if the identified barriers are sufficient to deem it incredible for a scenario to lead to criticality.

The NCSEs/NCSAs for CSTF have been developed over many years and are not all consistent in the use and meaning of terminologies (e.g., barriers, controls, elements of incredibility, assumptions) relied upon to demonstrate that criticality in not credible for a facility and/or process. This NCSASR summarizes the NCSEs/NCSAs that are applicable to CSTF operations and uses consistent language to describe and categorize the process conditions and assumptions important to maintain the criticality safety basis for

CSTF. This NCSASR updates the language used in the older but still valid NCSEs/NCSAs to ensure a consistent interpretation of all the CSTF NCSEs/NCSAs.

As many of the NCSEs in this document were written prior to Revision 25 of SCD-3 (Ref. 30), the term Elements of Incredibility is still used in this document (see explanation in the definitions). In addition, newer NCSEs use the term "assumptions" instead of bounding, restricted, or unrestricted assumptions. The Electronic Consolidated Hazard Analysis Process (E-CHAP) methodology has been used for several NCSEs and is used for new and revised NCSEs.

Note that some of the baseline NCSEs identified numerous barriers due to the conservative implementation of previous revisions of Reference 30. Some NCSEs do not credit the nature of the process to prevent a criticality; therefore, the identified barriers in some of the NCSEs are conservative relative to those which are necessary to implement the methodology in the current revision of Reference 30 (i.e., the NSCEs may identify more barriers than would be required). Reference 30 and Reference 79 discuss the methodology of functionally classifying barriers based on frequency and consequence to be used on a forward-fit basis. Previous NCSEs were evaluated for functional classification based on the methodology that barriers for incredible scenarios did not require inclusion in the DSA/TSR. An implementation plan was written for Reference 79. [Ref. 93] Reference 79 has been updated to Revision 1 (Ref. 100). The revision does not impact the applicability of Reference 93 and a new implementation plan was not developed.

Using the methodology from Reference 30 and Reference 79 for functional classification, and the justification found in Reference 93 for NCSEs written prior to issuance of Reference 79, none of the barriers identified in this NCSASR are required to be included in the DSA or TSR for criticality safety purposes.

4.0 SUMMARY OF BASELINE EVALUATIONS

The objective of this section is to summarize the results of the baseline NCSEs/NCSAs listed in Table 1. This section provides the reader an understanding of the processes to which the applicable NCSE or NCSA is applied. No new requirements are delineated in this section. Specific details and the bases for the barriers are documented in the specific baseline evaluation/assessment being summarized.

4.1 Generic Evaluations

This section summarizes the baseline NCSEs that are generic evaluations. Generic evaluations are not for any specific process; therefore specific contingencies were not discussed in these NCSEs. Before limits derived in a generic NCSE are used for future process, the NCSE should be assessed for applicability to that process.

4.1.1 NCSE: Minimum Safe Gadolinium to Plutonium Ratio in an Infinite System

N-NCS-H-00134 (Ref. 3) determined the minimum safe ratio of gadolinium to plutonium in an infinite sludge medium. This poison (neutron absorber) ratio is the primary means of criticality safety control during neutralization of the plutonium solutions and during the processing of neutralized solutions.

The H-Canyon facility transfers plutonium-bearing solutions to CSTF. Generally, this is accomplished by adding neutron poisons to the plutonium solution, neutralizing the resulting mixture, and transferring the solution to the CSTF waste tanks. The mixture of plutonium and poison is processed through H-Tank Farm facilities and is eventually vitrified into glass logs in the Defense Waste Processing Facility (DWPF). Because of the large quantity of plutonium in the mixture, the plutonium is transferred to CSTF in large kg batches (1 to 10 kg). This is much larger than the normal batch size, and given the nature of the Canyon process, the use of neutron poisons is the only means of ensuring criticality safety for this activity.

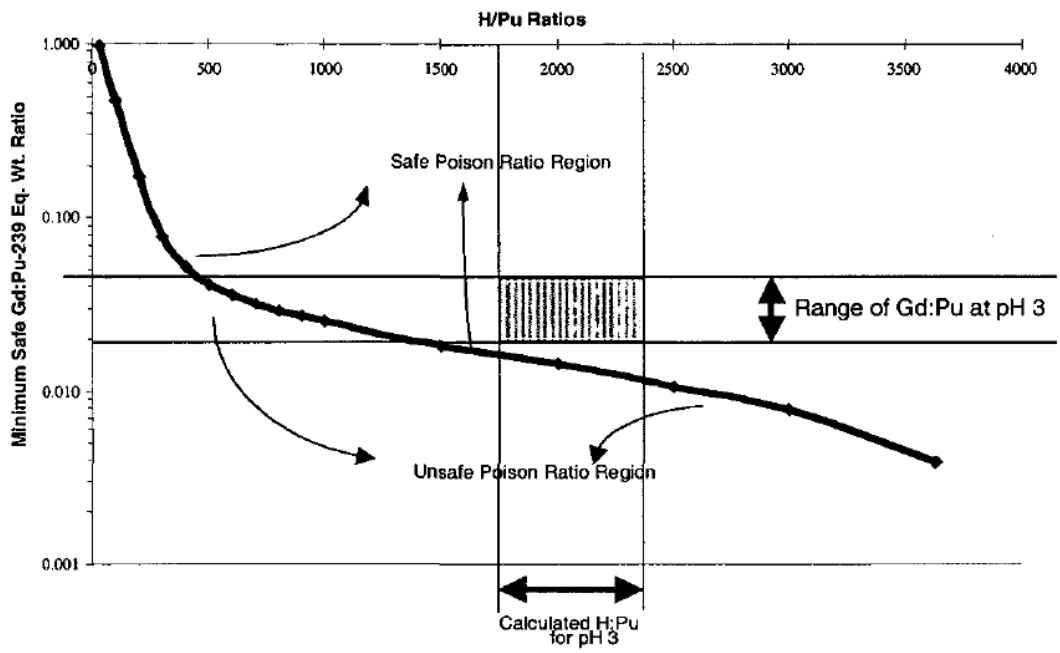
The normal poisons used in this process are iron (Fe) and depleted uranium (DU). However, for the future high mass transfers of plutonium, these poisons are not attractive due to the high poison to plutonium mass ratios that are required to obtain the necessary poison effect. The high poison mass dramatically increases the ultimate cost of disposing of the plutonium in DWPF glass logs. Gadolinium (Gd), a material with a better poison to mass ratio than Fe or DU, is needed to ensure the cost effectiveness of the process.

N-NCS-H-00134 concluded that an infinite mixture of gadolinium and plutonium (Pu) with a gadolinium-to-Pu-239 equivalent mass ratio greater than or equal to 1:1 and an H:Pu-239 ratio greater than or equal to 30 remains subcritical. Figure 4.1.1-1 illustrates the minimum safe Gd:Pu-239 ratio as a function of H:Pu-239. Increasing the H:Pu ratio decreases the reactivity of the system. N-NCS-H-00134 also demonstrated that slow precipitation of gadolinium during the neutralization of a plutonium/gadolinium mixture is not a criticality safety concern because sufficient gadolinium co-precipitates to protect the plutonium.

N-NCS-H-00134 requires that enough gadolinium is added so the Gd-154:Pu-239 equivalent mass ratio is at least 0.0213:1, the Gd-155:Pu-239 equivalent mass ratio is at least 0.1445:1, the Gd-156:Pu-239 equivalent mass ratio is at least 0.1999:1, the Gd-157:Pu-239 equivalent mass ratio is at least 0.1530:1, and the Gd-158:Pu-239 equivalent mass ratio is at least 0.2426: 1.

Figure 4.1.1-1 Safe Gd:Pu-239(eq) weight ratio as function of H/Pu-239

(Figure 6.3.1.1 from N-NCS-H-00134)



4.1.2 NCSE: Minimum Safe Gadolinium to Fissile Mass Ratio in an Infinite System

N-NCS-H-00172 (Ref. 4) expands the results of N-NCS-H-00134 (Ref. 3) to cover neutralized precipitate slurries containing uranium and gadolinium. N-NCS-H-00172 derives the minimum safe gadolinium to uranium mass ratio for an infinite system.

N-NCS-H-00172 documents a generic evaluation of neutralized precipitate slurries of uranium, plutonium, and gadolinium.

The conclusions of N-NCS-H-00172 are valid only if the following assumptions are met:

- The minimum Gd:Pu-239 mass ratio is maintained at ~1:1, and U-235 is treated as Pu-239 on a gram-for-gram basis.
- The H:fissile atomic ratio is maintained ≥ 30.
- Systems containing both Pu and U are mixed homogeneously.
- The U:Pu mass ratio is maintained ≥ 0:1 and ≤ 4.3:1.
- The U enrichment is maintained ≤ 75 wt% U-235.

N-NCS-H-00172 concludes the following:

- Fully neutralized precipitate slurries of U and Gd are significantly less reactive than corresponding slurries of Pu and Gd. In addition, any slurries containing mixtures of Pu and U will be mixed homogeneously, and no mechanism has been identified that would result in enhanced reactivity due to such homogeneous mixtures. Therefore, if the minimum safe Gd:Pu mass ratio derived in Reference 3 (Gd:Pu = 1) is applied to Pu-U-Gd slurries, with U-235 treated as Pu-239 on a gram-for-gram basis, subcriticality will be maintained.
- Mixed precipitate slurries containing Pu, U, and Gd remain subcritical during transient or incomplete neutralization.
- Self-shielding of UO_2, due to heterogeneities in the slurry, does not increase reactivity for particle sizes less than 100µm. Although self-shielding of Gd_2O_3 may result in small increases in reactivity, the magnitude of these increases does not affect the conclusions given above, for particle sizes less than 100µm. Note that per Reference 1, >90% of the salt particles in SRS wastes have sizes less than 60 µm.

N-NCS-H-00172 states "any NCSE that implements the limits derived herein must include suitable controls to ensure that neutralization runs to completion, or that the post-neutralization Gd:fissile ratio is at least 1:1."

4.1.3 NCSE: H-Canyon Waste Stream Poisoning with Fe and Mn (U)

N-NCS-H-00152 (Ref. 5) establishes the minimum safe mass ratios of two neutron absorbers, iron and manganese, to the fissile materials such that a mixture with equal to or greater than this combination mass ratio will always be safe assuming no other fissile or fissionable material is present.

The results obtained in N-NCS-H-00152 are limited to systems with H:fissile minimum atomic ratio greater than or equal to 50. N-NCS-H-00152 justified that HLW tanks cannot dry out beyond limits of the analysis without baking the sludge layer at high temperatures in a dry atmosphere.

N-NCS-H-00152 demonstrated the following:

- The safe Fe:U-235 mass ratio is 72:1 and the safe Mn:U-235 mass ratio is 14:1 for systems with a minimum H:U-235 of 50:1.
- The safe Fe/Mn combination mass ratios for U-235 systems can be calculated by the following equations with a minimum H:U-235 of 50:1:
  - For Fe Addition: [Fe:U-235] = (-5.8 x [Mn:U-235]) + 70 – [Known Fe:U-235]
  - For Mn Addition: [Mn:U-235] = (-0.17 x [Fe:U-235]) + 12 – [Known Mn:U-235]
- The safe Fe:Pu-239 mass ratio is 160:1 and the safe Mn:Pu-239 mass ratio is 29:1 for systems with a minimum H:Pu-239 of 50:1.

- The safe Fe/Mn combination mass ratios for Pu-239 systems can be calculated by the following equations with a minimum H:Pu-239 of 50:1:
  - For Fe Addition: [Fe:Pu-239] = (-5.7 x [Mn:Pu-239]) + 160 – [Known Fe:Pu-239]
  - For Mn Addition: [Mn:Pu-239] = (-0.17 x [Fe:Pu-239]) + 28 – [Known Mn:Pu-239]

Any uranium and plutonium systems that meet or exceed the above safe mass ratios will be subcritical and safe.

4.1.4 NCSE: Safe Neutron Poison Ratios for Neutron Poisons Commonly Found in Saltcake

N-NCS-H-00174 (Ref. 6) determined the safe mass ratios of neutron poisons (relative to U-235) commonly present in the High Level Waste (HLW) saltcake. The evaluation examined the effect four neutron absorbers (sodium, aluminum, chromium, and iron) have on uranium systems at varying uranium enrichments. Each poison mass ratio was increased until it became evident that a single poison could ensure criticality safety. Calculations were performed with uranium enrichments of 2, 10, 20, 30, 40, 50, 60, 66, 80 and 100 wt% U-235.

The following assumptions were made in this analysis:

1. The system will have a H:U-235 atom ratio greater than 50:l.
2. Salt dissolution does not change the poison to U-235 mass ratios.
3. Separation of insoluble poisons from the insoluble uranium is not possible.
4. The neutron absorbers can be shown to be mixed with the fissile uranium.
5. Tanks will be sampled by a method that will indicate the poison mass ratios present throughout the entire tank.

Appendices D-M in N-NCS-H-00174 show the tabulated results of the minimum safe mass ratios of neutron poisons commonly found in HLW salt tanks for various enrichments of uranium. Appendices D-M in N-NCS-H-00174 also show the minimum subcritical mass ratios (compared to U-235). Systems that have the combinations of poison (Na, Fe, Cr, Al) to equivalent U-235 ratios equal to or greater than the mass ratios indicated in Appendices D-M of N-NCS-H-00174 will be subcritical. If the system enrichment is between two tables, the table for the higher enrichment should be used to determine the minimum safe poison mass ratios.

4.1.5 NCSE: Minimum Safe Ratios of Inherent Saltcake Metals to Uranium

N-NCS-H-00034 (Ref. 7) provides the criticality safety basis for the dissolution of saltcake. The scope of this analysis is applicable to all saltcake storage facilities for the process of salt dissolution and saltcake removal. This analysis focused on uranium because its abundance is several orders of magnitude greater than plutonium in the saltcake.

N-NCS-H-00034 determined the safe mass ratios of inherent saltcake metal poisons to fissile material. These metal poison mass ratios are inherent in the typical HLW saltcake mixture; therefore, there is no discussion of the criticality concerns and credible variances in the contingency analysis.

For tanks dependent on the presence of neutron poisons to ensure criticality safety, N-NCS-H-00034 requires salt tank sampling prior to or during salt dissolution to obtain samples of the insoluble solids present in the saltcake. The sample results are needed to confirm the presence of sufficient neutron poisons in relationship to fissile material to maintain a subcritical condition.

N-NCS-H-00034 concluded that fissile material in an insoluble compound will be subcritical if the insoluble compound has a sodium to equivalent fissile uranium ratio greater than 150:1. An alternate minimum insoluble sodium and other insoluble metals to uranium ratio of sodium (90): chromium (10): zinc (10) to uranium (1) is also subcritical. Sampling of saltcake insoluble solids should continue to

confirm that the inherent chemistry of the deposited fissile material and associated metals is sufficient to maintain a subcritical condition for all operations of salt dissolution. Salt dissolution can proceed without specific criticality controls but with a characterization program that demonstrates the continued presence of adequate neutron poisons.

4.1.6 NCSA: Sorption of Fissile Material onto Sodium Aluminosilicates

N-ESR-G-00007 (Ref. 99) provides the criticality safety basis for the formation of sodium aluminosilicates (NAS) in systems that are not super-saturated with uranium or plutonium. The reference considered the amount of uranium and plutonium that is adsorbed on NAS and compared the resultant Na:Al:U-235 ratio to the requirement found in Reference 6. Reference 99 demonstrates that the NAS would have over 20 times the amount of Na necessary to render the uranium adsorbed on NAS infinitely safe.

4.2 Waste Storage and Transfer

This section summarizes the baseline NCSEs/NCSAs pertaining to the storage and transfer of HLW in the CSTF.

4.2.1 NCSE: Tank 50 Valve Box Transfers

N-NCS-H-00132 (Ref. 8) evaluated scenarios involving inadvertent transfers through the Tank 50 Valve Box into the ETP; such inadvertent transfers may result from planned transfers to Tank 50 and transfers inadvertently sent to the Tank 50 Valve Box.

The purpose of the ETP is to collect and treat process waste water, which may be contaminated with small quantities of radionuclides and process chemicals. The primary sources of waste water include F-Area and H-Area Canyon/Outside Facilities, F-Area laboratories, and Tank Farm evaporator overheads.

N-NCS-H-00132 identifies barriers/process conditions that ensure a criticality in the ETP associated with a transfer through the Tank 50 Valve Box is incredible. Moreover, N-NCS-H-00132 did not identify any credible criticality scenarios involving inadvertent transfers through the Tank 50 Valve Box into Tank 50.

Although the focus of N-NCS-H-00132 is on the assumptions, administrative controls, and design features that prevent mass accumulation in ETP, N-NCS-H-00132 states that a criticality is not possible due to fissile mass accumulation in ETP as the fissile material would be required to be concentrated and separated from associated poison (neutron absorber) to result in a critical configuration.

N-NCS-H-00132 states that criticality in the ETP, due to the transfers through the Tank 50 Valve Box, is not credible provided the barriers (see Section 5.2.1 of this NCSASR) identified are maintained. N-NCS-H-00132 identified no credible scenarios for accumulation of sufficient fissile material to form a critical configuration in Tank 50 due to transfers through Tank 50 VB into Tank 50.

4.2.2 NCSE: Tank 50 Solids

N-NCS-H-00165 (Ref. 9) evaluated the potential of a criticality accident due to the Tank 50 Solids. Tank 50 serves as the staging tank for decontaminated liquid waste for disposal in the Saltstone Facility. CSTF Operations personnel observed two mounds of solids at the bottom of Tank 50 in October 2002. These solids were sampled and determined to contain low enriched uranium. Therefore, concerns arose about the ability of these solids to collect fissile material, and whether the collection of fissile material could result in a criticality. N-NCS-H-00165 demonstrated that, based on the small amount of the fissile material that can be removed from solution by the Tank 50 solids, Tank 50 will remain subcritical during continued operations.

There are no barriers credited in N-NCS-H-00165.

4.2.3 NCSA: Resolution of Actinide Loading in Tank 50 Solids and Extension to the Rest of the Tank Farm

N-ESR-G-00006 (Ref. 10) assessed the fissile material loading on the solids that were discovered in Tank 50 that showed the potential to remove actinides from solution. The solids were tested by the Savannah River Technology Center (SRTC) and shown in N-ESR-G-00006 to be subcritical due to the water content and the low fissile loading. Additional testing was performed by SRTC to identify those components of the Tank 50 solids that were responsible for actinide removal. A few species were identified as removing actinides via an absorption mechanism. N-ESR-G-00006 assessed the criticality safety of the Tank Farm based on the information obtained from loading experiments and other available literature. Based on available data for Pu and U loading onto several different solid species, N-ESR-G-00006 concluded that there is not a criticality concern since the inherent poison-to-fissile ratio derived from the fissile material loading data was greater than the minimum poison:fissile mass values required to prevent criticality in all cases where there was a measurable loading. Np also had very low loadings, and was ruled out as a criticality concern due to the small amount of neptunium in the Tank Farm.

4.2.4 NCSA: H Canyon HEU Transfer to Tank 50

N-ESR-H-00005 (Ref. 11) assessed the transfer of Highly Enriched Uranium (HEU) to Tank 50. Unirradiated HEU material is dissolved in H-Canyon and transferred to Tank 50 for eventual disposition in the Saltstone Facility. Due to the unique transfer path and radionuclide content of this material, N-ESR-H-00005 reviewed several NCSEs (Refs. 8, 9, 15 and 72) to ensure the HEU stream was within the bounds previously established for the Tank 50 Valve Box, the Tank 50 Solids, the 2H Evaporator, and the Tank 41 Salt. N-ESR-H-00005 concluded that the barriers established in the NCSEs are sufficient to ensure criticality remains incredible within the Tank Farms and the ETP during the transfer of the HEU material from H-Canyon to Saltstone via Tank 50.

4.2.5 NCSA: Transfer of Tank 23 and Tank 49 Supernate into Tank 50

N-ESR-H-00011 (Ref. 12) assessed the criticality safety of the transfer of Tank 23 and Tank 49 material into Tank 50. The maximum concentrations of plutonium and uranium are less than the minimum solubility limits calculated for Tank 23, Tank 49, and Tank 50. Therefore, all three tanks are subsaturated with respect to uranium and plutonium. N-ESR-H-00011 demonstrated that fissile material precipitation will not occur during the mixing of Tank 49 and/or Tank 23 material with the Tank 50 material.

DWPF Recycle material was shown to be subsaturated with respect to plutonium, so there is no plutonium precipitation concern if the Tank 23 or Tank 49 comes into contact with DWPF Recycle during the transfers.

DWPF Recycle is supersaturated with respect to uranium so additional analysis was performed to assess potential fissile material precipitation if the Tank 23 or Tank 49 contents come into contact with DWPF Recycle during the transfers. N-ESR-H-00011 states that Tank 23 and DWPF Recycle contain DU (0.7% U-235). Therefore, any precipitation of fissionable material in Tank 50 will not be of a criticality concern as the overall enrichment will be less than the ANSI/ANS-8.1 subcritical enrichment limit of 0.93% U-235 (Ref. 31).

Any solids carried over from Tank 49 and/or Tank 23 are conservatively ignored since both contain significant poisons.

Tank 49 contains enriched uranium. The transfer path from Tank 49 to H-Area Diversion Box (HDB) 7 to Tank 50 does not have any pump tanks. There is a criticality concern if the Tank 49 enriched uranium material is inadvertently transferred to the H-Area Pump Tanks (HPT) 8 or 10 which contain DWPF Recycle supersaturated uranium during the vent/drain process. The inadvertent transfer may be due to valve misalignments or a leaky jumper in HDB-8 since the HDB-8 sump gets transferred to HPT-10.

Calculations in N-ESR-H-00011 have shown it would take Tank 49 material being transferred into the pump tank 630 times due to vent/drain operations during the Tank 49 to Tank 50 transfer in order to reach a critical mass. A conservative bound on the number of shutdowns for the Tank 49 to Tank 50 transfer is 60, which is significantly less than 630 shutdowns that would be required to reach the ANSI/ANS-8.1 U-235 subcritical mass limit of 700 grams (Ref. 31) assuming all the enriched uranium material that reached the pump tank precipitated. N-ESR-H-00011 concluded that the transfer of Tank 23 and Tank 49 supernate into Tank 50 is not a criticality concern.

4.2.6 NCSA: Tank 48 (U)

WSRC-TR-2003-00169 (Ref. 13) analyzed the criticality safety of material in Tank 48. A large amount of MST was added to Tank 48 at one time as part of the ITP Project. Based on this information, and the uniform distribution of MST in the tank, it is reasonable to judge that the 6.3 kg uranium in the tank is adsorbed on the entire MST mass (approximately 2300 kg) and therefore, because of the low fissile loading on MST, a critical configuration can not be assembled due to the accumulation of MST in one location. The presence of sodium tetraphenylborate (NaTPB) does not pose a criticality concern in Tank 48.

4.2.7 NCSA: H-Canyon Neptunium Solution Transfer to HLW Tank Farm (U)

WSRC-TR-2003-00343 (Ref. 14) assessed the criticality safety of receiving and processing H-Canyon neptunium solution in the CSTF. H-Canyon recovers neptunium from a high sulfate stream in existing process equipment. The neptunium will be transferred to the CSTF and eventually processed through the DWPF. WSRC-TR-2003-00343 assessed the transfer, storage, and sludge washing of the neptunium in the CSTF. The processing of the neptunium solution through DWPF is outside the scope of WSRC-TR-2003-00343.

The projected amount of Np-237 to be received/disposed of in the CSTF is much less than 60 kg. This mass is well below the ANSI/ANS-8.15 subcritical mass limit for NpO_2 (with a water reflector) which is 140 kg of Np-237 (Ref. 32). The neptunium transfer will also contain small amounts of other fissile materials [uranium and plutonium that are poisoned in the H-Canyon to meet the requirements of the CSTF WAC (Ref. 33)]. Despite the high Np:fissile materials ratio in the waste, the CSTF WAC requires neutron poisons in a waste stream to poison the fissile constituents.

The assessment concluded that the transfer of the proposed neptunium solution from H-Canyon to the CSTF will remain subcritical in any configuration (including possible settling in the pump tank).

4.3 Salt Dissolution

This section summarizes the baseline NCSEs/NCSAs pertaining to the salt dissolution operation in CSTF. It is noted that these NCSEs/NCSAs were performed for past activities and are listed in this document for completeness.

4.3.1 NCSE: Tank 41

N-NCS-H-00151 (Ref. 15) demonstrated that Tank 41 is inherently subcritical during interstitial liquid removal, dry saltcake storage, salt dissolution, and tank heel management for all identified normal and credible abnormal conditions. Heel removal is outside the scope of N-NCS-H-00151. N-NCS-H-00151 supersedes N-NCS-H-00147 (Ref. 55).

The dry saltcake present in Tank 41 has been shown to be inherently safe due to the presence of sufficient quantities of iron and manganese in the insoluble solids. In addition to the presence of iron and manganese, Tank 41 has also been shown to remain subcritical based on the low areal density of the fissile material.

There are no barriers required to ensure the criticality safety of Tank 41 during salt dissolution or tank heel management.

4.3.2 NCSE: Salt Dissolution Process of F-Area Tanks 1, 2, and 3 (U)

N-NCS-H-00164 (Ref. 16) evaluated the criticality safety of saltcake dissolution in F-Area Tanks 1, 2, and 3. The criticality safety of Tanks 1, 2, and 3 was evaluated by determining the uranium enrichment of the saltcake (soluble) as well as the uranium enrichment and poison-to-fissile mass ratios of the sludge layers.

It is assumed that most of the insoluble fissile content within the salt will settle to the bottom of the tank with the sludge layer during salt dissolution. Therefore, the postulated fissile content of the salt was added to the sludge layer. Tanks 1, 2, and 3 meet the required poison-to-fissile mass ratios for both iron (Fe:Pu-239 >160) and manganese (Mn: Pu-239 >29). Therefore, due to the adequate amount of neutron poisons present, the sludge in Tanks 1, 2, and 3 is considered subcritical.

N-NCS-H-00164 demonstrated that there were no normal or credible abnormal upset conditions in the salt dissolution process for Tanks 1, 2, and 3 that would result in criticality. No barriers were established as a result of N-NCS-H-00164 because the fissile component of the sludge cannot be separated from the neutron poisons under the alkaline chemistry of the HLW system.

There are no barriers required to ensure the criticality safety of salt dissolution process in Tanks 1, 2, or 3.

4.3.3 NCSE: Tank 41 Dissolved Salt Solution Transfer to Tank 49 (U)

N-NCS-H-00184 (Ref. 17) evaluated the criticality safety of Tank 49 during the receipt of Tank 41 dissolved salt solution. Due to the difference in the uranium concentrations between DWPF-originated waste and that typically found throughout the rest of the Tank Farms (including the salt solutions created in Tank 41) and expected solubility limits based on solubility models, it is postulated that the mixing of these wastes could result in some amount of uranium precipitation.

N-NCS-H-00184 demonstrated that Tank 49 will remain subcritical during and after the Tank 41 dissolved salt solution transfers into Tank 49 and all normal and credible upset conditions related to the receipt and storage of dissolved salt solution. The maximum mass of possible U-235 precipitate was shown to remain subcritical due to the low areal density of the material. The fissile-bearing Tank 49 supernate, both before and after the transfer of Tank 41 salt solution, was shown to remain subcritical due to the low concentration of soluble fissile materials. Both the areal density of the precipitated uranium and the concentration of fissile material in the supernate were shown to be well below the corresponding subcritical limits from ANSI/ANS-8.1 (Ref. 31). Therefore, a criticality in Tank 49 during the receipt and storage of Tank 41 dissolved salt solution is judged to be incredible.

There are no barriers relied upon to demonstrate the criticality safety of the Tank 41 to Tank 49 transfer of dissolved salt solution.

4.3.4 NCSE: Tank 25 Salt Dissolution

N-NCS-H-00234 (Ref. 18) evaluated the criticality of Tank 25 during salt dissolution and the subsequent transfer of the dissolved salt solution to Tank 41. Heel removal is outside the scope of this NCSE. N-NCS-H-00234 supersedes N-ESR-F-00036 (Ref. 37) and WSRC-TR-2007-00201 (Ref. 38).

N-NCS-H-00234 demonstrated that criticality is not credible for the activities during Tank 25 salt dissolution and the subsequent transfer of the dissolved salt solution to Tank 41 because the Fe:U-235(eq) mass ratio is greater than the safe mass ratio of 72:1.

The barriers credited in N-NCS-H-00234 are listed in Section 5.3.4 of this NCSASR.

4.3.5 NCSE: Salt Dissolution in Tank 37H

N-NCS-H-00255 (Ref. 84) used an areal density argument based on process knowledge of the Tank 37H contents to demonstrate subcriticality of the proposed saltcake dissolution in Tank 37H.

N-NCS-H-00255 applies to 3H Evaporator receipt tanks, or tanks of a similar saltcake-producing process in which only canyon-like wastes are involved, where saltcake in these tanks has an uranium concentration of no more than 11 mg/L and a plutonium concentration of no more than 0.12 mg/L.

The analysis approach in N-NCS-H-00255 can be applied to saltcake dissolution in any saltcake tank (including saltcake buildup in Tank 37H subsequent to completion of the proposed dissolution addressed in N-NCS-H-00255) as long as the assumptions used in N-NCS-H-00255 can be demonstrated to be applicable to that saltcake dissolution. The approach is dependent on being able to determine a valid and bounding estimate of the insoluble fissile material in the tank and demonstrate that the areal density of the insoluble fissile material in the tank is significantly less than the ANSI/ANS-8.1 single parameter subcritical areal density limit for U-235.

N-NCS-H-00255 concluded that a criticality in Tank 37H during the salt dissolution was not credible.

4.3.6 NCSA: Generic Saltcake Mining for Samplers

N-ESR-G-00008 (Ref. 19) assessed the generic mining of holes in the salt for samplers. In order to mine the saltcake, a sampling caisson is installed to the desired sample elevation and the salt inside this caisson is dissolved by adding a small amount of inhibited water via a wand. After the salt has been dissolved and radionuclides allowed to settle out sufficiently, the Impact Sample Thief and/or the Supernate Sampler is extended to the bottom of the caisson to collect the requisite sample(s). Samples removed from the tank are placed in a sample cask for transport to the laboratory. Flush water via a spray ring is used during removal of extension poles and caisson sections from the tank.

N-ESR-G-00008 is applicable for saltcake mining for samplers in all salt tanks except the drop tank of the 2H Evaporator that receives the DWPF recycle streams.

N-ESR-G-00008 concluded that criticality is not a concern during the mining of saltcake for samplers since there will not be sufficient mass of U-235 (eq) released during operations to exceed the ANSI/ANS-8.1 subcritical mass limit of 700 grams U-235 (Ref. 31).

4.3.7 NCSA for Tank 38H Salt Dissolution and NCSA for HPT-7 Drainback into Tank 38H

WSRC-TR-96-0336, Rev. 3 (Ref. 20) assessed the insoluble solids sample results from a previous Tank 38H sample (HTF-093), which indicate that there are sufficient neutron poisons (i.e., sodium and iron) present in the waste to maintain Tank 38H subcritical. The past sample analysis showed a sodium to uranium mass ratio of 6333 to 1 and an iron to uranium mass ratio of 355 to 1. However, the January 2001 sample (HTF-395/396) showed an insufficient sodium or iron inventory to serve as sufficient neutron poison to ensure subcriticality, therefore the uranium concentration was calculated in Reference 20.

The calculated fissile inventory of the top layer (i.e., 5 feet of supernate above the salt surface) of Tank 38H is ~13.2 kg U. Therefore, even if the top layer mass were to fill the entire tank, the areal density (~47 g/ft^2) would be ~8X less than the ANSI/ANS-8.1 single parameter areal density limit of 372 g/ft^2 (Ref. 31). The Tank 38H bulk salt inventory (i.e., the lower salt layer) is considered safe based upon the historical sample results in previous revisions of WSRC-TR-96-0336.

Note that Revision 4 (Ref. 39) did not supersede Revision 3. Revision 4 evaluated the HPT- 7 drainback into Tank 38H and this activity is complete. Revision 4 does not analyze any future HPT 7-10 transfers into the 242-16H evaporator system after the initial Tank 38H recycle has been transferred into Tank 43H. Therefore, Revision 4 of WSRC-TR-96-0336 is no longer applicable and it is listed in Appendix 1.

4.3.8 NCSA: Incidental Salt Dissolution in Tank 27 (U)

WSRC-TR-2004-00017 (Ref. 21) assessed the incidental salt dissolution in Tank 27 that results from the Tank 39 supernate transfer. Supernate from Tank 39 is transferred to Tank 27, which may dissolve the saltcake in Tank 27. Because the exact volume fraction of the saltcake that will be dissolved in Tank 27 was not known, WSRC-TR-2004-00017 conservatively assumed that the entire inventory of the tank content below the liquid level was saltcake, and the transfer supernate dissolved all of this saltcake and any exposed saltcake above the liquid level.

The potential for criticality during the bulk salt dissolution is that fissile material compounds, which precipitated when the supernate was concentrated by evaporation or cooling, will not dissolve as rapidly as the bulk saltcake and will accumulate on the saltcake surface. If sufficient neutron poisons are not maintained in this layer, a bulk agitation could redistribute the insoluble fissile material into a critical configuration.

WSRC-TR-2004-00017 concluded that the supernate in Tank 27 or the transfer supernate from Tank 39 did not pose a criticality concern because the concentration of fissile material in the supernate was far below the ANSI/ANS-8.1 concentration limit of 11.6 g U-235/L (Ref. 31). WSRC-TR-2004-00017 stated that, due to the highly enriched uranium content of Tank 39 supernate and subsequent salt formation by the 242-16F evaporator output in Tank 27, all further bulk salt dissolution of Tank 27 saltcake will require further assessment.

4.3.9 NCSA: Saltcake Dissolution during Slurry Operations in Tank 4F

N-NCS-F-00128 (Ref. 80) assessed the salt dissolution in Tank 4F that results during mixing of the tank contents for transfer to Tank 51H. The potential for criticality during a saltcake dissolution was that insoluble fissile material compounds, which precipitate when the saturated supernate is cooled, will not dissolve as rapidly as the bulk saltcake and may accumulate in one location. If sufficient neutron poisons are not present in the saltcake layer, a bulk agitation could redistribute the insoluble fissile material into a critical configuration.

N-NCS-F-00128 concluded that the slurry operations in Tank 4F did not pose a criticality concern. The conclusion was based on the low U-235(eq) concentration and mass within the saltcake and sufficient neutron poison (Fe) in the sludge layer to protect the fissile material.

4.3.10 NCSE: Analysis of Inherent Saltcake Metals to Uranium in Tank 37H (U)

N-NCS-H-00141 (Ref. 54) analyzed the bulk dissolution of radioactive saltcake waste material in Tank 37H down to the 150-inch level. The potential for criticality during a saltcake dissolution was that insoluble fissile material compounds, which precipitate when the saturated supernate is cooled, will not dissolve as rapidly as the bulk saltcake and may accumulate in one location. If sufficient neutron poisons are not present in the saltcake layer, a bulk agitation could redistribute the insoluble fissile material into a critical configuration.

N-NCS-H-00141 concluded that the salt dissolution to the 150 inch jet height level is inherently safe. The conclusion was based on the analyzed combined neutron poison to U-235 ratio.

4.3.11 NCSA: Tank 41H Saltcake Dissolution Using DWPF Recycle Material (U)

N-NCS-H-00260 (Ref. 86) assessed the salt dissolution in Tank 41H using DWPF recycle as the dissolving solution. The DWPF recycle may contain fissile material. Reference 15 (the Tank 41 NCSE) did not consider fissile material in the dissolving solution.

The NCSA demonstrated that the fissile material associated with up to 5,000,000 gallons of DWPF recycle would not result in Tank 41H having either the subcritical fissile areal density exceeded or the subcritical poison to fissile mass ratio less than that established in Reference 15.

4.3.12 NCSA: Saltcake Mining at Tank 41H C1 Riser for Transfer Pump Installation (U)

N-NCS-H-00275 (Ref. 92) assessed whether the mining of saltcake inside the Tank 41H C1 riser caisson, using hydro lancing, from the 102 inch level down to the tank bottom, is compliant with the Salt Dissolution in Tank 37H NCSE (Ref. 84).

The NCSA demonstrated that criticality is not a concern during the mining of saltcake since the mining operation uses a small volume of water (≤10,000 gal.) to dissolve saltcake. The dissolved saltcake will not release enough U-235 (eq) of fissile material to reach the subcritical mass limit of 700g for U-235. The process was determined to be compliant with Reference 84.

4.3.13 NCSA: Saltcake Dissolution during Slurry Operations in Tank 13H

N-NCS-H-00281 (Ref. 95) assessed whether the bulk sludge removal, along with the associated saltcake dissolution, in Tank 13H was within the criticality safety basis of N-NCS-H-00255 (Ref. 84) and N-NCS-H-00152 (Ref.5).

The NCSA demonstrated that the saltcake volume in Tank 13 was less than the volume determined in N-NCS-H-00255 to contain a mass of fissile material equivalent to the subcritical mass limit of U-235. The NCSA also demonstrated that the areal density of all fissile material (U-235 equivalent) in Tank 13 was below 10% of the subcritical areal density limit of 0.4 g/cm^2 for uranium-235, the criteria established in N-NCS-H-00255. Finally, the NCSA demonstrated that the poison:fissile ratio in the sludge after the addition of the fissile material contained in the saltcake was greater than the subcritical ratios required by N-NCS-H-00152.

4.3.14 NCSA: Bulk Saltcake Dissolution in Tank 10H (U)

N-NCS-H-00282 (Ref. 96) assessed whether the saltcake dissolution in Tank 10H is is within the criticality safety basis of N-NCS-H-00255 (Ref. 84). The NCSA only covers the bulk saltcake dissolution in Tank 10H

The NCSA demonstrated that the areal density of all fissile material (U-235 equivalent) in Tank 10 was below 10% of the subcritical areal density limit of 0.4 g/cm^2 for uranium-235, the criteria established in N-NCS-H-00255.

4.3.15 NCSE: Salt Dissolution in Tank 33F (U)

N-NCS-F-00137 (Ref. 98) analyzed the potential salt dissolution in Tank 33F during bulk sludge removal. The potential for criticality is that insoluble fissile material compounds will not dissolve as rapidly as the bulk saltcake and may accumulate in one location. Based on historical data for both the contents of Tank 33F and the Fe:U-235 ratio found in previous salt samples throughout the Tank Farm, N-NCS-F-00137 determined that salt dissolution in Tank 33F was not a criticality concern. This NCSE applies to salt dissolution in Tank 33F during bulk sludge removal or bulk salt dissolution. It only apples to bulk sludge removal using mixing pumps with associated salt dissolution and bulk salt dissolution using one of the following methods: Drain, Add, Sit, Remove; Modified Density Gradient (with and without interstitial draining), Semi-Continuous Dissolution, and Feed and Bleed. This NCSE does not apply to heel removal of material from Tank 33F.

4.3.16 NCSA: 2014 Tank 37H Saltcake Dissolution

N-NCS-H-00290 (Ref. 102) assessed whether the saltcake dissolution in Tank 37H is within the criticality safety basis of N-NCS-H-00255 (Ref. 84). The NCSA only covers the saltcake dissolution in Tank 37H.

The NCSA demonstrated that the assumptions in Reference 84 were valid and the areal density of all fissile material (U-235 equivalent) in Tank 37H was below 10% of the subcritical areal density limit of 0.4 g/cm^2 for uranium-235, the criteria established in N-NCS-H-00255.

4.4 Waste Evaporation

This section summarizes the baseline NCSE pertaining to the waste evaporation operation.

4.4.1 NCSE: Operation of the 2H Evaporator System

N-NCS-H-00180 (Ref. 22) evaluates the operation and the chemical cleaning of the 242-16H (or 2H) evaporator and demonstrate that there are no credible criticality concerns provided the bounding assumptions, design features, and administrative controls identified in Section 7 of N-NCS-H-00180 are met.

The scenarios evaluated in N-NCS-H-00180 are:

- Buildup of Saltcake on the 2H Evaporator Cell Floor
- Exceeding the Maximum Subcritical U-235 (eq) Enrichment
  - Inadvertent HEU Supernate Transfer into the 2H Evaporator System
  - Inadvertent Fissionable Material Addition to Tank 43 Supernate from Tank 43 Sludge Layer
  - Enrichment Exceeded in Waste Streams Added to Tank 43
- Criticality Concerns in Tank 38
  - Accumulation/Concentration of Fissionable Materials in Tank 38
  - HEU Addition to Tank 38 due to DB-7 Sump Transfers
- Exceed Fissionable Mass Limit During NAS Removal From the 2H Evaporator

N-NCS-H-00180 demonstrated that in the absence of NAS fissionable material buildup in the evaporator, the 2H Evaporator System remains subcritical for all credible fissionable material concentrations and enrichments in H-Tank Farm waste. Only when significant NAS fissionable material buildup has formed in the evaporator pot does criticality become a concern. Therefore, the evaporator pot needs to be inspected periodically to ensure that the NAS fissionable material buildup in the evaporator pot does not exceed the safe NAS mass/volume for chemical cleaning of the pot determined in N-NCS-H-00180.

The barriers specified in Section 7 of N-NCS-H-00180 are listed in Section 5.4.1 of this NCSASR.

4.4.2 NCSA: 2013 Chemical Cleaning of 242-16H Evaporator (U)

N-NCS-H-00287 (Ref. 101) evaluates the chemical cleaning of the 242-16H (or 2H) evaporator. Due to the build-up of NAS scale in the 2H Evaporator since the previous chemical cleaning in 2010, Liquid Waste decided in 2013 to chemically clean the evaporator to remove the NAS scale from the evaporator. NCSE N-NCS-H-00180 (Ref. 22) requires that an NCSA be written to demonstrate the chemical cleaning process would be in compliance with the NCSE. Reference 101 uses inspection reports and sample analysis to show that the 2013-planned cleaning of the 2H Evaporator would be within the limits of Reference 22.

4.5 Actinide Removal

This section summarizes the baseline NCSEs/NCSAs pertaining to the ARP/MCU Facility in CSTF.

4.5.1 NCSE: Actinide Removal Process and Modular CSSX Unit

N-NCS-H-00192 (Ref. 23) evaluated the criticality safety of the ARP/MCU Facility. The ARP includes both the 241-96H and 512-S facilities. N-NCS-H-00192 also evaluates the areas of the Low Point Pump Pit (LPPP) [e.g., the Precipitate Pump Tank (PPT) and the jumper connection between lines SDP1 and SDP25 in the Sludge Pump Tank (SPT) cell] related to the receipt of waste from the ARP/MCU.

N-NCS-H-00192 does not cover the criticality safety of the qualification tank or Tank 49. Tank 49 acts as the feed tank for the process and the qualification tank is a convenient control point for characterizing and approving a macro batch of salt solution as acceptable for ARP/MCU processing. Both are considered a storage tank and the criticality safety of storage tanks is based on the low concentration of fissile material and/or presence of sufficient neutron poisons, per the DSA (Ref. 1). However, barriers were identified in N-NCS-H-00192 to prevent adding uncharacterized waste to the tanks to protect the ARP/MCU process.

The 241-96H and Tank 49 valve boxes allow the various transfers needed to operate the ARP/MCU. There are no limits on the volume of the salt solution processed through the ARP/MCU provided the solution characteristics given in N-NCS-H-00192 are maintained.

The qualification tank supplying salt solution to Tank 49 will be sampled. After sample results have been obtained, the sample results will be reviewed by Tank Farm Engineering to verify the results are within expected values. After the sample results are determined to be acceptable, the results will be documented to demonstrate the soluble U-235 (eq) enrichment and the soluble uranium and plutonium concentrations in a macro batch are within the N-NCS-H-00192 specified values.

N-NCS-H-00192 has demonstrated that no credible potential for a criticality exists due to the processing of the identified fissile material through the ARP/MCU provided the barriers in Section 5.5.1 of this NCSASR are met.

Revision 12 added new scenarios to Reference 23. However, the new scenarios associated with CST operations/facilities (e.g., 96H Strike Tanks) were determined to have an unmitigated frequency of BEU due to the limited amount of MST in the Strike Tanks. The existing CST scenarios were not impacted. Therefore, in accordance with the guidance in Reference 100, the current functional classification of no control being elevated to the CST DSA/TSR is still appropriate. Revision 14 was primarily to acknowledge that the B3 riser pump could become the operational pump. As the pump used in Tank 49 does not impact the scenarios, the scenarios only had minor revisions.

4.5.2 NCSE: MST in Tank 49

N-NCS-H-00210 (Ref. 24) evaluated the criticality safety of a limited amount of MST in Tank 49 and determined the maximum amount of MST that can be present in a tank without adsorbing more than a subcritical fissile mass. N-NCS-H-00210 also evaluates inadvertent transfer of MST-containing salt solution from a strike tank in 241-96H to Tank 49 or any other Tank Farm waste tanks.

N-NCS-H-00210 demonstrated that the potential for a criticality as a result of inadvertent transfer of MST-containing material to Tank 49 is not credible. In addition, a criticality due to an inadvertent transfer of MST-containing material to a Tank Farm waste tank has been demonstrated to be not credible.

Barriers credited in N-NCS-H-00210 are listed in Section 5.5.2 of this NCSASR.

Revision 4 was a significant revision to Reference 24. No new scenarios were identified; however, the scenarios were significantly revised to address operational experience at ARP, new projected project lifetime, new MST adsorption data, and using a new riser pump. The scenarios were still determined to

have an unmitigated frequency of BEU due to the limited amount of MST in the Strike Tanks and the fact there is no mechanism in a waste tank to mix the MST with sufficient supernate and then bring it back into a critical configuration. Therefore, in accordance with the guidance in Reference 100, the current functional classification of no control being elevated to the CST DSA/TSR is still appropriate.

4.5.3 NCSA: Tank 49H Macro Batch 4 to ARP/MCU

N-NCS-H-00269 (Ref. 89) demonstrated that the Tank 49H Macro Batch 4A and 4B are compliant with the requirements of the current revision of the ARP/MCU NCSE (Ref. 23). N-NCS-H-00269 demonstrated that the soluble uranium and plutonium concentrations and the U-235(eq_sol) enrichment values are no more than the limits of 50 mg/L, 0.1 mg/l and 3 wt%, respectively. This was accomplished using analysis of samples taken from feed tanks to Tank 49H and process knowledge.

N-NCS-H-00269 includes the heel in Tank 49H remaining from Macro Batch 3. N-NCS-H-00269 refers to N-NCS-H-00192 (Ref. 23) to justify the criticality safety of any heel remaining in ARP/MCU.

N-NCS-H-00269 concludes that Macro Batch 4 met all the requirements of the current ARP/MCU NCSE(Ref. 23) and could be processed in the ARP/MCU.

4.6 Sludge Heel Removal

This section summarizes the baseline NCSEs pertaining to the sludge heel removal of HLW in CSTF.

4.6.1 NCSE: Tanks 4, 5, 6 and 8 Chemical Heel Removal

N-NCS-F-00122 (Ref. 26) evaluated the criticality safety of the proposed chemical heel removal of sludge in Tanks 4, 5, 6 and 8 (treatment tanks) and the subsequent transfer of the spent acid waste from the treatment tanks to Tank 7.

The scope of N-NCS-F-00122 includes:

- Chemical heel removal operation in Tanks 4, 5, 6 and 8 using oxalic acid and the subsequent transfer of the resultant sludge slurry or spent acid waste to Tank 7.
- Draining the spent acid waste from the transfer lines into the FPT-1.
- Inadvertent transfers of HLW and oxalic acid to/from Tank Farm Tanks.
- Inadvertent transfers of spent acid waste to FPT-1, VB 1-5, and FDB-2.
- Criticality safety of the waste in Tank 7 after all the waste from Tanks 4, 5, 6, and 8 is transferred to Tank 7.

The scope of N-NCS-F-00122 does not include decanting supernate from the treatment tanks prior to the chemical heel removal process. Such operations are covered under the CSTF DSA (Ref. 1).

N-NCS-F-00122 demonstrated that criticality is not credible for the activities involving the chemical cleaning of Tanks 4, 5, 6 and 8 and the subsequent transfer of the spent acid waste into Tank 7. This is due to the low concentration of fissile material in the sludge heel and the low fissile material areal density that would result in the event of precipitation. After chemical cleaning of the treatment Tanks 4, 5, 6 and 8, a trace amount of fissile material, void of any criticality concern, will remain in these tanks. The waste in Tank 7 remains subcritical due to the low concentration of fissile material, large amount of neutron poisons, and the large cross section area of the tank.

Barriers credited in N-NCS-F-00122 are listed in Section 5.6.1 of this NCSASR.

4.6.2 NCSE: Tanks 9 through 15 Chemical Heel Removal

N-NCS-H-00274 (Ref. 91) evaluated the criticality safety of the proposed Chemical Heel Removal of sludge in Treatment Tanks in H-Area and the subsequent transfer of the spent acid waste from the Treatment Tanks to the Receipt Tank.

The scope of this NCSE includes:

Chemical Heel Removal operation in Treatment Tanks (Type I and II tanks) using oxalic acid and the subsequent transfer of the resultant spent acid waste and flushes to the Receipt Tank.

Rinsing with water and stabilization of Treatment Tank after Chemical Heel Removal.

Addition of supernate from other CSTF Tanks or caustic solution to the Receipt Tank to neutralize the spent acid received from the Treatment Tanks.

Presence of acid waste leaks in valve boxes or diversion boxes during transfers.

Inadvertent transfers of High Level Waste (HLW) and oxalic acid to/from Tank Farm Tanks.

Waste in the Receipt Tank after receipt from Treatment Tanks.

The scope of N-NCS-H-00274 applies to the Chemical Heel Removal processing for Tanks 9 through 15. Some of the scenarios cover the specific transfer route for Treatment Tanks 11 and 12 to Receipt Tank 51. Also covered is the use of above grade hose-in-hose transfer lines and Tank 13 as an alternate receipt tank. For other transfer routes, an NCSA is required to show that the scenarios in this NCSE bound any scenarios of the new transfer route. The scope of this NCSE does not include the mechanical sludge removal operations or decanting supernate from the Treatment Tanks prior to the Chemical Heel Removal process. Such operations are covered under the CSTF DSA (Ref. 1).

This NCSE demonstrates that criticality is not credible for the activities involving the chemical cleaning of Tanks 9 through 15 and the subsequent transfer of the spent acid waste into a Receipt Tank. This is due to the low concentration of fissile material in the sludge heel and the low fissile material areal density in a tank that results if fissile material does not dissolve or is precipitated. In addition, a large amount of poison remains with the fissile material which dissolves and also with the fissile material that does not dissolve. After chemical cleaning of the Treatment Tanks, some fissile material may remain. This residual fissile material remains subcritical because its areal density in a Treatment Tank is substantially below the single parameter subcritical areal density limit and has poisons associated with it. The waste in Receipt Tank 51 remains subcritical due to the low concentration of fissile material, large amount of neutron poisons, and the large cross sectional area of the tank.

Barriers credited in N-NCS-H-00274 are listed in Section 5.6.2 of this NCSASR.

4.6.3 NCSA: Tank 12H Chemical Heel Removal (U)

N-NCS-H-00274 (Ref. 91) requires an NCSA to demonstrate the proposed chemical cleaning is compliant with the requirements of the NCSE.

N-NCS-H-00279 (Ref. 97) demonstrated that the fissile mass and poison:fissile ratios are within the requirements of N-NCS-H-00274. The NCSA also stated the transfer path was the same as the transfer path analyzed in the NCSE.

4.7 Aluminum Dissolution

This section summarizes the baseline NCSE pertaining to aluminum dissolution in sludge.

4.7.1 NCSE: Aluminum Dissolution in Sludge

N-NCS-H-00211 (Ref. 27) evaluated the impact on criticality safety of dissolution of aluminum in the sludge in any Tank Farm tank (hereinafter referred to as "dissolution tank") and of the resultant aluminum-laden supernate that will be decanted out of the dissolution tank to a staging tank until it is mixed with other supernate, processed prior to disposal at Saltstone, or sent to one of the evaporators for volume reduction.

The scope of N-NCS-H-00211 includes:

- Transfer of sodium hydroxide to the dissolution tank,
- Inadvertent transfer of sodium hydroxide to other Tank Farm tanks,
- Dissolution of aluminum in the dissolution tank,
- Decanting the aluminum-laden supernate to another tank (decant tank),
- Criticality safety of sludge during aluminum dissolution,
- Criticality safety of the aluminum-laden supernate,
- Inadvertent transfer of aluminum-laden supernate to other Tank Farm tanks or equipment,
- Inadvertent transfer from other Tank Farm tanks to the dissolution tank, and
- Criticality safety of sludge in the dissolution tank after the aluminum dissolution process.

Several incredible scenarios associated with aluminum dissolution activities, which would involve a sufficient volume of caustic/decant to be of a criticality concern, were identified and evaluated in N-NCS-H-00211. N-NCS-H-00211 demonstrated that assembling a critical configuration is not credible provided the requirements identified in Section 7 of N-NCS-H-00211 are met.

Barriers credited in N-NCS-H-00211 are listed in Section 5.7.1 of this NCSASR.

5.0 BARRIERS

This section summarizes the barriers required according to the baseline NCSEs listed in Table 1 for a criticality accident to be incredible during operations of the CSTF. This section categorizes the bounding assumptions, design features, and barriers (administrative controls and EOIs). For each EOI, the credited procedure or program is identified.

Note that the design features and administrative controls are also listed in Chapter 6 of the CSTF DSA (Ref. 1). BA's are not listed in the DSA because they are initial design constraints and physical process conditions not frequently altered, thereby requiring no operator action, and do not require validation. However, they are listed here for completeness.

5.1 Generic Evaluations

Generic evaluations are not for any specific process; therefore, specific contingencies were not discussed in these NCSEs. Thus there are no barriers in the generic evaluations. Before limits derived in generic NCSEs are to be used for a future process, the NCSE(s) should be evaluated for applicability to that process.

5.2 Waste Storage and Transfer

This section summarizes the barriers from NCSEs pertaining to the storage and transfer of HLW in CSTF.

5.2.1 NCSE: Tank 50 Valve Box Transfers (N-NCS-H-00132) (Ref. 8)

1. The maximum volume of an inadvertent transfer would be 15,000 gallons. [Bounding Assumption (BA)]

2. DWPF recycle transfer waste streams with U-235 (eq) enrichment in excess of 5.5 wt% are not sent through the Tank 50 VB. [BA]

3. Uranium and plutonium in DWPF recycle transfer salt solution have a solubility of no greater than 25 mg/L and 1.68 mg/L, respectively. [BA]

4. ETP includes a check valve (i.e., WEE-V-616) that prevents back flow into the Waste Concentrate Hold Tank and Waste Concentrate Tanks. [Design Feature]

5. Tank 50 Valve Box includes double valve isolation (i.e., WTS-V-282 and WTS-V-285). [Design Feature]

6. The ETP system includes the ETP Diversion Box valve (WEE-V-480). [Design Feature]

7. Valves in the H-Area Diversion Box 7 are connected in order to direct waste transfers to the desired location. Jumpers are fabricated stainless steel pipe segments, and are designed to complete a specific transfer route. [Design Feature]

8. The Tank 50 VB includes an isolation valve to isolate ETP from HDB 7 (i.e., WTS-V-274). [Design Feature]

9. Controls governing waste transfers shall be established addressing determination of transfer path. This evaluation shall identify the necessary process area(s) and leak detection locations needed to support the transfer. [administrative control][2]

10. Controls governing waste transfers shall be established addressing independent verification of correct transfer path alignment. Use of correct motive force shall be independently verified after initiating the transfer. [administrative control][2]

11. Controls governing waste transfers shall be established addressing determination of equipment needed to stop transfers and siphons. [administrative control][2]

12. Waste transfers shall be monitored periodically for indications of transfer events (i.e., Transfer Error, siphoning, leakage). Monitoring shall extend beyond the transfer path to locations (e.g., pump tanks, waste tanks) determined by evaluation. Waste transfers with the potential to transfer material in amounts greater than 15,000 gallons require material balances. [administrative control][2]

13. The ETP system shall be isolated from the Tank 50 Valve Box, except during ETP effluent transfers, by alignment of the Tank 50 Valve Box valves (i.e., WTS-V-282 and WTS-V-285) in accordance with the influent/effluent transfer procedure valve alignment checklist for Tank 50. [administrative control]

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[2] Per Reference 8, Barriers 9-12 only apply to DWPF Recycle transfers that go through HDB-7.

14. The isolation of the ETP system from flow through the Tank 50 Valve Box is ensured by the ETP Diversion Box valve (WEE-V-480). The HTF Shift Manager, First Line Manager, or Control Room Manager is to request ETP Personnel to ensure valve WEE-V-480 is closed in accordance with the governing transfer procedure for Tank 50, except during ETP transfers. [administrative control]

15. CSTF Operations personnel shall periodically (i.e., annually) ensure that the functionality of the ETP check valve (i.e., WEE-V-616) is demonstrated via an approved leak test procedure. [administrative control]

16. Where Tank 50 is neither the receipt nor the source tank, the isolation of the HDB-7 transfer from the Tank 50 Valve Box is ensured by the valves in HDB-7 by alignment of the valves in accordance with the transfer procedure valve alignment checklist. [administrative control]

17. The ETP system shall be isolated from HDB-7 transfers by alignment of the valves in the Tank 50 VB (i.e., WTS-V-274), (except for transfers between HDB-7 and the Tank 50 VB). The transfer procedure for transfers through HDB-7 is in place to prevent misalignment. [administrative control]

5.2.2 NCSE: Tank 50 Solids (N-NCS-H-00165) (Ref. 9)

There are no barriers specified in N-NCS-H-00165.

5.3 Salt Dissolution

This section summarizes the barriers from NCSEs pertaining to salt dissolution in CSTF.

5.3.1 NCSE: Tank 41 (N-NCS-H-00151) (Ref. 15)

There are no barriers identified in N-NCS-H-00151.

5.3.2 NCSE: Salt Dissolution Process of F-Area Tanks 1, 2, and 3 (N-NCS-H-00164) (Ref. 16)

There are no barriers identified in N-NCS-H-00164.

5.3.3 NCSE: Tank 41 Dissolved Salt Solution Transfer to Tank 49 (N-NCS-H-00184) (Ref. 17)

There are no barriers identified in N-NCS-H-00184.

5.3.4 NCSE: Tank 25 Salt Dissolution (N-NCS-H-00234) (Ref. 18)

1. The sample analysis of Tanks 25 and 41 contents is representative of the fissile inventory in these tanks. [BA]

2. The Corrosion Control Program shall maintain pH requirements in the alkaline region for CSTF Tanks (except for tanks in the Chemical Cleaning Mode). [CCP]

3. The following TSR controls are credited to support the incredibility arguments for inadvertent waste transfers in N-NCS-H-00234. The numbering and text are taken from the TSR (Ref. 2).

   5.8.2.21 Transfer Control Program [TCP]

A Transfer Control Program shall be established governing all FACILITY initiated WASTE TRANSFERS and transfers associated with Tank 50. The program shall at a minimum include the following attributes:

c. Monitoring and material balance requirements for WASTE TRANSFERS to detect transfer events shall be determined.

f. Prior to TRANSFER INITIATION, independent verification of correct TRANSFER PATH alignment shall be completed.

h. Prior to TRANSFER INITIATION, procedures shall identify the functional equipment needed to stop transfers and siphons. (SAC)

o. ACTUAL MISSING WASTE shall be less than or equal to 5,000 gallons. If ACTUAL MISSING WASTE is greater than 5,000 gallons, then the affected WASTE TRANSFERS shall be terminated immediately. (SAC)

p. Material balance discrepancies shall be less than or equal to 15,000 gallons. If material balance discrepancies are greater than 15,000 gallons, then the affected WASTE TRANSFERS shall be terminated immediately. (SAC)

5.3.5 NCSE: Salt Dissolution in Tank 37H (N-NCS-H-00255) (Ref. 84)

1. The saltcake that was formed from the evaporation in the 3H Evaporator is of canyon-like waste, and contains no DWPF recycle waste. [Assumption]

2. Carryover of sludge into the 3H Evaporator and Tank 37H is not a criticality concern due to the inherent sludge composition. The insoluble HLW sludge components contain sufficient quantities of iron, manganese, and other neutron absorbing elements to maintain any fissile material in the sludge inherently safe (subcritical). [Assumption]

3. For the purpose of estimating the total mass of insoluble fissile material in the saltcake, insoluble fissile solids in Tank 37H are considered to not leave the tank during the saltcake dissolution process. [Assumption]

4. The maximum concentration of uranium and plutonium in saltcake is 11 mg/L and 0.12 mg/L, respectively. [Assumption]

5. The maximum U-235 enrichment of the uranium in saltcake is 66 wt%. [Assumption]

6. Due to the nature of the process, criticality is not credible. Because the Transfer Control Program is credited with limiting undetected inadvertent transfers, it is included in the listing of requirements. The wording and numbering is taken from Reference 84.

   7.3.1 Tank Farm Transfer Control Program, administrative control 5.8.2.21 in the Liquid Waste (LW) Technical Safety Requirements (TSR), includes the following attributes:

   - Prior to transfer initiation, the transfer path including isolation points shall be determined.
   - Prior to transfer initiation, leak detection locations for the transfer path shall be determined.
   - Prior to transfer initiation, procedures shall identify the functional equipment needed to stop transfers and siphons.

- Affected waste transfers shall be terminated immediately if the actual missing waste is greater than 5000 gallons.
- If material balance discrepancies are greater than 15,000 gallons, then the affected waste transfers shall be stopped immediately.
- The maximum volume of an inadvertent waste transfer in CSTF that can remain undetected is 15,000 gallons.

5.3.6 NCSE: Analysis of Inherent Saltcake Metals to Uranium in Tank 37H (U) (N-NCS-H-00141) (Ref. 54)

1. The jet height is not lower than 150 inches from the bottom of the tank. [BA]
2. During evaporator operation where dilute (i.e., low sodium concentration) supernate can be added into the evaporator drop tanks, some amount of the insoluble neutron poisons (principally sodium) dissolve and the fissile material bound in the saltcake settles on top of the new saltcake layer. However, since the fissile material would uniformly settle, criticality safety is ensured by maintaining a sufficient insoluble neutron poison-to-fissile material ratio in this newly settled layer. Subsequent transfers out of the drop tank (e.g., to recycle resulting concentrated supernate back into the evaporator feed tank) would not cause this layer of fissile material on top of the saltcake to redistribute into a favorable geometry. [BA]
3. Additional salt tank sampling shall be required for confirmed bulk salt dissolution below 150 inches [Administrative Control]

5.3.7 NCSE: Salt Dissolution in Tank 33F (U) (N-NCS-F-00137) (Ref. 98)

1. The saltcake that is found in Tank 33F is of canyon-like waste, and contains no DWPF recycle waste. [Assumption]
2. For the purpose of estimating the total mass of insoluble fissile material in the saltcake, insoluble fissile solids in Tank 33F are considered to not leave the tank during the saltcake dissolution process. [Assumption]
3. The maximum concentration of uranium and plutonium in saltcake is 11 mg/L and 0.12 mg/L, respectively. [Assumption]
4. It is assumed there are 84 inches of saltcake below the sludge and 10 inches of saltcake above the sludge. [Assumption]
5. The maximum U-235 enrichment of the uranium in saltcake from H-Area is 66 wt%. [Assumption]
6. The maximum U-235 enrichment of the uranium in saltcake from F-Area is 0.72 wt%. [Assumption]
7. Prior to beginning the sludge removal process in Tank 33F, an engineering evaluation shall be written to demonstrate that the solids layer in Tank 33F is not above 117.7 inches. [Administrative Control]
8. Because the Transfer Control Program is credited with limiting undetected inadvertent transfers, it is included in the listing of requirements. The wording and numbering is taken from Reference 98.

   7.4.1 Tank Farm Transfer Control Program, administrative control 5.8.2.21 in the Liquid Waste (LW) Technical Safety Requirements (TSR), includes the following attributes: (CST)

   - Prior to transfer initiation, the transfer path including isolation points shall be determined. (SAC)
   - Prior to transfer initiation, leak detection locations for the transfer path shall be determined. (SAC)

- Prior to transfer initiation, procedures shall identify the functional equipment needed to stop transfers and siphons. (SAC)
- Affected waste transfers shall be terminated immediately if the actual missing waste is greater than 5000 gallons. (SAC)
- If material balance discrepancies are greater than 15,000 gallons, then the affected waste transfers shall be terminated immediately. (SAC)
- The maximum volume of an inadvertent waste transfer in CSTF that can remain undetected is 15,000 gallons.

5.4 Waste Evaporation

This section summarizes the barriers from the NCSE pertaining to waste evaporation in CSTF.

5.4.1 NCSE: Operation of the 2H Evaporator System (N-NCS-H-00180) (Ref. 22)

1. The maximum uranium content in the NAS fissionable material is 8.9 wt%. [BA]
2. The maximum uranium concentration in the supernate of the 2H Evaporator System is 250 mg/L. [BA]
3. The maximum uranium concentration in Tank 38 saltcake is 2 g/L. [BA]
4. The maximum volume of an inadvertent waste transfer to the 2H Evaporator System that can remain undetected is 15,000 gallons. [BA]
5. The maximum flow rate of incoming supernate to Tank 38 from the 2H Evaporator is 20 gallons per minute. [BA]
6. U-235 enrichment in some supernate other than the 2H Evaporator System supernate is 100 wt%. [BA]
7. All future modifications of the 2H Evaporator System components pertinent to criticality safety as discussed in Section 5 shall be evaluated prior to processing salt solutions. Specifically, modification to the steam tube bundle, warming coil, evaporator shell thickness, evaporator shell insulation material, or to the evaporator pot shall be evaluated for criticality safety prior to implementing the modification. The use of a different insulation material could produce better moderation than the water assumed to surround the evaporator in the current analysis. There are no criticality concerns about the thickness or condition of the current evaporator insulation material since calculations reported in Section 6.2 and Appendix F indicate that the presence or absence of the insulation does not significantly impact k_{eff}. [Design Feature]
8. The DSA/TSRs level monitoring program minimizes the waste leakage to the DB-7 sump. [EOI]
9. The maximum subcritical equivalent U-235 enrichment for the 2H Evaporator System is 5.5 wt%. [Administrative Control]

- The following equations from Reference 41 shall be used to determine the equivalent U-235 and equivalent uranium.

$$\text{U-235(eq)} = \text{U-235} + 1.4 \times \text{U-233} + 2.25 \times (\text{Pu-239} + \text{Pu-241})$$

$$\text{U (eq)} = \text{U} + 2.25 \times (\text{Pu-239} + \text{Pu-241})$$

- The plutonium content of the fissionable elements in the 2H supernate shall not exceed 2 wt%.
- Waste streams with U-235 (eq) enrichment in excess of 5.5 wt% shall not be added to the 2H Evaporator System. The CSTF Evaporator Feed Qualification Program shall ensure that the U-235 (eq) enrichment in the 2H supernate does not exceed 5.5 wt%.

Exceptions: Waste transfers from DB-7 sump to Tank 38 and from the 2H Evaporator sump to Tank 43 are exempted from this requirement.

10. For a waste transfer that could inadvertently be sent to the 2H Evaporator feed or drop tank, [EOI (TCP, EFQP)]

    - Transfer paths shall be set using approved procedures.
    - Transfer paths shall be verified by a second operator prior to the transfer.
    - During the transfer, the sending and receiving tank levels shall be checked every 30 minutes for the first two hours of the transfer.
    - Material balance shall be performed for transfers exceeding 15,000 gallons. The feed to the 2H Evaporator shall be secured if a transfer discrepancy exceeds 15,000 gallons.
    - Resumption of feed to the 2H Evaporator shall not begin until it is verified that the inadvertent transfer exceeding 15,000 gallons was not received in the 2H Evaporator feed or drop tank or a documented engineering evaluation has been performed demonstrating that the evaporator will remain subcritical with the additional material.

11. Tank 43 is the designated feed tank for the 242-16H Evaporator System. The use of any other tank as the feed tank is not addressed in this NCSE. [Administrative Control]

    - Influents to Tank 43 shall be positioned at least 36 inches above the sludge layer (e.g., side wall penetrations, downcomers, backflush valve, etc.).
    - A minimum of 24 inches of separation shall be maintained between the sludge level and the feed pump recycle discharge in Tank 43.
    - A minimum of 24 inches of separation shall be maintained between the sludge layer and the feed pump suction in Tank 43.
    - If there is an indication that the separation distances above are not met (e.g., turbidity measurement, suspended solids in samples, error in specification on drawings),
      - The feed to the evaporator shall be secured, but waste may be received in Tank 43,
      - The feed to the evaporator shall NOT be restored until:

        1) the minimum separation distance(s) is restored, and
        2) the turbid zone of suspended solids is demonstrated to be less than or equal to 5.5 wt% U-235 (eq) enrichment.

      - If the separation distances cannot be restored, or the turbid zone was determined to be greater than 5.5 wt% U-235 (eq) enrichment, a nuclear criticality safety evaluation shall be performed prior to restoring feed to the evaporator.
    - A minimum of 36 inches of supernate shall be maintained above the sludge layer in Tank 43.
    - Flygt mixers in Tank 43 shall be permanently disabled/de-energized.

12. Tank 38 is the designated drop tank for the 242-16H Evaporator System. The use of any other tank as the drop tank is not addressed in this NCSE. [Administrative Control]

    - Influents to Tank 38 shall be positioned at least 36 inches above the sludge layer (e.g., side wall penetrations, downcomers, backflush valve, etc.).

13. Sampling and associated analysis of supernate in Tanks 38 and 43 to determine the U-235 (eq) enrichment shall be performed every 26 weeks. [Administrative Control]

- Samples shall be taken at a minimum of two supernate depths.
- Sample results shall be available within 60 days of sampling. In the case of an extended outage, the 26 week sampling requirement may be deferred, but the sampling and analysis shall be completed and the results shall be available prior to restarting the evaporator.
- Sample results (accounting for measurement uncertainty) shall be reviewed by engineering for acceptability and consistency with past results and/or current process knowledge.
- If sample results indicate a U-235 (eq) enrichment greater than 5.5 wt%, the feed to the 242-16H evaporator shall be secured until a documented engineering evaluation demonstrates subcriticality.

14. Prior to transfer into Tank 43, the U-235 (eq) enrichment of a non-qualified waste batch shall be determined based on the sample analysis of individual waste streams making up the waste batch or based on the sample analysis of the blended waste batch. [Administrative Control]

    - Samples shall be taken at a minimum of two depths of the waste batch.

      Exception: For mixed tanks, one sample is sufficient to characterize the tank contents when performed in accordance with the Sampling Methodology document (Ref. 28).

    - Sample results (accounting for measurement uncertainty) shall be reviewed by engineering for acceptability and consistency with past results and/or current process knowledge.

15. The maximum subcritical NAS deposit for the cleaning process shall be determined from Tables 5.4.1-1 and 5.4.1-2 using the U-235 (eq) enrichment in the 2H Evaporator System and the maximum known equivalent uranium loading in the NAS deposit. [Administrative Control]

Table 5.4.1-1: Subcritical NAS Buildup Volume (gallon) for the 2H Evaporator
NAS Buildup Cleaning Process

Uranium Loading	Enrichment Wt% U-235 (eq)					
	≤ 0.96	> 0.96, ≤ 2	>2, ≤3	>3, ≤4	>4, ≤5	5.5*
2 wt% U(eq) in NAS	-	2,076	475	256	170	153
3 wt% U(eq) in NAS	-	1,363	312	168	111	100
4 wt% U(eq) in NAS	-	1,022	234	126	83	75
5 wt% U(eq) in NAS	-	818	187	101	67	60
6 wt% U(eq) in NAS	-	668	153	82	54	49
7 wt% U(eq) in NAS	-	573	131	70	46	42
8.9 wt% U(eq) in NAS	-	450	103	55	36	33
10 wt% U(eq) in NAS	-	394	90	48	32	29

* Calculated NAS mass limits based on 10 wt% below the corresponding values at 5 wt% U-235 (eq) enrichment. This is conservative based on Figure 22 of Reference 42.

Table 5.4.1-2: Subcritical NAS Buildup Mass (kg) for the 2H Evaporator NAS Buildup Cleaning Process

Uranium Loading	Enrichment Wt% U-235 (eq)					
	≤ 0.96	> 0.96, ≤ 2	>2, ≤3	>3, ≤4	>4, ≤5	5.5
2 wt% U(eq) in NAS	-	20,000	4,583	2,475	1,640	1,476
3 wt% U(eq) in NAS	-	13,333	3,055	1,650	1,093	984
4 wt% U(eq) in NAS	-	10,000	2,291	1,237	820	738
5 wt% U(eq) in NAS	-	8,000	1,833	990	656	590
6 wt% U(eq) in NAS	-	6,666	1,527	825	546	492
7 wt% U(eq) in NAS	-	5,714	1,309	707	468	421
8.9 wt% U(eq) in NAS	-	4,494	1,029	556	368	331
10 wt% U(eq) in NAS	-	4,000	916	495	328	295

16. The evaporator shall be inspected periodically to ensure the maximum subcritical NAS deposit for the cleaning process is not exceeded. The estimate of the NAS deposit accumulation shall account for measurement uncertainty. [Administrative Control]

    Using the NAS upper range accumulation rate of 15 gallons/month and the obtained maximum subcritical NAS deposit (in 15), the maximum time to inspection can be determined. Operations inspects for NAS deposit accumulation periodically at least once in the period between the evaporator start time and the maximum time to inspection. Note that the frequency of inspections is based on the evaporator operation hours, not the calendar hours.

17. Prior to introduction of a cleaning agent into the 2H Evaporator, two NAS deposit samples shall be taken and analyzed to determine the U-235 (eq) enrichment and the equivalent uranium loading. [Administrative Control]

    a. NAS deposit sample results shall be reviewed by engineering for acceptability and consistency with past results and/or current process knowledge.

    b. Using the obtained NAS deposit U-235 (eq) enrichment and the equivalent uranium loading (in 17), determine the maximum subcritical NAS deposit from Table 5.4.1-1 or 5.4.1-2.
    c. The estimated (accounting for the measurement uncertainty) NAS deposits inside the evaporator (obtained in 15) shall be less than the maximum subcritical NAS deposit obtained in 17.b.
    d. If the estimated (accounting for the measurement uncertainty) NAS deposits inside the evaporator (obtained in 15) exceeds the maximum subcritical NAS deposit obtained in 17.b, a documented engineering evaluation shall be performed to demonstrate subcriticality.

18. An NCSA shall be written for each evaporator cleaning cycle to demonstrate the NAS cleaning process will meet the requirements of this NCSE. [Administrative Control]

5.5 Actinide Removal

This section summarizes the barriers from NCSEs pertaining to actinide removal in CSTF.

5.5.1 NCSE: Actinide Removal Process and Modular CSSX Unit (U) (N-NCS-H-00192) (Ref. 23)

N-NCS-H-00192 covers facilities in both CSTF and DWPF. The NCSE lists each barrier and which facility is responsible for the barrier. Only those barriers for which CSTF are responsible are listed below.

1. There are no known mechanisms in Tank 49 or the qualification tank to separate the insoluble neutron poisons (e.g., Fe, Mn, Na, and Al) from the insoluble fissile material. [Assumption]
2. There are no known mechanisms in the process tanks in ARP to separate the insoluble neutron poisons (e.g., Fe, Mn, Na, Al, and Ti) from the insoluble fissile material. [Assumption]
3. Salt batch qualification samples are representative of the qualification tank contents. [Assumption]
4. Procedures specify a minimum contact time between the MST and salt solution in the Strike Tanks of 8 hours. [Assumption]
5. Procedures specify the addition of sufficient MST to a Strike Tank to result in 0.2 g/L MST in the Strike Tank for optimum adsorption of soluble strontium. [Assumption]
6. Each Strike Tank has an overflow volume of 6,221 gallons, and a heel volume of 1,440 gallons. [Design Feature]
7. Each SSRT has a volume of 8,206 gallons, a working volume of ~6,300 gallons, and an overflow volume of 7,755 gallons. Each SSRT has an internal sump with a volume of 9.2 gallons and the transfer pump suction is located no more than 5" above the sump bottom. [Design Feature]
8. The SSFT has a volume of 8,201 gallons, a working volume of ~6,300 gallons, and an overflow volume of 7,751 gallons. The SSFT has an internal sump with a volume of 4.7 gallons and the transfer pump suction is located no more than 5" above the sump bottom. [Design Feature]
9. An extraction contactor has an internal volume of less than 7.6 gallons. [Design Feature]
10. A scrub, wash, or strip contactor has an inner diameter of 5 inches. [Design Feature]
11. The outer diameters of the extraction aqueous heat exchanger and the extraction solvent heat exchanger are 6 inches or less. [Design Feature]
12. The CSS line from the LWHT is the only fissile material transfer line to MCU. [Design Feature]
13. The DSS hydraulic accumulator has a volume of 80 gallons. [Design Feature]
14. The DSS coalescer has an inner diameter of 4 inches or less. [Design Feature]
15. The DSS pre-filter has an inner diameter of 8 inches or less. [Design Feature]
16. The DSS Decanter has a volume of 1,025 gallons. [Design Feature]
17. The DSSHT has a volume of 8,208 gallons. The DSSHT has an internal sump with a volume of 11.2 gallons and the transfer pump suction inlet is one inch above the sump bottom. [Design Feature]
18. The CDT transfer pump suction inlet is 1.75 inches above the tank bottom. [Design Feature]
19. The following programs are credited in the Contingency Analyses of Reference 23 to help demonstrate incredibility of criticality in ARP/MCU. The numbering and text is taken from Reference 23. [Programmatic]

    7.3.1 Tank Farm Transfer Control Program, administrative control 5.8.2.21 in the Liquid Waste (LW) Technical Safety Requirements (TSR), includes the following attributes:

    - Prior to transfer initiation, the transfer path including isolation points shall be determined.
    - Prior to transfer initiation, leak detection locations for the transfer path shall be determined.
    - Prior to transfer initiation, procedures shall identify the functional equipment needed to stop transfers and siphons.
    - Affected waste transfers shall be terminated immediately if the actual missing waste is greater than 5000 gallons.
    - If material balance discrepancies are greater than 15,000 gallons, then the affected waste transfers shall be stopped immediately.
    - The maximum volume of an inadvertent waste transfer in CSTF that can remain undetected is 15,000 gallons.

7.3.2 Sampling Methodology

7.3.3 Prohibited Activities (Specific Administrative Control)

Sludge slurry transfers into Tank 49 are prohibited.

20. Documentation shall be written for each salt macro batch. The document shall determine the soluble uranium and plutonium concentration and the U-235(eq_sol) enrichment for each macro batch in Tank 49 using [Administrative Control]:

    i. Sample analysis results for the qualification tank,

    ii. U-235(eq_sol) enrichment (wt%) = 100% x (U-235 + 1.4 * U-233) ÷ U

    Where U is the summation of all uranium isotopes (e.g., U-233, U-234, U-235, U-236, U-238).

21. a. The document shall demonstrate that the soluble uranium and plutonium concentration and the U-235(eq_sol) enrichment in the batch salt solution are less than or equal to 50 mg/L, 0.3 mg/L, and 3 wt%, respectively. [Administrative Control]

    b. The document shall demonstrate that the free hydroxide concentration of qualified salt solution in the qualification tank is a minimum of 1.68 molar. [Administrative Control]

22. The nitric acid available to MCU is ≤ 15 wt%. [Administrative Control]

5.5.2 NCSE: MST in Tank 49 (U) (N-NCS-H-00210) (Ref. 24)

1. Bounding Loading of Uranium and Plutonium on MST [BA]

Actinide	mg Actinide/g MST	wt% Actinide/g MST
Plutonium	48.2	4.82
Uranium	242	24.2

2. The maximum number of transfer line flush occurrences containing MST into Tank 49 and any Tank Farm waste tank for the remaining projected lifetime of the ARP/MCU facility is 50.[3] [BA]

3. The maximum volume of MST-containing material in each transfer from 241-96H to 512-S is 4,000 gallons. This does not apply to flushing activities of the Strike Tanks in 96H where no MST is added to the tank and the only MST is from the residual tank heel. [BA]

4. There are no known mechanisms to load the MST with fissile material and then accumulate the fissile material adsorbed on MST into a critical configuration. [BA]

5. Isolation of the Tank 49 B3 and B5 riser transfer paths can be accomplished by any of the following methods: 1) installation of a blank in the transfer line, 2) use of double isolation valves, or 3) use of a single leak checked isolation valve.* [Design Feature]

   *valves also include an element of administrative barriers

6. Prior to initiating a transfer from 241-96H to 512-S, the transfer path shall be isolated from Tank 49 and HDB-7. A second qualified person shall verify the transfer path is isolated.* Isolation shall

[3] The assumption about maximum number of transfer line flushes only refers to the flushing of the transfer line from 241-96H to 512-S.

remain for the entire 241-96H to 512-S MST-containing material transfer operation. Isolation shall consist of: [Administrative Control]

- Closing WTS-V-2013 and WTS-V-2014 in the Tank 49 Valve Box (B5 riser) and equivalent isolation method for B3 riser when installed to accommodate the B3 riser pump (e.g., blank, double valve isolation, single leak-checked valve),
- Closing WTS-V-1 and WTS-V-2 in the 241-96H Valve Box,
- Closing WTS-V-165 in HDB-7.

  * does not apply if a blank is used for isolation

7. Operations shall limit the MST-containing material transfer volume out of 241-96H to values corresponding to the U-235 (eq) enrichment in Tank 49 and the MST concentration in the strike tank as shown in Table 5.5.2-1. [Administrative Control]
8. During transfers from 241-96H to 512-S, the LWPT level shall be monitored to ensure the material containing MST is received in the LWPT. If waste is not received within the time period expected after transfer initiation (see Table 5.5.2-1), the transfer from 241-96H shall be stopped immediately. Transfer from 241-96H may resume after determination that intended transfer conditions exist. [Administrative Control]
9. Following an inadvertent transfer of material containing MST from 241-96H to a waste tank, transfer into and out of the receiving tank shall be stopped until the impact to criticality safety is evaluated. [Administrative Control]

Table 5.5.2-1. Maximum Allowed MST Mass/Volume based on U-235 (eq) Enrichment
(Bolded values correspond to volumes less than 4,000 gallons, the nominal transfer batch from 241-96H to 512-S)

U-235 Enrichment (wt. %)	Safe Grams MST	Transfer Volume (gal) to reach safe loading limit -Tank 49 with Initial MST (loading at 1139g) at 0.2 g MST/L	Transfer Volume (gal) to reach safe loading limit -Tank 49 with Initial MST (loading at 1139g) at 0.4 g MST/L	Transfer Time at 0.2 g MST/L min at 150 gal/min for column 3 volume	Transfer Time at 0.4 g MST/L min at 150 gal/min for column 4 volume
100	2384	**1645**	**822**	11	5
66	3213	**2740**	**1370**	18	9
45	4093	**3902**	**1951**	26	13
35	4706	4712	**2356**	31	16
25	5535	5808	**2904**	39	19
3	9041	10439	5219	70	35

5.6 Sludge Heel Removal

This section summarizes the barriers from the NCSEs pertaining to sludge heel removal in CSTF.

5.6.1 NCSE: Tanks 4, 5, 6 and 8 Chemical Heel Removal (U) (N-NCS-F-00122) (Ref. 26)

1. This NCSE assumes that the drawings and operating sequences for Tanks 4 and 8 will be similar to Tanks 5 and 6. [BA]
2. The maximum volume of an inadvertent transfer is 15,000 gallons (Ref. 1). [BA]
3. Design features credited for nuclear criticality safety in this evaluation are the FPT-1 and FPT-2 diameters and volumes. These features are credited in determining the amount of fissile material that can be present within the tank. All future modifications of the process flow pertinent to criticality safety shall be evaluated prior to processing waste[4]. [Design Feature]
4. Prior to starting the chemical heel removal process for a specific treatment tank, the fissionable material inventories (including enrichment) of the treatment tank shall be evaluated to ensure that they are less than or equal to the projected inventories shown in Table 5.6.1-1. (uranium enrichment ≤ 0.96 wt%). [Administrative Control]

Table 5.6.1-1. Initial Sludge Volume and Fissionable Material Inventory

		Tank 4	**Tank 5**	**Tank 6**	**Tank 7**	**Tank 8**
Sludge Volume (Bounding)	gallons	10,000	10,000	6,583	102,000	4,510
U-233	kg	0.000	0.000	0.000	0.000	0.000
U-234	kg	0.000	0.000	0.000	0.000	0.000
U-235	kg	6.474	32.467	10.830	99.764	2.033
U-236	kg	0.000	0.000	0.000	0.000	0.000
U-238	kg	1,791	4,943	2,603	20,024	601
Total U	kg	1,797	4,976	2,614	20,124	603
U-235 Enrichment	wt%	0.36	0.65	0.41	0.50	0.34
Pu-238	kg	0.000	0.000	0.000	0.000	0.000

[4] Modifications to the FPT-1 and FPT-2 diameters and volumes are the only modifications pertinent to the criticality safety for N-NCS-F-00122.

Pu-239	kg	1.732	5.197	1.442	21.039	0.928
Pu-240	kg	0.106	0.339	0.137	1.545	0.059
Pu-241	kg	0.011	0.039	0.021	0.194	0.006
Pu-242	kg	0.001	0.006	0.016	0.100	0.006
Total Pu	kg	1.850	5.581	1.616	22.878	0.999
$Fe(OH)_3$	kg	3,836	16,218	3,551	67,765	2,109
MnO_2	kg	1,169	3,656	838	10,735	196
$NaAlO_2$	kg	1,262	2,362	2,321	26,604	1,171
$Ni(OH)_2$	kg	1,962	3,675	2,514	14,981	200

5. Prior to transferring oxalic acid to the treatment tank, operators shall set the oxalic acid transfer path using operating procedures to ensure that the oxalic acid will be transferred to the intended treatment tank (Tank 4, 5, 6 or 8). Prior to transferring oxalic acid from the tanker to the treatment tank, a second person shall verify that the oxalic acid transfer path is connected to the intended treatment tank. [Administrative Control]
6. Before and after the completion of each batch transfer (one tanker truck) of oxalic acid, the level in the receiving treatment tank (Tank 4, 5, 6 or 8) shall be checked. If there is no increase in the intended receiving treatment tank level, additional oxalic acid transfers shall be stopped until an investigation determines why the level did not increase. [Administrative Control]
7. The following TSR controls are credited to support the incredibility arguments for inadvertent waste transfers in N-NCS-F-00122. The text is taken from the DSA (Ref. 1).

5.8.2.13 Corrosion Control Program

Programmatic controls shall be implemented to ensure that waste chemistry and characteristics are controlled so that corrosion will not impact the ability of safety related equipment to perform its function when required. This program shall include as a minimum, limits on pH, OH^-, NO_2^-, NO_3^-, SO_4^{-2}, and Cl^- as necessary to account for equipment corrosion.

5.8.2.21 Transfer Control Program

A Transfer Control Program shall be established governing all FACILITY initiated WASTE TRANSFERS and transfers associated with Tank 50. The program shall at a minimum include the following attributes:

- Prior to TRANSFER INITIATION, determine the following: (SAC)
  - TRANSFER PATH (including isolation points)
  - LEAK DETECTION LOCATIONS for the TRANSFER PATH
- Monitoring and material balance requirements for WASTE TRANSFERS to detect transfer events shall be determined.
- Prior to TRANSFER INITIATION, independent verification of correct TRANSFER PATH alignment shall be completed.
- Prior to TRANSFER INITIATION, procedures shall identify the functional equipment needed to stop transfers and siphons. (SAC)
- ACTUAL MISSING WASTE shall be less than or equal to 5,000 gallons. If ACTUAL MISSING WASTE is greater than 5,000 gallons, then the affected WASTE TRANSFERS shall be terminated immediately. (SAC)

- Material balance discrepancies shall be less than or equal to 15,000 gallons. If material balance discrepancies are greater than 15,000 gallons, then the affected WASTE TRANSFERS shall be terminated immediately. (SAC)

5.8.2.53 Waste Tank Chemical Cleaning Program

- Prior to TRANSFER INITIATION of a CHEMICAL CLEANING TRANSFER (or related vent/drain operations), isolation (single leak-tested valve, double valve isolation, or blank) shall be established to preclude CHEMICAL CLEANING TRANSFERS from entering pump tanks, waste tanks, or transfer facilities outside of the intended TRANSFER PATH. (SAC)
- Prior to transfer initiation of a CHEMICAL CLEANING TRANSFER (or related vent/drain operations), verification shall be performed to ensure that sufficient inhibitors are present in the applicable receipt waste tank to ensure neutralization of the acidic waste. (SAC)
- During receipt of CHEMICAL CLEANING TRANSFERS (or related vent/drain operations), at least one waste tank mixing device shall be operating in the applicable receipt waste tank to ensure adequate mixing and neutralization of the acidic waste. (SAC)

5.6.2 NCSE: Tanks 9 through 15 Chemical Heel Removal (U) (N-NCS-H-00274) (Ref. 91)

1. Chemical cleaning occurs in the Type I or II waste tanks. [Assumption]
2. The maximum volume of an inadvertent transfer is 15,000 gallons. [Assumption]
3. Any sludge transferred into the Receipt Tank not associated with BOAC activities (i.e., not from the treatment tank) meets the Tank Farm WAC poison requirements. [Assumption]
4. Transfer line jumpers and connectors – Jumpers are specially fabricated stainless steel pipe segments, with connectors for connecting to diversion box nozzles and are designed to complete a specific transfer route. [Design Feature]
5. Core Pipe and jacket/encasement – The core pipe is a Safety Class SSC transfer line within Safety Significant SSC transfer line jacket/encasement which "provide passive containment (confinement for encasements) of liquid waste … to channel core pipe leakage to a Leak Detection Location.". [Design Feature]
6. Prior to starting the Chemical Heel Removal process for a specific Treatment Tank an NCSA shall be completed to demonstrate the following: [Administrative Control]

   a) The U-235eq mass is less than or equal to 10 kg and the Pu-239 + Pu-241 mass is less than or equal to 6 kg. Where U-235eq = U-235 + 1.4 x U-233. U-235eq does not have to be counted if the U-235eq enrichment is ≤ 0.96 wt%. Where U-235eq enrichment = U-235eq ÷ U_{tot}

      (where U_{tot} = U-233 + U-234 + U-235 + U-236 + U-238)

   b) The poison requirements of the Tank Farm WAC (Ref. 33) are met.

   Poison to Fissile Mass Ratios

Poison-to-Fissile Mass ratios for 100 wt% U-235 Enrichment

Neutron Poison	Minimum Safe Mass Ratios
Fe	72:1
Mn	14:1

Poison-to-Fissile Mass ratios for 100 wt% Pu-239 Enrichment

Neutron Poison	Minimum Safe Mass Ratios
Fe	160:1
Mn	29:1

Equivalency Factors (EF)

Conversion	EF
Pu-239 to U-235	2.25
U-233 to U-235	2.25
U-235 to Pu-239	0.44

For mixtures of fissile material;

$$\text{U-235eqt} = \text{U-235} + 2.25 \times (\text{U-233} + \text{Pu-239} + \text{Pu-241})$$

$$\text{Pu-239eqt} = \text{Pu-239} + \text{Pu-241} + \text{U-233} + 0.44 \times \text{U-235}$$

There are sufficient individual poisons for subcriticality if the above poison-to-fissile mass ratios are met. If the individual poison ratio is not met both poisons may be credited. For crediting both Fe and Mn, use the following equation for A. In the equation, A must be zero or negative for the poison and fissile material mixture to be subcritical.

$$A = -(70 \div 12) \times (\text{Mn} \div \text{U-235eqt}) + 70 - (\text{Fe} \div \text{U-235eqt})$$

Where : Mn, Fe, and U-235eqt are mass or concentration values.

c) Where a spent oxalic acid transfer path other than from Treatment Tanks 11 or 12 to Receipt Tank 51 (or 13) is involved, the transfer path shall be evaluated to ensure: 1) that any Diversion Boxes, Valve Boxes, or Drain Valve Boxes not covered in this NCSE do not contain significant sludge heel (see photos in Appendix A of N-NCS-H-00274 [Ref. 91]) and 2) the existing scenarios bound the new transfers and any potential leaks on the transfer path.

7. Prior to transferring oxalic acid to the Treatment Tank, operators shall set the oxalic acid transfer path using operating procedures to ensure that the oxalic acid will be transferred to the intended Treatment Tank. Prior to transferring oxalic acid from the tanker to the Treatment Tank, a second person shall verify that the oxalic acid transfer path is connected to the intended Treatment Tank. [Administrative Control]

8. Before and after the completion of each batch transfer (one tanker truck) of oxalic acid, the level in the receiving Treatment Tank shall be checked. If there is no increase in the intended receiving Treatment Tank level, additional oxalic acid transfers shall be stopped until an investigation determines why the level did not increase. [Administrative Control]

9. The following TSR controls are credited to support the incredibility arguments for inadvertent waste transfers in N-NCS-H-00274. These controls are in the TSR for reasons other than criticality safety. The text is taken from the DSA (Ref. 1).

5.8.2.13 Corrosion Control Program

Programmatic controls shall be implemented to ensure that waste chemistry and characteristics are controlled so that corrosion will not impact the ability of safety related equipment to perform its function when required. This program shall include as a minimum, limits on pH, OH^-, NO_2^-, NO_3^-, SO_4^{-2}, and Cl^- as necessary to account for equipment corrosion.

5.8.2.21 Transfer Control Program

A Transfer Control Program shall be established governing all FACILITY initiated WASTE TRANSFERS and transfers associated with Tank 50. The program shall at a minimum include the following attributes:

- Prior to TRANSFER INITIATION, determine the following: (SAC)
  - TRANSFER PATH (including isolation points)
  - LEAK DETECTION LOCATIONS for the TRANSFER PATH
- Monitoring and material balance requirements for WASTE TRANSFERS to detect transfer events shall be determined.
- Prior to TRANSFER INITIATION, independent verification of correct TRANSFER PATH alignment shall be completed.
- Prior to TRANSFER INITIATION, procedures shall identify the functional equipment needed to stop transfers and siphons. (SAC)
- ACTUAL MISSING WASTE shall be less than or equal to 5,000 gallons. If ACTUAL MISSING WASTE is greater than 5,000 gallons, then the affected WASTE TRANSFERS shall be terminated immediately. (SAC)
- Material balance discrepancies shall be less than or equal to 15,000 gallons. If material balance discrepancies are greater than 15,000 gallons, then the affected WASTE TRANSFERS shall be terminated immediately. (SAC)

5.8.2.53 <u>Waste Tank Chemical Cleaning Program</u>

- Prior to TRANSFER INITIATION of a CHEMICAL CLEANING TRANSFER (or related vent/drain operations), isolation (single leak-tested valve, double valve isolation, or blank) shall be established to preclude CHEMICAL CLEANING TRANSFERS from entering pump tanks, waste tanks, or transfer facilities outside of the intended TRANSFER PATH. (SAC)

- Prior to transfer initiation of a CHEMICAL CLEANING TRANSFER (or related vent/drain operations), verification shall be performed to ensure that sufficient inhibitors are present in the applicable receipt waste tank to ensure neutralization of the acidic waste. (SAC).
- During receipt of CHEMICAL CLEANING TRANSFERS (or related vent/drain operations), at least one waste tank mixing device shall be operating in the applicable receipt waste tank to ensure adequate mixing and neutralization of the acidic waste. (SAC).

5.7 Aluminum Dissolution

This section summarizes the barriers from the NCSE pertaining to aluminum dissolution in CSTF.

5.7.1 NCSE: Aluminum Dissolution in Sludge (N-NCS-H-00211) (Ref. 27)

1. The maximum soluble uranium concentration in the salt solution for all ranges of NaOH concentration in this evaluation is no more than 55 mg/L. [BA]
2. The maximum plutonium soluble concentration in salt solution for all ranges of NaOH concentration greater than or equal to than 5 M is no more than 100 mg/L. [BA]
3. The maximum soluble plutonium concentration in salt solution with NaOH concentration less than 5 M is no more than 1.68 mg/L. [BA]
4. The maximum volume of an inadvertent transfer is 15,000 gallons (Ref. 1). [BA]
5. Tank Farm tanks contain a large amount of waste and for criticality purposes these tanks are considered as infinite waste medium. Some scenarios in this NCSE credit the size of Pump Tanks (with a diameter of 144 inches) as part of the incredibility arguments in the event of precipitation of fissile material. Pump Tanks are assumed to be used for transfer of caustic solution from the tanker truck to the dissolution tank. Therefore, use of other tanks with a diameter smaller than 144 inches or with working volume greater than 7,100 gallons for transfer of caustic solution requires re-evaluation. [Design Feature]
6. The following TSR controls are credited to support the incredibility arguments for inadvertent waste transfers in N-NCS-H-00211. The numbering and text are taken from the TSR (Ref. 2).

   5.8.2.21 Transfer Control Program [EOI (TCP)]

   A Transfer Control Program shall be established governing all FACILITY initiated WASTE TRANSFERS and transfers associated with Tank 50. The program shall at a minimum include the following attributes:

   c. Monitoring and material balance requirements for WASTE TRANSFERS to detect transfer events shall be determined.

   f. Prior to TRANSFER INITIATION, independent verification of correct TRANSFER PATH alignment shall be completed.

   h. Prior to TRANSFER INITIATION, procedures shall identify the functional equipment needed to stop transfers and siphons. (SAC)

   o. ACTUAL MISSING WASTE shall be less than or equal to 5,000 gallons. If ACTUAL MISSING WASTE is greater than 5,000 gallons, then the affected WASTE TRANSFERS shall be terminated immediately. (SAC)

p. Material balance discrepancies shall be less than or equal to 15,000 gallons. If material balance discrepancies are greater than 15,000 gallons, then the affected WASTE TRANSFERS shall be terminated immediately. (SAC)

5.8.2.25 Evaporator Feed Qualification Program

The Evaporator Feed Qualification Program shall ensure that the composition of waste streams received into the evaporator feed tanks is within analyzed limits. Waste streams received into the evaporator feed tanks shall comply with the inhalation dose potential and hydrogen generation rate limits prior to transfer to the evaporator pot. (SAC)

NOTE: The above TSR statement is from the TSR revision in effect when the NCSE was first written (Revision 16). The current revision of the TSR (Ref. 2) does not require compliance with an inhalation does potential limit. For criticality safety, the NCSE states that a feed qualification program for the evaporators shall ensure that the composition of the waste streams received into the evaporator feed tanks is within analyzed limits. This requirement remains in Reference 2.

5.8.2.49 Tank 48 Unauthorized Operations (SAC)

Controls shall be implemented to ensure the following operations are not permitted:

e. Waste additions/transfers to or from Tank 48.

6.0 SUMMARY AND CONCLUSIONS

This NCSASR summarizes the baseline CSTF NCSEs and NCSAs that analyzed or assessed postulated criticality scenarios. The NCSEs identified barriers that render criticality incredible in the CSTF. These NCSEs provide the technical safety bases for criticality safety of the CSTF. The NCSEs demonstrate that no credible criticality scenarios exist for the processing (e.g., salt dissolution, salt processing, salt solution evaporation, sludge preparation, chemical cleaning of evaporator, chemical cleaning of tanks) and storage of HLW in CSTF.

Reference 30 states that a Criticality Accident Alarm System (CAAS) is not required if a criticality is not credible. However, the criticality accident frequency criterion for CAAS is based on the total frequency of all scenarios for a facility or sub-facility and not applied to individual scenarios. The criticality safety of scenarios in CST relies on the nature of process of the waste (e.g., large area of the waste tanks and the presence of large amounts of neutron poisons). In addition, the CST is segmented into many facilities and sub-facilities, particularly due to distance between waste tanks/facilities. The interaction between facilities (e.g., transfers) is considered in the NCSEs and included in the determination that a criticality is not credible. Therefore, considering the total frequency of a criticality accident as applied to all scenarios in a facility or sub-facility, a criticality accident is still not credible in CST.

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APPENDIX 1. SUPERSEDED AND NOT APPLICABLE NCSES AND NCSAS

The following table lists the NCSEs and NCSAs that have been superseded or are no longer applicable to the CSTF but may contain useful information for future analyses or assessments. Note that this list may not be all-inclusive.

Document	Title
EPD-CTG-940070 (Ref. 43)	Nuclear Criticality Safety Evaluation Tank 41 Salt Removal
N-CLC-H-00506 (Ref. 44)	NCSA of Tank 11H Slurrying Operations
N-ESR-F-00035 (Ref. 45) Superseded by Ref. 80	NCSA: Tank 4 Saltcake Dissolution
N-ESR-F-00036 (Ref. 37) Superseded by Ref. 18	NCSA: Salt Dissolution Process of F-Area Tanks 25, and 28
N-ESR-G-00009 (Ref. 46)	NCSA: Transfer Line Fissile Mass Contribution To ARP/MCU Facility During Integrated Runs
N-ESR-H-00003 (Ref. 47)	NCSA: Incidental Salt Dissolution in Tank 29
N-ESR-H-00007 (Ref. 48)	NCSA: Transfer of Tank 49 to Tank 24
N-ESR-H-00010 (Ref. 49)	NCSA: Tank 37H Saltcake Dissolution (U)
N-NCS-E-00007 (Ref. 50)	NCSASR: A Summary of Criticality Safety of Tank 41 Salt Dissolution, In-Tank Precipitation (ITP), and Extended Sludge Processing (ESP) (U)
N-NCS-H-00006 (Ref. 51)	Nuclear Criticality Safety Evaluation For Tank 41H Salt Removal Without Slurry Pump Operation
N-NCS-H-00110 (Ref. 52)	NCSE: 242-16H Evaporator Sodium Aluminosilicate Scale Cleaning and Neutralization Byproduct Disposition (U)
N-NCS-H-00125 (Ref. 53)	CSSX Salt Processing Pilot Plant (U)
N-NCS-H-00147 (Ref. 55) Superseded by Ref. 15	NCSE: Tank 41 Dry Saltcake
N-NCS-H-00181 (Ref. 56)	Criticality Safety of Tank 38H After Inadvertent Salt Dissolution
WER-HLE-921198 (Ref. 57)	Nuclear Criticality Safety Analysis WM-92-4, Safe Disodium Uranate Mass at Various U-235 Enrichments
WER-HLE-930861 (Ref. 58)	Nuclear Criticality Safety Analysis WM-32-2 Minimum Safe Ratios of Cr and Zn to Pu-239 and U-235 in an Infinite System
WSRC-TR-95-0348 (Ref. 59)	Nuclear Criticality Safety Evaluation of the In-Tank-Precipitation (ITP) Chemical Addition Process
WSRC-TR-96-0336 (Ref. 39)	Rev. 4: NCSA for HPT-7 Drainback into Tank 38H
WSRC-TR-97-00221 (Ref. 60)	NCSA: Tank 12H Sludge Removal

Document	Title
WSRC-TR-98-00160 (Ref. 61)	Nuclear Criticality Safety Assessment of Tank 8F Sludge Removal
WSRC-TR-2000-00043 (Ref. 62)	Criticality Safety Assessment of the 242-16F Evaporator (Transfer of Tank 5F Supernate to Tank 46F)
WSRC-TR-2000-00069 (Ref. 63)	Criticality Assessment of the 242-25H Evaporator System
WSRC-TR-2000-00096 (Ref. 64)	Nuclear Criticality Safety Assessment For Tank 19 Waste Removal
WSRC-TR-2001-00556 (Ref. 65)	Nuclear Criticality Safety Assessment for Tank 37H Surface Layer
WSRC-TR-2001-00016 (Ref. 66)	Criticality Safety Assessment: Impact of the November/December 2000 Tank 40H Decant on the Criticality Safety Assessment of the 242-25H Evaporator System (WSRC-TR-2000-00069)
WSRC-TR-2001-00196 (Ref. 67)	Criticality Safety Assessment: Impact of Tank 40H Sludge Batch 2 Decant 2 on the Criticality Safety Assessment of the 242-25H Evaporator System (WSRC-TR-2000-00069)
WSRC-TR-2001-00380 (Ref. 68)	NCSA: Tank 34F Salt Dissolution
WSRC-TR-2001-00321 (Ref. 69)	NCSA: Tank 7F Sludge
WSRC-TR-2001-00273 (Ref. 70)	Criticality Safety Assessment for 2F Evaporator
WSRC-TR-2004-00061 (Ref. 71)	NCSA: Incidental Salt Dissolution in Tank 12H
WSRC-TR-2007-00201 (Ref. 38) Superseded by Ref. 18	NCSA: Tank 25 Transfer to Tank 41
N-NCS-H-00121 (Ref. 72) Superseded by Ref. 22	NCSE: 242-16H Evaporator Restart
N-NCS-H-00188 (Ref. 73) Superseded by Ref. 22	NCSE: Chemical Cleaning of the 242-16H Evaporator
N-NCS-H-00235 (Ref. 74)	NCSA: Scale Cleaning of the 2H Evaporator System
N-ESR-H-00009 (Ref. 75)	NCSA: Impact of Plutonium in Scale on the 2H Evaporator NCSE
N-ESR-H-00013 (Ref. 76)	NCSA: Enrichment Sampling During 2H Evaporator Outage
WSRC-TR-2004-00068 (Ref. 77)	NCSA: 299-H Facility HEU Waste Transfers to 242-16H Evaporator
N-NCS-H-00233 (Ref. 78)	NCSA: HEU Supernate Transfers to the 2H Evaporator System
N-NCS-H-00225 (Ref. 25)	NCSA: Tank 49H Macro Batch 1 to ARP/MCU
N-NCS-H-00246 (Ref. 82)	NCSA: MST Strikes in Strike Tank 1 in 241-96H
N-NCS-H-00245 (Ref. 81)	NCSA: Tank 49H Macro Batch 2 to ARP/MCU
N-NCS-H-00254 (Ref. 85)	NCSA: Tank 49H Macro Batch 3 to ARP/MCU

Document	Title
N-NCS-H-00268 (Ref. 88)	NCSA: Double Strike of MST to Strike Tank 2 at 241-96H
N-NCS-H-00273 (Ref. 90)	NCSA: Draining ETP to Tank 50 Transfer Line to HPT-10
N-NCS-H-00264 (Ref. 87)	NCSA: Chemical Cleaning of 242-16H Evaporator (U)