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Draft Section 3116 Determination for Closure of Tank 19 and Tank 18 at the Savannah River Site

September 30, 2005

APPROVED for Unlimited Release
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TABLE OF CONTENTS

LIST OF FIGURES **4**

LIST OF TABLES **7**

LIST OF ACRONYMS **8**

1.0 INTRODUCTION AND PURPOSE **11**

1.1 Introduction **11**

1.2 Purpose **15**

2.0 BACKGROUND **16**

2.1 SRS Tank Farms **17**
2.1.1 SRS Tank Farm Overview 17
2.1.2 SRS Federal Facility Agreement 19

2.2 Tank 19 and Tank 18 Location and Design **20**

2.3 Tank 19 and Tank 18 Waste Sources/Attributes **27**
2.3.1 Waste Sources In Tank 19 27
2.3.2 Tank 19 Waste Attributes (Primarily a Saltcake Bearing Tank) 30
2.3.3 Waste Sources In Tank 18 31
2.3.4 Tank 18 Waste Attributes (Primarily a Sludge Bearing Tank) 33

2.4 Tank 19 and Tank 18 Waste Removal **34**
2.4.1 Tank 19 Waste Removal 34
2.4.2 Tank 18 Waste Removal 48

2.5 Radionuclide Inventory **61**
2.5.1 Tank 19 Residual Waste Characterization Process 61
2.5.2 Tank 18 Residual Waste Characterization Process 69

2.6 Tank Stabilization **76**
2.6.1 Tank 19 and Tank 18 Stabilization 77
2.6.2 Tank 19 and Tank 18 Post Stabilization 79

3.0 SECTION 3116 OF THE RONALD W. REAGAN NATIONAL DEFENSE AUTHORIZATION ACT FOR FISCAL YEAR 2005 ..... **81**

4.0 WASTE DOES NOT REQUIRE PERMANENT ISOLATION IN A DEEP GEOLOGIC REPOSITORY FOR SPENT FUEL OR HIGH-LEVEL RADIOACTIVE WASTE **82**

5.0 WASTE HAS HAD HIGHLY RADIOACTIVE RADIONUCLIDES REMOVED TO THE MAXIMUM EXTENT PRACTICAL **84**

5.1 Highly Radioactive Radionuclides **85**
5.1.1 Approach 85
5.1.2 Radionuclides from Performance Objective Demonstration 86

5.2 Radionuclide Removal to the Maximum Extent Practical **88**
5.2.1 Percentage of Radionuclides Removed 92

5.3 Risk Benefit Analyses for Further Residual Waste Removal 96
5.3.1 Residual Heel Removal Methods 97
5.3.2 Comparative Analysis 99
5.3.3 Conclusions 100

6.0 RADIONUCLIDE CONCENTRATIONS OF STABILIZED WASTE 101

6.1 Waste Concentrations in Grouted Tanks 19 and 18 103

7.0 THE WASTE WILL BE DISPOSED OF IN ACCORDANCE WITH THE PERFORMANCE OBJECTIVES SET OUT IN 10 CFR 61, SUBPART C 106

7.1 10 CFR Part 61 Performance Objectives 106
7.1.1 Section 61.40, General Requirement 107
7.1.2 Section 61.41, Protection of the General Population from Releases of Radioactivity 107
7.1.3 Section 61.42, Protection of Individuals from Inadvertent Intrusion 107
7.1.4 Section 61.43, Protection of Individuals During Operations 107
7.1.5 Section 61.44, Stability of the Disposal Site After Closure 107

7.2 10 CFR Part 61 General Requirements 107
7.2.1 Section 61.40, General Requirement 107
7.2.2 Section 61.41, Protection Of The General Population from Releases of Radioactivity 108
7.2.3 Section 61.42, Protection of Individuals From Inadvertent Intrusion 133
7.2.4 Section 61.43, Protection Of Individuals During Operations 139
7.2.5 Section 61.44, Stability Of The Disposal Site After Closure 151

8.0 STATE-APPROVED CLOSURE PLAN 154

9.0 CONCLUSION 156

10.0 REFERENCES 157

APPENDICES

Page

Appendix A: System Description…….. A-1

Appendix B: Environmental Regulatory Integration for the HLW Tank System…….. B-1

Appendix C: Tank 19 and Tank 18 Waste Removal Program Technology Review….. C-1

Appendix D: Waste Characterization System…… D-1

Appendix E: F-Tank Farm Residual Material Inventory…… E-1

Appendix F: Management Control Systems…….. F-1

LIST OF FIGURES

Figure 2-1: General Layout of F-Tank Farm 21
Figure 2-2: Tank 19 and Tank 18 in their Operational Grouping 21
Figure 2-3: Aerial Photograph of Tank 19 and Tank 18 in their Operational Grouping in FTF 22
Figure 2-4: Tank 17 through Tank 20 Concrete Basemat 23
Figure 2-5: Steel Tank Walls 23
Figure 2-6: Tank Concrete Walls 24
Figure 2-7: Type IV Tank 24
Figure 2-8: Examples of Steel Truss Work 25
Figure 2-9: Tank 19 Riser Opening Locations and Associated Equipment 26
Figure 2-10: Tank 18 Center and Perimeter Risers 26
Figure 2-11: Tank 18 Riser Opening Locations and Associated Equipment 27
Figure 2-12: Tank 19 PUREX Waste Stream and 242-F Evaporator Diagram 29
Figure 2-13: Cesium Removal Column on Tank Top 30
Figure 2-14: Tank 18 Operational Diagram 32
Figure 2-15: Tank 18 Waste Removal Diagram 33
Figure 2-16: Tank 19 Historical Timeline 34
Figure 2-17: Tank 19 Initial Condition Prior to Waste Removal 35
Figure 2-18: Tank 19 Salt Removal Campaign 36
Figure 2-19: Typical Standard Mixer Pump 37
Figure 2-20: Hourglass Shaped Landscape following Bulk Salt Removal 39
Figure 2-21: Diagram of Tank 19 Access Area for Heel Removal Equipment 40
Figure 2-22: Prototype Mixer Pumps 44
Figure 2-23: Tank 19 Heel Removal Rate 46
Figure 2-24: Tank 19 Remaining Heel Volume 46
Figure 2-25: Hydro-Lance in Operation Breaking Up Mound 47
Figure 2-26: Tank 19 Waste Removal Effectiveness – Volume 48
Figure 2-27: Tank 18 Historical Timeline 49
Figure 2-28: Tank 18 Initial Condition 50
Figure 2-29: Tank 18 Sludge Removal Campaign 51

Figure 2-30: Heel Map Following Bulk Sludge Removal ... 53
Figure 2-31: Diagram of Tank 18 Access Area for Heel Removal Equipment ... 55
Figure 2-32: Advanced Design Mixer Pump (ADMP) ... 56
Figure 2-33: Modeling and Test Results for Tank 18 ... 57
Figure 2-34: Tank 18 Heel Removal Rate ... 58
Figure 2-35: Tank 18 Remaining Heel Volume ... 59
Figure 2-36: Tank 18 Heel Removal-Mixing Rate ... 59
Figure 2-37: Tank 18 Waste Removal Effectiveness – Sludge Volume ... 60
Figure 2-38: Tank 19 Sample Tools ... 62
Figure 2-39: Samples in the Laboratory ... 62
Figure 2-40: Similarity of Tank 19 Samples ... 63
Figure 2-41: Tank 19 Exposed Solids Maps ... 64
Figure 2-42: Tank 19 Residual Material Configuration and Sample Locations ... 66
Figure 2-43: Tank 18 Sample Tools ... 70
Figure 2-44: Samples in the Laboratory ... 70
Figure 2-45: Variation Among Tank 18 Samples ... 71
Figure 2-46: Tank 18 Exposed Solids Maps ... 72
Figure 2-47: Tank 18 Residual Material Configuration and Sample Locations ... 73
Figure 2-48: Typical Tank 19 and Tank 18 Grout Pour ... 77
Figure 5-1: Tank 19 Heel Removal Rate ... 90
Figure 5-2: Tank 18 Heel Removal Rate ... 91
Figure 6-1: Tank 19 Radionuclide Concentrations ... 105
Figure 6-2: Tank 18 Radionuclide Concentrations ... 105
Figure 7-1: Location of Tank 19 and Tank 18 ... 110
Figure 7-2: Location of Hypothetical Adult Resident in Relation to Tank 19 and Tank 18 ... 112
Figure 7-3: Typical View of Fourmile Branch ... 113
Figure 7-4: Seepline Area at Fourmile Branch ... 113
Figure 7-5: All Exposure Pathways for Member of Public ... 116
Figure 7-6: Conceptual Model Showing Assumptions and Key Parameters Prior to Hydraulic Grout Failure ... 118
Figure 7-7: Tank 19 All-Pathways Beta Peak Dose ... 122

Figure 7-8: Tank 19 All-Pathways Alpha Peak Dose 122

Figure 7-9: Tank 18 All-Pathways Beta Peak Dose 123

Figure 7-10: Tank 18 All-Pathways Alpha Peak Dose 123

Figure 7-11: Uncertainty Analysis Process for Member of the Public Model..... 127

Figure 7-12: Plume Centerline Overlap..... 131

Figure 7-13: Conceptual Representation of Inadvertent Intruder-Agricultural Scenario (10 CFR 61.42)..... 135

Figure 7-14: Uncertainty Analysis Process for Inadvertent Model..... 137

LIST OF TABLES

Table 2-1: Tank 19 Saltcake Dissolution Campaigns 38
Table 2-2: Tank 18 Sludge Removal Campaigns 52
Table 2-3: Tank 19 Radionuclide Inventory 67
Table 2-4: Tank 18 Radionuclide Inventory 75
Table 2-5: Tank 19 and Tank 18 Grout Properties 78
Table 2-6: Tank 19 and Tank 18 Grout Components 79
Table 5-1: List of Highly Radioactive Radionuclides 87
Table 5-2: Percentage of Highly Radioactive Radionuclides Removed from Tank 19 94
Table 5-3: Percentage of Highly Radioactive Radionuclides Removed from Tank 18 95
Table 5-4: Considerations for Further Residual Waste Removal 100
Table 6-1: 10 CFR 61.55 Table 1 103
Table 6-2: 10 CFR 61.55 Table 2 103
Table 7-1: Maximum Annual All-Pathway Doses to a Member of the Public with Institutional Controls in Place, 100 – 10,000 Years 121
Table 7-2: Peak Annual Whole-Body Dose Contributors 124
Table 7-3: Peak Annual All-Pathways FTF Dose 128
Table 7-4: Comparison of Modeling Assumptions for F-Tank Farm to Actual Conditions 130

LIST OF ACRONYMS

ADMP	Advanced Design Mixer Pump
ALARA	As Low As Reasonably Achievable
ARP	Actinide Removal Process
ASTM	American Society for Testing and Materials
BTP	Branch Technical Position
BW	Bearing Water
CEDE	Committed Effective Dose Equivalent
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
Ci	curie
CRC	Cesium Removal Column
Cs-137	Cesium-137
CSEE	Confined Sluicing End Effectors
CSTF	Concentration, Storage, and Transfer Facilities
DDA	Deliquification, Dissolution, and Adjustment
DOE	Department of Energy
DSA	Documented Safety Analysis
DWPF	Defense Waste Processing Facility
ECR	Effective Cleaning Radius
EIS	Environmental Impact Statement
EPA	Environmental Protection Agency
FFA	Federal Facility Agreement
FTF	F-Tank Farm
FY	Fiscal Year
GAAT	Gunite and Associated Tanks
GCP	General Closure Plan
gpm	Gallons Per Minute
GSA	General Separations Area
GTS	Groundwater Transport Segment
GWSB	Glass Waste Storage Building
HA	Hazard Analysis
HAW	High activity waste
HEU	Highly Enriched Uranium
HLW	High-Level Waste
hp	Horsepower
hr	Hour
HTF	H-Tank Farm
IW	Inhibited Water

K_d	Distribution Coefficients
kgal	Thousand Gallons
LAW	Low activity waste
LLW	Low-Level Waste
MEPAS	Multimedia Environmental Pollutant Assessment System
mg	milligrams
Mgal	Million Gallons
mrem	millirem
MSL	Mean Sea Level
MWMF	Mixed Waste Management Facility
NASA	National Aeronautics and Space Administration
NDAA	National Defense Authorization Act
NIST	National Institute of Standards and Technology
NPH	Natural Phenomena Hazard
NRC	Nuclear Regulatory Commission
OA	Oxalic Acid
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
OU	Operable Units
PEL	Permissible Exposure Limit
PODD	Performance Objective Demonstration Document
PNNL	Pacific Northwest National Laboratory
psi	pounds per square inch
Pu	Plutonium
PUREX	Plutonium Recovery and Extraction
QA	Quality Assurance
RBA	Radiological Buffer Area
RESRAD	RESidual RADioactivity Computer Software
ROD	Record of Decision
rpm	Revolutions Per Minute
RPP	Radiation Protection Program
SARA	Superfund Amendments and Reauthorization Act
SCDHEC	South Carolina Department of Health and Environmental Control
SEE	System Engineering Evaluation
SMA	Strong Motion Accelerometers
SOF	Sum-of-Fractions
Sr-90	Strontium-90
SRID	Standards/Requirements Identification Document
SRNL	Savannah River National Laboratory
SRS	Savannah River Site
SWPF	Salt Waste Processing Facility
Tc-99	Technetium-99

TEDE	Total Effective Dose Equivalent
TFA	Tanks Focus Area
TRU	Transuranic
TSR	Technical Safety Requirement
TTP	Telescoping Transfer Pump
WCS	Waste Characterization System
WR	Waste Removal
WSRC	Westinghouse Savannah River Company

1.0 INTRODUCTION AND PURPOSE

Section Purpose

The purpose of this section is to describe the purpose and scope of this document and to provide brief background information to lay the foundation for detailed discussions in later sections.

Section Contents

This section briefly summarizes the general background related to the underground waste tanks at the Savannah River Site (SRS), with emphasis on Tanks 19 and 18 and their history, and then describes the purpose of the document.

Key Points

- The Department of Energy (DOE) is issuing this draft waste determination pursuant to Section 3116 of the Ronald W. Reagan National Defense Authorization Act (NDAA) for Fiscal Year 2005 (Public Law 108-375, 2004).
- It applies to permanent disposal of residual materials inside two underground waste tanks at the SRS – Tanks 19 and 18 – and not to any other site facilities or waste.
- The Department will issue this draft waste determination in final form after completion of required consultation with the Nuclear Regulatory Commission (NRC).
- Tanks 19 and 18 are 85-foot diameter, 34-foot high carbon steel tanks with nominal operating volumes of 1,300,000 gallons each; waste has been removed from each tank to the extent practicable.
- Tank 19 contains approximately 15,100 gallons (15.1 kgal) of residual materials with an average equivalent depth of 4.2 inches and approximately 96,000 curies of radioactivity.
- Tank 18 contains approximately 4.3 kgal of residual materials with an average equivalent depth of 1.2 inches and approximately 28,000 curies.
- These tanks are being closed in place on a schedule provided in the site Federal Facility Agreement (FFA), to which the State of South Carolina is a signatory, in accordance with a State-approved closure plan.
- The DOE has committed to the State of South Carolina to close Tank 19 by October 31, 2006 and Tank 18 by February 28, 2007.
- The tanks will be filled with reducing grout to stabilize the residual materials in a solidified waste form; strong grout will be poured into the top part of the tanks to serve as an engineered barrier to inadvertent intrusion.

1.1 Introduction

This draft 3116 Determination is being issued pursuant to Section 3116 of the Ronald W. Reagan NDAA for Fiscal Year 2005 (Public Law 108-375, 2004). This document concerns the permanent in-situ disposal of the solidified residual tank wastes (designated Tank 19 and Tank 18) at the F-Tank Farm (FTF) at SRS near Aiken, South Carolina.

NOTE

Throughout this document, Tank 19 is referred to before Tank 18 because DOE is scheduled to close Tank 19 first, in accordance with the FFA, as explained below. [1]

Construction of SRS and the Underground Waste Tanks

In 1951, construction of the SRS began under the direction of the U. S. Atomic Energy Commission on approximately 300 square miles of property purchased by the Federal government. The mission of this new Federal facility was to be production of nuclear material for national defense, research, medical, and space programs. [2] The production of this nuclear material involved numerous complex processes, including the separation of fissionable nuclear material from components called targets that were irradiated in the site's nuclear reactors and from irradiated fuels used to operate these reactors. The Atomic Energy Commission built two production complexes to support the separations process. [2] Known as F Area and H Area, these complexes were located in the central part of the Federally-owned property. This location, which is naturally bounded by three small black water streams called Upper Three Runs, Fourmile Branch and McQueen Branch – became known as the General Separations Area (GSA). [2]

The central facility in each complex was the canyon building. These 800-foot-long structures housed equipment used to dissolve irradiated targets and fuel and separate nuclear materials. The separations processes carried out in the canyons generated large quantities of liquid radioactive waste. To store this waste until it could eventually be disposed of, a tank farm was built for each complex. FTF eventually included 22 large underground liquid waste storage tanks used for this purpose, including Tank 19 and Tank 18. [2] Construction of Tank 19 and Tank 18 began in 1956. They were built along with two others designated Tanks 17 and 20. These four tanks, which became known as the "four pack", are located in the northwest corner of the tank farm, in a depressed area whose surface lies approximately 20-feet lower than the surrounding grade. [2]

Tank 19 and Tank 18 are single-shell carbon steel tanks, with no secondary containment. Each tank is 85-feet in diameter and 34-feet high, with a domed roof and a nominal operating volume of 1,300,000 gallons (1,300 kgal). Tank 19 entered service in 1961, while Tank 18 was placed into service in 1960.

Storage and Processing of Waste

Before these two tanks entered service, other underground storage tanks in FTF and in the similar tank farm in H Area began receiving waste from the canyons. This waste takes two distinct forms known as sludge and salt waste. Sludge, the more highly radioactive component, consists of metals such as strontium and plutonium that precipitated out of solution to form a layer on the bottom of the tank. Salt waste is comprised of liquid salt solution and crystallized saltcake, which is a solid composed mainly of nitrate and nitrite salts, with only small concentrations of radioactive material. [3] The total waste volume received in the two tank farms since 1954 exceeds 140 million gallons (Mgal). Much of this waste resulted from the reprocessing of spent nuclear fuel for defense purposes, which has been commingled with non-

reprocessing waste resulting from the production of nuclear weapons materials and the production of radiological materials for National Aeronautics and Space Administration (NASA) space missions. Today, the underground waste tanks contain approximately 3 Mgal of radioactive sludge and approximately 34 Mgal of salt waste, with these volumes fluctuating slightly due to factors such as receipt of waste from the canyons, the utility of the evaporators, and waste treatment activities. [3] To complete this brief review of the background on the issues related to disposal of residual waste in Tank 19 and Tank 18, it is useful to consider the present contents of these two tanks, and the reasons DOE needs to close them as soon as practical.

Tank 19

From 1961 to 1980, Tank 19 received unconcentrated Plutonium Recovery and Extraction (PUREX) waste and concentrated waste from F-Canyon, along with concentrated waste from an FTF evaporator system. Tank 19 also received zeolite, an ion exchange column resin used to remove cesium-137 (Cs-137) from evaporator condensate streams. Because of its process history, Tank 19 was primarily a salt-bearing tank.

Waste removal from Tank 19 was initiated in 1981. The volume stored in Tank 19 at the beginning of waste removal was approximately 1,300 kgal. Waste removal operations were completed in 2001 and Tank 19 now holds approximately 15.1 kgal of residual material, with an equivalent average depth or heel of approximately 4.2 inches in the 34-foot-high tank. [4] The residual waste (excluding tank structures and equipment) that will be permanently disposed of in Tank 19 at the time of closure is comprised of, by weight:[5]

- ~66% zeolite
- ~26% aluminum hydroxides, which originated from the reprocessing of aluminum clad, spent nuclear fuels and targets
- ~8% iron oxides, which originated from the PUREX process
- <<1% radioactive material (~96,000 curies)

Tank 18

Tank 18 was placed into service in 1960 and operated as the 242-F evaporator feed tank until 1976. After 1976 it was primarily used as the receipt tank to facilitate salt and sludge removal campaigns from Tank 17, Tank 19, and Tank 20.

During its active service life, Tank 18 received both concentrated and unconcentrated PUREX waste. Because of its use as a receipt tank after 1976, it was primarily a sludge-bearing tank.

Tank 18 was designed to be the lone transfer tank out of the four pack area and thus was the last of these tanks to undergo waste removal activities. Consequently, almost 100% of the sludge sent to the four pack tanks passed through Tank 18. In addition, Tank 18, as a result of the waste removal efforts for Tank 19, received some zeolite resin.

The volume stored in Tank 18 at the beginning of waste removal was approximately 1,300 kgal. Waste removal operations were initiated in 1985 and completed in July 2003. Tank 18 now holds approximately 4.3 kgal of residual material with an equivalent average depth of approximately 1.2 inches in the 34-foot-high tank. [4]

The residual waste (excluding tank structure and equipment) that will be permanently disposed of in Tank 18 at the time of closure is comprised of (by weight):[6]

- ~46% zeolite
- ~43% iron oxides, which originated from the PUREX process
- ~11% aluminum hydroxides, which originated from the reprocessing of aluminum clad, spent nuclear fuels and targets
- <<1% radioactive material (~27,600 curies)

Schedule and Plans for Closing Tanks

Tanks at the site are being closed in accordance with a formal agreement among DOE, Region IV of the U.S. Environmental Protection Agency (EPA), and the South Carolina Department of Health and Environmental Control (SCDHEC) expressed in the SRS FFA. [1] SRS tanks that do not meet secondary containment standards, as established in the FFA, must be removed from service per the FFA schedule. [1],[7] Waste tanks that do not meet the standards established in Appendix B of the SRS FFA have closure priority. [1] Tank 19 and Tank 18 are the next two tanks listed in the FFA schedule.

Tank 19 is scheduled for closure per the FFA by October 31, 2006 and Tank 18 is scheduled for closure by February 28, 2007. These closure dates constitute enforceable deadlines under the FFA.

DOE and the regulators selected Tanks 17, 18, 19 and 20 for early closure because bulk waste removal efforts during the 1980s produced a state of readiness for closure and because SRS had no future operational plans for these four tanks. In addition, due to the close proximity of these tanks, SRS could close out an entire geographic section of FTF.

Tank waste storage and removal are governed by an industrial wastewater operating permit issued by the SCDHEC. After the conclusion of this waste determination process, the overall plan for closing FTF and H-Tank Farm (HTF), typically referred to as the General Closure Plan (GCP), will be updated and submitted to the State of South Carolina for approval. A specific closure plan for each tank, commonly referred to as a tank closure module, will be developed and submitted to the State of South Carolina for approval and this approval will be received before activities to close each tank proceed.

Tank Closure System

In the High-Level Waste Tank Closure Final Environmental Impact Statement (EIS) (May 2002) Record of Decision (ROD), DOE selected the alternative to fill the tanks with reducing grout to stabilize the residual material to create a solidified waste form. This method was chosen as the least hazardous and the environmentally preferable alternative for permanent disposal of the residual waste in Tank 19 and Tank 18. [8]

The solidified waste form will provide a chemical environment to reduce migration of contaminants into the environment; prevent inadvertent intrusion; minimize free-standing liquids; and minimize void spaces in the tank. After final pouring of the stabilizing grout, a layer of higher-strength grout will be poured into the tank up to the riser openings to further discourage human/animal inadvertent intrusion.

Demonstrating Compliance with Section 3116 of the NDAA

In order to proceed with closing Tank 19 and Tank 18, DOE must first demonstrate that the closed tanks will comply with Section 3116 of the NDAA (Public Law 108-375, 2004).

Section 3116 of the NDAA specifies that the term "high-level radioactive waste" does not include radioactive waste that results from reprocessing spent nuclear fuel if the Secretary of Energy determines, in consultation with the NRC, that the waste meets certain criteria, which specify (1) that the waste must not require disposal in a deep geologic repository (3116(a)(1)), (2) must have had highly radioactive radionuclides removed to the maximum extent practical (3116(a)(2)), (3) must meet certain concentration limits and performance objectives for low-level-waste (LLW) (3116(a)(3)), and (4) must be disposed of pursuant to a State-approved closure plan or State-issued permit (3116(a)(3)). In this document, the Secretary of Energy demonstrates that the resulting solidified residual tank wastes (including tank structure and equipment) in Tank 19 and Tank 18, after final closure activities are completed will meet all of these criteria. Accordingly, these solidified residual wastes may be determined not to be high-level waste (HLW) and may be grouted and disposed of in place as LLW.

To demonstrate that the Tank 19 and Tank 18 residual waste at final closure will meet the Section 3116 criteria, historical waste management information and sampling and analysis results from the Tank 19 and Tank 18 cleaning activities were reviewed, performance assessment modeling was performed, and the results from all of these activities were analyzed. Details are provided in the additional sections of this document.

1.2 Purpose

The purpose of this draft 3116 Determination is therefore to demonstrate and document that the solidified residual tank wastes (including tank structure and equipment) in Tank 19 and Tank 18 meet the 3116 criteria and therefore may be determined not to be HLW and may be disposed of as LLW in-situ, in accordance with Section 3116 of the NDAA (Public Law 108-375, 2004).

This draft 3116 Determination specifically addresses the closure of Tank 19 and Tank 18. The scope of this draft 3116 Determination does not include any other facilities or systems at FTF, nor does it include waste removed from the tanks, because, among other reasons, Section 3116(c), in conjunction Section 3116(d), provides that Section 3116 does not apply to waste transported from the State of South Carolina. Future draft 3116 Determination documents will be developed to address remaining SRS tanks either individually or in groups.

This draft 3116 Determination will be finalized after DOE has completed consultation with the NRC and the Secretary has made a Determination pursuant to Section 3116. Though not required by Section 3116, DOE is also issuing this draft 3116 Determination for public review and comment.

2.0 BACKGROUND

Section Purpose

The purpose of this section is to provide detailed background information to support discussions in later sections about compliance with Section 3116 of the NDAA.

Section Contents

Section 2.1 describes the operational history of FTF and the regulatory history of FTF related to the FFA. Section 2.2 describes the location and design of each tank. Section 2.3 describes waste sources and attributes for each tank. Section 2.4 describes waste removal efforts. Section 2.5 describes the waste characterization process and residual radionuclide inventories. Section 2.6 discusses stabilization of the tanks.

Key Points

- The tanks of interest are located in the 22-acre FTF, which lies in the GSA.
- Both tanks were full (1,300,000 gallons) at the beginning of waste removal efforts.
- A variety of waste removal techniques were evaluated to select the most efficient techniques, which were used over many years until constraints related to tank design and remaining materials prevented removal of the final heels.
- Tank residual materials consist of a combination of (1) an alkaline liquid salt solution, (2) sludge comprised of fine insoluble solid particles, and (3) hardened zeolite that was used as an ion exchange medium.
- Residual radionuclide inventories were conservatively estimated based on sample analysis results, residuals mapping, and volume estimation.
- For radionuclides that were not analyzed in the laboratory, the Waste Characterization System (WCS) was used to determine radionuclide inventory based on process knowledge. Additionally, special analyses were performed which included fission yield ratios to other radionuclides, and equations for progeny radionuclides in terms of the appropriate parent radionuclides.
- Tank 19 and Tank 18 residual radionuclides are dominated by Cs-137 and its progeny, Ba-137m, which make up more than 99% of the radioactivity inside Tank 19 as of the 2002 inventory reference date, and 86% inside Tank 18, as of the 2003 inventory reference date for Tank 18.
- Nearly all this Cs-137 is chemically bound to zeolite and therefore will not leach out of the stabilized waste form.
- Reducing grout composed primarily of sand, water, fly ash, slag and cement, will provide long-term stability and minimize leaching of radionuclides; the strong grout poured into the top of the tank to prevent future inadvertent intrusion will have a compressive strength greater than 2,000 pounds per square inch.
- The earthen cover above the tanks will be at least 10-feet thick and will be installed when the FTF area is closed in accordance with the FFA.

2.1 SRS Tank Farms

2.1.1 SRS Tank Farm Overview

Construction of SRS began in 1951, and the site began producing nuclear materials in 1953 for national defense, research, medical, and space programs. These production missions have resulted in the generation of large quantities of radioactive waste, which is currently stored onsite in 49[1] underground carbon steel waste storage tanks in two tank farms. [2]

Two tank farms were designed and constructed to support the operation of various SRS production, processing, and laboratory facilities. The support function of these tank farms is to isolate from the environment, SRS workers, and the public, the radioactive wastes generated by these facilities. The tank farms are located in proximity to the processing area that they serve. The F-Area Tank Farm (FTF) is located adjacent to the F Area processing area and consists of 20 active waste tanks, two closed waste tanks, evaporator systems, and transfer systems on a 22-acre site. The H-Area Tank Farm (HTF) is located adjacent to the H Area processing area and consists of 29 active waste tanks, evaporator systems, and transfer systems on a 45-acre site. [2]

Since initiation of operations at SRS, the Tank Farms together have received over 140 Mgal of liquid waste from the chemical separation processes in F- and H-Canyons associated with the reprocessing of spent nuclear fuel, the production of nuclear materials for weapons, and the production of material for NASA space missions. [9] In addition to the canyon waste streams, the Defense Waste Processing Facility (DWPF) has transferred a liquid waste stream (typically referred to as "recycle")[2] to the Tank Farms. This liquid waste stream is a by-product of the production of waste canisters. These waste canisters contain SRS tank farm waste mixed with borosilicate glass and are planned to be disposed of in a deep geologic repository at Yucca Mountain, Nevada, for spent nuclear fuel and high-level radioactive waste after the repository is licensed and operational. [3] The F- and H-Canyon chemical separation processes both used acids to dissolve irradiated reactor or target fuel slugs to prepare the desired products for extraction. The DWPF process also uses acid to prepare the sludge feed for downstream processing prior to the sludge feed entering the glass melter. The resultant waste stream from all of these processes is acidic. Prior to transfer of the waste material from the F- and H-Canyons and DWPF to the tank farms, sodium hydroxide is added to adjust the waste to an alkaline state to prevent corrosion of the carbon steel waste tanks. This chemical adjustment results in the precipitation of solids. These solids settle in the waste tanks forming a layer that is commonly referred to as "sludge". These solids are comprised of fine particles of settled metal oxides, including strontium (Sr) and plutonium (Pu) hydroxides. These solids are insoluble due to the chemical conditions of the solution. After settling of the solids has occurred, the liquid salt waste solution (supernate) above this sludge layer is transferred out of the tank. In order to

[1] Following approval of closure modules by the state of South Carolina and oversight by EPA, DOE operationally closed Tank 17 and Tank 20 in December 1997 and June 1997, respectively. On June 30, 2000, the NRC issued to DOE its final report confirming the SRS approach.

[2] The DWPF recycle waste stream is a generally very low-activity stream that consists of condensate from chemical processing and melter operation, waste from decontamination activities, and waste from miscellaneous drains and sumps in DWPF. Before the recycle stream is transferred to the Tank Farm, it is chemically adjusted with sodium nitrite and sodium hydroxide.

maximize the space available in the tanks for storing additional waste, DOE practice at SRS has been to use the Tank Farm evaporator systems to reduce the volume of the decanted supernate by concentrating the waste. Providing additional space in the tanks allows additional receipt of waste from the processing facilities into the tanks. [10]

During the evaporation process, the liquid salt waste is concentrated and forms two distinct phases – concentrated supernate solution and solid saltcake (collectively called salt waste). The predominant radionuclide present in the salt waste is Cs-137. Because of the high solubility of Cs-137, approximately 95% of the Cs-137 is present in the concentrated supernate solution and the liquid found within the interstitial spaces in saltcake. The solid saltcake is composed predominantly of nitrate and nitrite salts and contains relatively small quantities of radioactive material such as C-14, Sr-90, Y-90 and Tc-99. [11],[3] See Appendix A for additional information on tank farm operations.

As the result of the evaporation process, over 140 Mgal of liquid waste originally received in F- and H- Area Tank Farms has been reduced to the present volume of approximately 37 Mgal. [12] Evaporator operations have been extremely effective in minimizing waste volume stored in SRS waste tanks, but because the majority of the waste has been fully concentrated using the available SRS equipment, further reductions via evaporation of the total stored waste volume are not possible. SRS no longer conducts the weapons or NASA-related materials production activities or the weapons-related spent nuclear fuel reprocessing that generated the original waste. DOE is now in the process of deactivating and decommissioning F-Canyon, one of the two chemical separations canyon facilities, and cleaning out and closing the Tank Farms. H-Canyon continues to generate some radioactive waste in performing stabilization missions such as dissolving non-irradiated fuel and blending highly enriched uranium (HEU) for non-defense related use. [10]

Nevertheless, DOE continues to have a need for tank space. Most of the SRS compliant waste storage tanks are already at or near full capacity. Projected available tank space is carefully tracked to ensure the tank farms do not become “water logged,” a term meaning that so much of the usable compliant tank space has been filled that normal operations and waste removal and processing operations cannot continue. Substantial amounts of tank space are required in order to safely and effectively remove tank waste and prepare it for disposal. This includes the vitrification of high-activity sludge waste through DWPF which produces “recycle” that is returned to the tank farms. The preparation of saltcake for disposition also requires significant tank space because the solid saltcake must be dissolved to make it mobile for processing. The dissolution of saltcake typically requires a ratio of approximately three gallons of water to one gallon of saltcake in order to properly dissolve the saltcake back into the salt solution. [13] Tank space for this liquid addition to the tank farm inventory must be available to allow for efficient salt processing and disposition, and ultimately tank farm space recovery. A portion of the available tank space must also be reserved as contingency space should a new tank leak be realized. The tank farms also receive new waste from the H Canyon and wash water from sludge washing. Most of this waste has been volume reduced to the limit of the existing evaporator systems.

The working capacity of the tank farms has steadily decreased, and this trend will continue until several years after the SWPF becomes operational at full capacity, or the system becomes water logged. Beginning in fiscal year (FY) 2001 the working inventory of FTF and HTF has ranged from a low of approximately 1,400 kgal to a high of only 3,800 kgal. [14] The current working inventory (August, 2005) is approximately 2,500 kgal. [14] Diminishing storage space in the SRS Tank Farms is already challenging the site's ability to continue to reduce the overall risk level by processing sludge at DWPF, to prepare batches for the SWPF, and to accept future missions in the country's efforts to clean up legacy nuclear waste material. [9] Aggressive management and conservation efforts are in place to maximize limited storage capacity; however, current waste tank space limitations in the tank farms have the potential to impact the ability to prepare future sludge batches for feed to DWPF. [15]

2.1.2 SRS Federal Facility Agreement

The FFA [1] between the South Carolina Department of Health and Environment Control (SCDHEC), the U.S. DOE, and U.S. EPA was issued to "govern the corrective/remedial action process for site investigation through site remediation and describe procedures for the process." The FFA establishes the regulatory framework for the operation, new construction, and eventual closure of the liquid waste tank systems.

The FFA results in enforceable timetables for the closure of tanks as well as provisions for new construction and prevention and mitigation of releases or potential releases from the tank systems. Pursuant to the FFA, Section XI, SRS submitted applications to SCDHEC for a Clean Water Act/South Carolina Pollution Control Act wastewater construction permit and operation permits for the tank systems on July 8, 1991 and received construction and operating approval from SCDHEC on March 3, 1993 (Permit # 17,424-IW). [16] The FFA, Section IX.E., addresses the eventual removal of tanks and ancillary equipment from service and the final closure of the tanks. For tanks and systems that are governed by a wastewater permit, the closure must be performed in accordance with the South Carolina Pollution Control Act, S. C. Code Ann., Section 48-1-10, et seq. (1985) (PCA) and all regulations implementing that Act.

SRS tanks that do not meet secondary containment standards, as established in the FFA, must be removed from service per the FFA schedule. There are a total of 24 tanks at SRS that do not meet the secondary containment standards and are scheduled for closure by 2022. Two FTF tanks, Tank 17 and Tank 20 have been previously closed, and Tank 19 and Tank 18 are the next two tanks listed in the FFA schedule. [1],[7] Tank 17, Tank 18, Tank 19 and Tank 20 were selected for early closure because of the bulk waste removal efforts during the 1980's (state of readiness for closure) and because DOE had no future operational plans for these four tanks. In addition, due to the close proximity of these tanks, DOE could close out an entire geographic section of FTF.

DOE has determined that there are previous spill sites in the tank systems that are not covered by the wastewater permits which will require response actions under the FFA. These spill sites were previously placed on the FFA by DOE at the time of the FFA approval for evaluation and possible remediation under a separate schedule. [1]

For more information on the regulatory structure under which Tank 19 and Tank 18 are operated and closed, see Appendix B of this document.

2.2 Tank 19 and Tank 18 Location and Design

Tank 19 and Tank 18 are part of the FTF. The FTF occupies a 22-acre site within an area of the SRS commonly referred to as the GSA, which encompasses E-, F-, H- and S-Areas. FTF consists principally of three control rooms, 45,000-feet of transfer lines, six diversion boxes with three pump tanks, two evaporators, and twenty-two waste tanks (Figure 2-1). [2] In 1997, following approval by the State of South Carolina with oversight by EPA, DOE operationally closed Tank 17 and Tank 20 in December 1997 and June 1997, respectively. [2] All four are Type IV tanks in FTF, which were built from 1956 to 1958 (Figure 2-2 and Figure 2-3). [2] In FTF, there are three different types of waste tanks known as Type I, Type III, and Type IV. A fourth type, Type II, is only in HTF. [2] Each of the three types of waste tanks in FTF are described below. The nominal capacity (operating volume) listed for each tank is the standard operating level for each tank and its associated volume. The nominal capacity does not represent the maximum fill limit but rather is a volume that is below the maximum fill limits dictated by the construction parameters of the tank and technical safety standards. Occasionally tank operating volumes were allowed to exceed the "nominal capacity" levels when evaluated against the technical standards and approved on a case-by-case basis. [17] The Type I tanks are single-walled (do not meet the SCDHEC/EPA secondary containment standards, as defined in the FFA), and have a 750 Kgal nominal capacity. The Type III or IIIA tanks are double-walled (i.e., have full secondary containment and meet FFA standards) and have a 1,300 Kgal nominal capacity. The Type IV tanks are single-walled (do not meet the FFA secondary containment standards) and have a 1,300 Kgal nominal capacity. [2] A total of eight Type IV tanks were built, including four in HTF.

Figure 2-1: General Layout of F-Tank Farm [2]

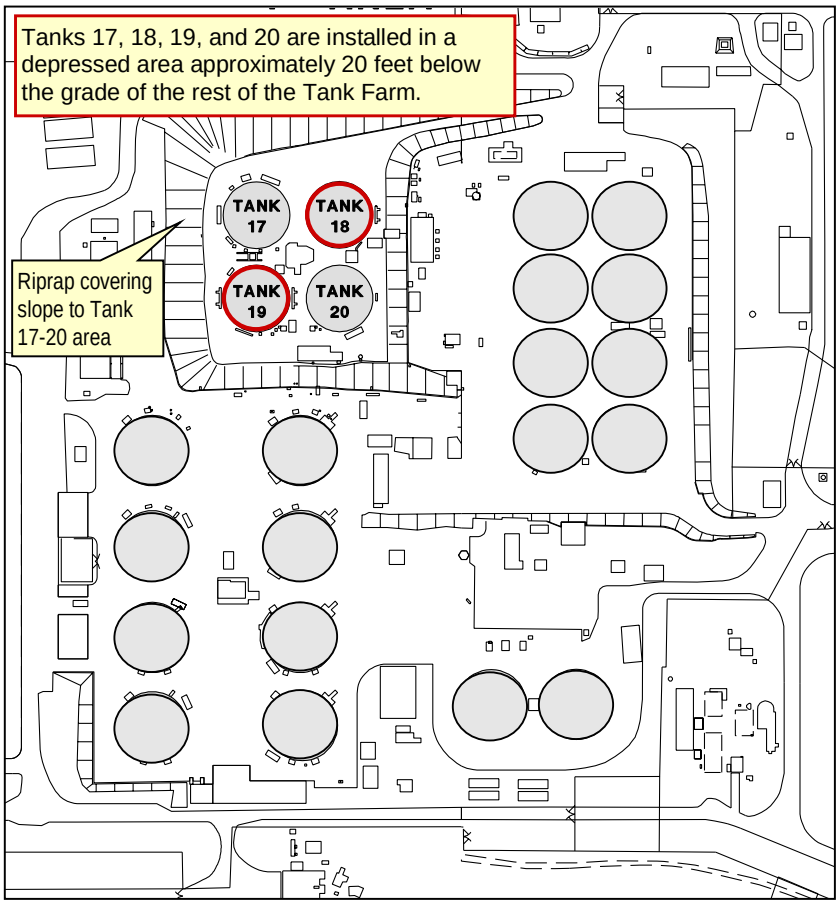


Figure 2-2: Tank 19 and Tank 18 in their Operational Grouping

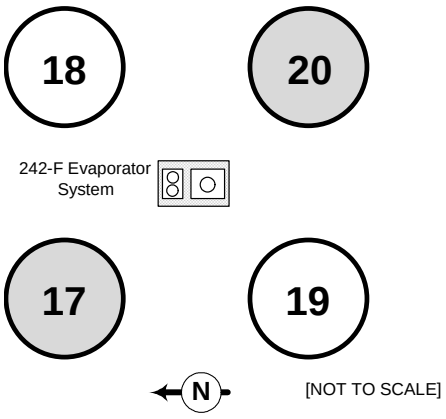


Figure 2-3: Aerial Photograph of Tank 19 and Tank 18 in their Operational Grouping in FTF

Tank 19 and Tank 18 were constructed in 1956, and placed in service in 1961 and 1960, respectively. [10] Tank 19 and Tank 18 have carbon-steel-lined, single-shelled construction with a concrete vault reinforcing the tank walls and a concrete domed roof. See Figure 2-4, Figure 2-5, and Figure 2-6 for construction photos of Tank 17 through Tank 20. Each tank is 85-feet in diameter, approximately 34-feet high at the tank perimeter and approximately 45-feet from the tank bottom to the apex of the domed roof (Figure 2-7). [19],[20],[21],[22],[23] The tank steel conforms to the American Society for Testing and Materials (ASTM) A285-ST Grade B. The concrete basemat beneath the tanks is seven inches thick. A structural steel support system was added to both Tank 19 and Tank 18 prior to the beginning of their respective waste removal campaigns (Figure 2-8). These steel support systems consisting of columns, beams, girders, struts and other associated structures were significant modifications to the tank area. The support systems were designed to bear both live loads (personnel, 50 pounds per square foot) and dead loads such as grating, pumps, and piping (up to 20,000 pounds per square foot). They also were designed to perform to category 3 seismic loads and category 2 wind loads (105 mph). [24],[25]

Figure 2-4: Tank 17 through Tank 20 Concrete Basemat

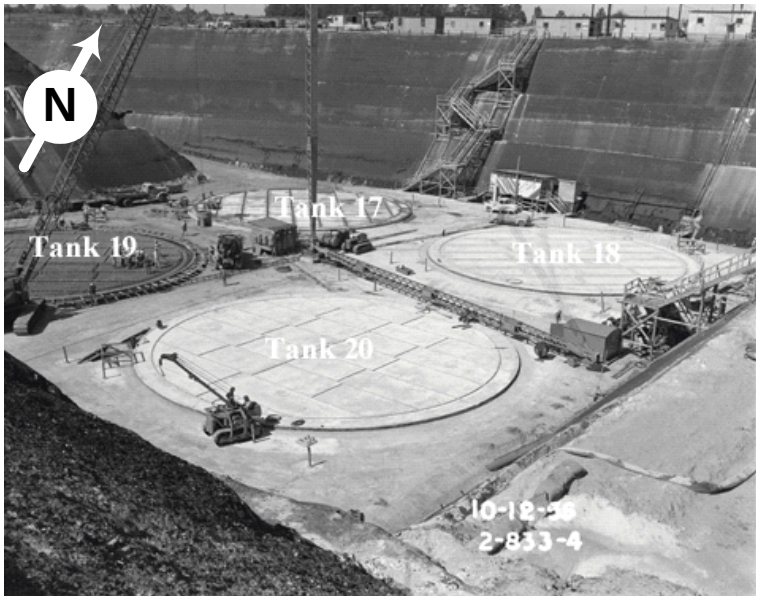
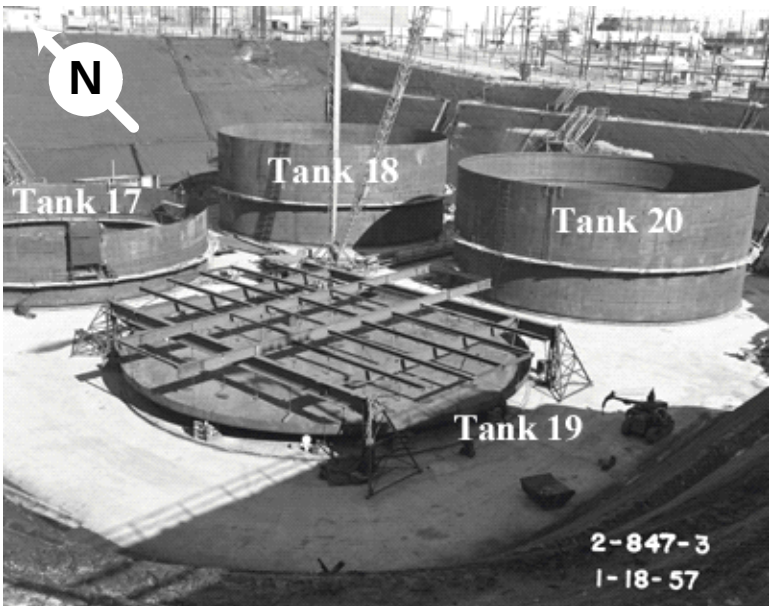


Figure 2-5: Steel Tank Walls



Tank diameter is 85-feet.

Figure 2-6: Tank Concrete Walls

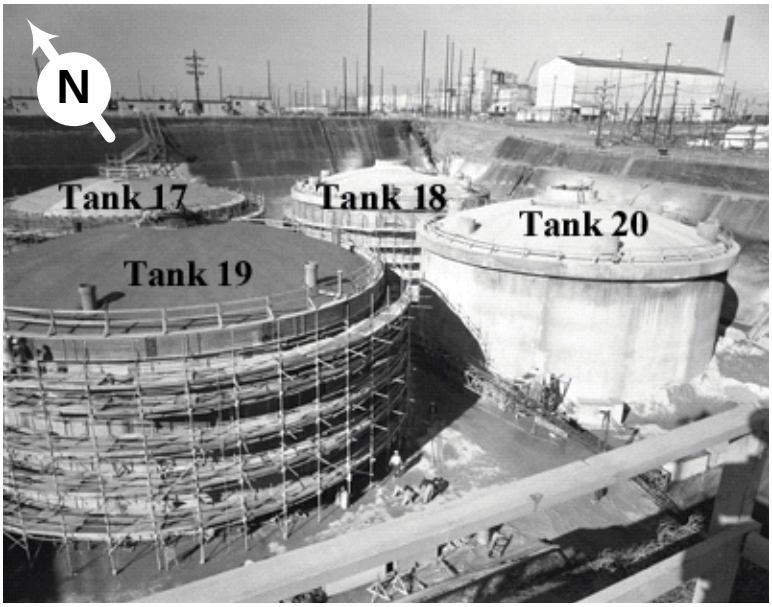


Figure 2-7: Type IV Tank [19],[20],[21],[22],[23]

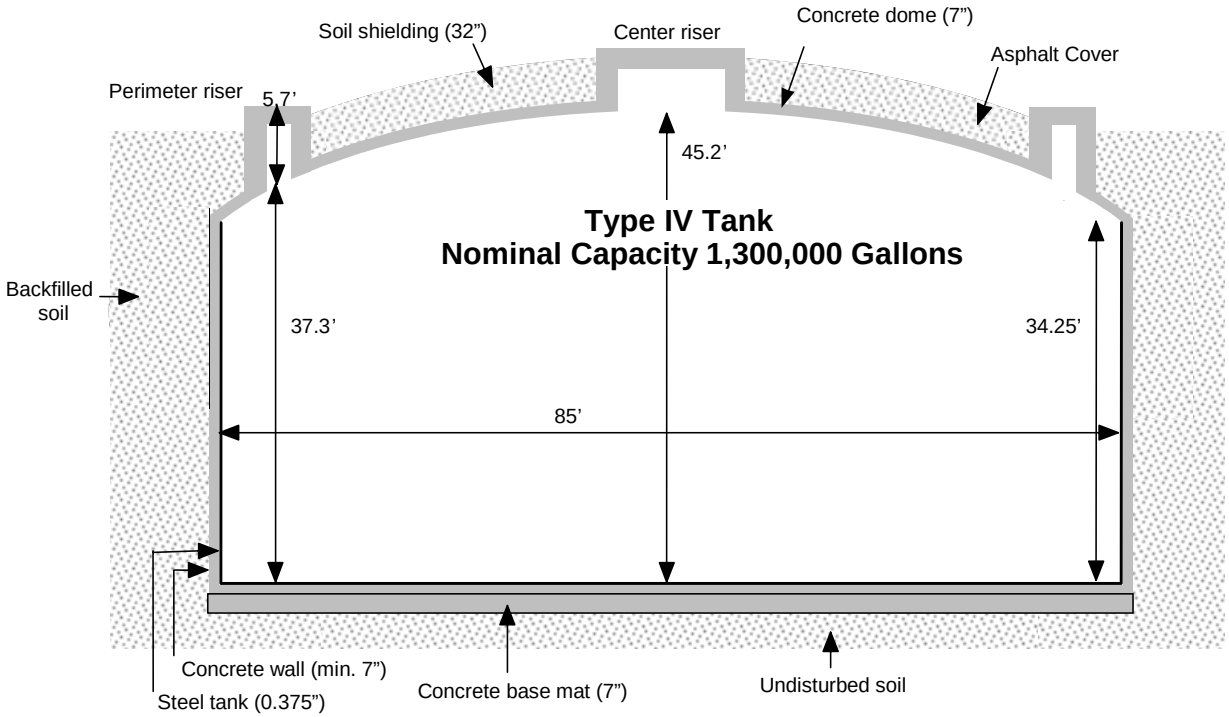


Figure 2-8: Examples of Steel Truss Work

As originally designed and constructed, the dome of Tank 19 had seven access risers. Two additional access points were created during the waste removal process (Figure 2-9). [19] The six original perimeter risers are only two-feet in diameter (opening to tank interior), approximately five-feet long and approximately 37-feet from the bottom of the riser to the tank bottom. The single center riser is approximately eight-feet in diameter (opening to tank interior) and approximately five-feet-long, and 45-feet from the bottom of the riser to the bottom of the tank (Figure 2-7 and Figure 2-10). Access to the center riser is obstructed by steel trusses and a bearing water (BW) tank that were built as part of the tank's infrastructure to support waste removal efforts. Additional access to these points would require removal and disposition of the legacy equipment (failed or abandoned equipment left in the tank). The two additional openings in Tank 19 provided a sample access point in the north side of the tank and a transfer pump access point. Each of the riser ports seen depicted in Figure 2-9 contains legacy equipment from either normal operations or waste removal efforts.

Figure 2-9: Tank 19 Riser Opening Locations and Associated Equipment

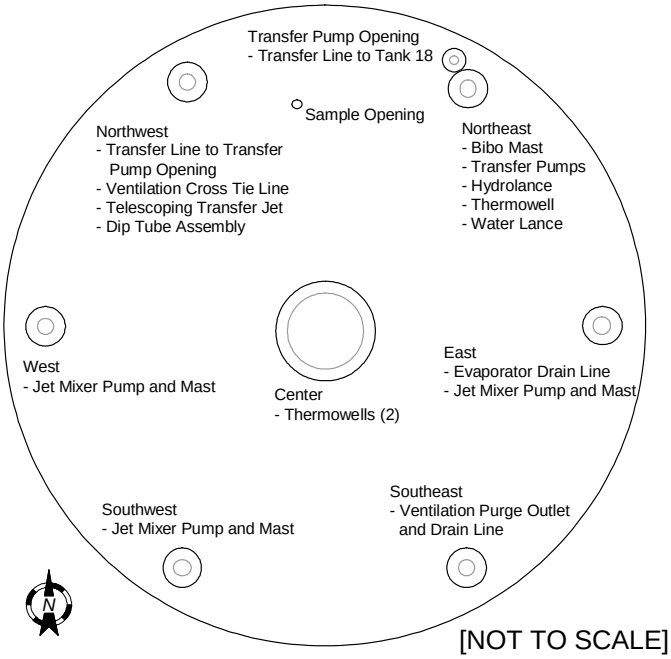
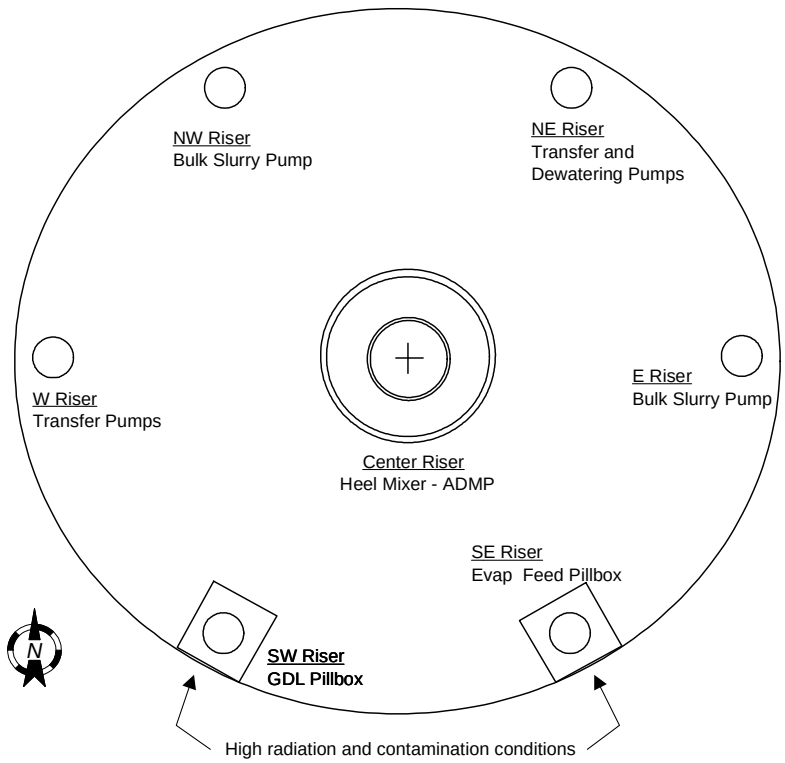


Figure 2-10: Tank 18 Center and Perimeter Risers

The dome of Tank 18 has seven access risers that are identical to those of Tank 19 (Figure 2-11). However, two of the six perimeter risers were unsuitable for access. Both the southeast and southwest risers were modified with vaults commonly called “pill boxes”. These pill boxes were constructed of steel reinforced concrete walls that were built around the risers. These pill boxes were designed to accommodate the original 242-F Evaporator feed equipment and internal piping

to establish specific flow paths. During the early years of operation leaks developed that contaminated the pill boxes. The 242-F evaporator feed systems on Tank 18 were eventually abandoned and the pill boxes were no longer used for their intended purpose (Figure 2-11). The combination of the higher than normal contamination and radiation levels in these areas, coupled with the configuration of the installed equipment, made use of these risers for tank heel removal impractical. [19] The two pill boxes, their risers, and the other five risers have residual contamination associated with them, which is outside the tank proper, but inside the riser. The pill box vaults will be left in place and filled with grout similarly to the other tank top risers.

Figure 2-11: Tank 18 Riser Opening Locations and Associated Equipment



NOTE: The pillboxes are concrete encased vaults.

2.3 Tank 19 and Tank 18 Waste Sources/Attributes

2.3.1 Waste Sources In Tank 19

There were three sources of waste sent to Tank 19 over its service life as a waste receipt tank. [26] The maximum operating volume (tank waste volume) as a result of these waste streams received and stored in Tank 19 was approximately 1,350 kgal which occurred in 1963. [27] The three sources were: (1) the F-Canyon PUREX process, (2) the 242-F evaporator system, and (3) the 242-F evaporator ion-exchange column.

2.3.1.1 The F-Canyon PUREX Process

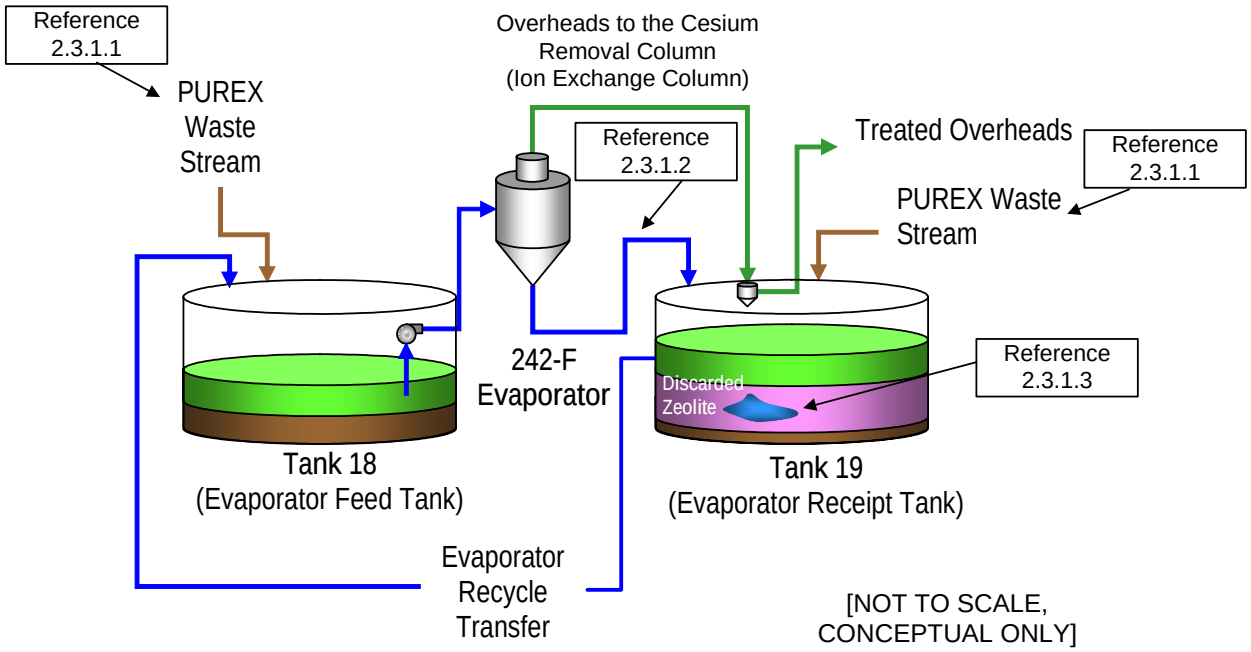
This waste stream was received from the F-Canyon PUREX process. (Figure 2-12). This waste stream resulted primarily from the stripping and recovery of nitric acid from the first and second uranium cycle operation. [28] This waste stream contained sludge solids, insoluble salts, and other inert materials that settled on the bottom of Tank 19 prior to its use as the 242-F evaporator receipt tank (see Section 2.3.1.2 below). These solids are insoluble primarily due to the basic pH of the solution, which is required to protect the carbon-steel tank from corrosion. These waste streams were originally received in Tank 17 and, prior to a settling period, approximately 2,400[3] kgal were transferred to Tank 19 from 1961 through 1972.[3] [26]

2.3.1.2 The 242-F Evaporator System

PUREX waste streams were received from the 242-F Evaporator system into Tank 19 (Figure 2-12). [10] The 242-F evaporator concentrated the liquid waste from the PUREX process using Tank 18, which also received PUREX waste as the feed tank for the 242-F evaporator system. The evaporation process removes the excess water from the PUREX waste streams, concentrating the waste and forming saltcake and concentrated liquid salt solution. The precipitated salt waste is comprised of nitrites and nitrates that crystallized when the waste was concentrated above its saturation levels and precipitated out of the concentrated liquid as it cooled in Tank 19. The saltcake looks like a salt mound (white-gray color) and therefore is known as saltcake. During this process, a small quantity of sludge particles are entrained in the saltcake. The concentrated liquid salt solution is returned to Tank 18 (commonly referred to as an evaporator recycle transfer) for further processing in the 242-F evaporator. Approximately 8,700[3] kgal of concentrated salt waste were received into Tank 19 from the 242-F evaporator between 1962 and 1976. [26]

[3] Volume of material received into a tank are accumulative volumes that have been received during the lifetime of the tank. During tank operation material is removed from the tank and stored in another waste tank to accommodate additional material receipts.

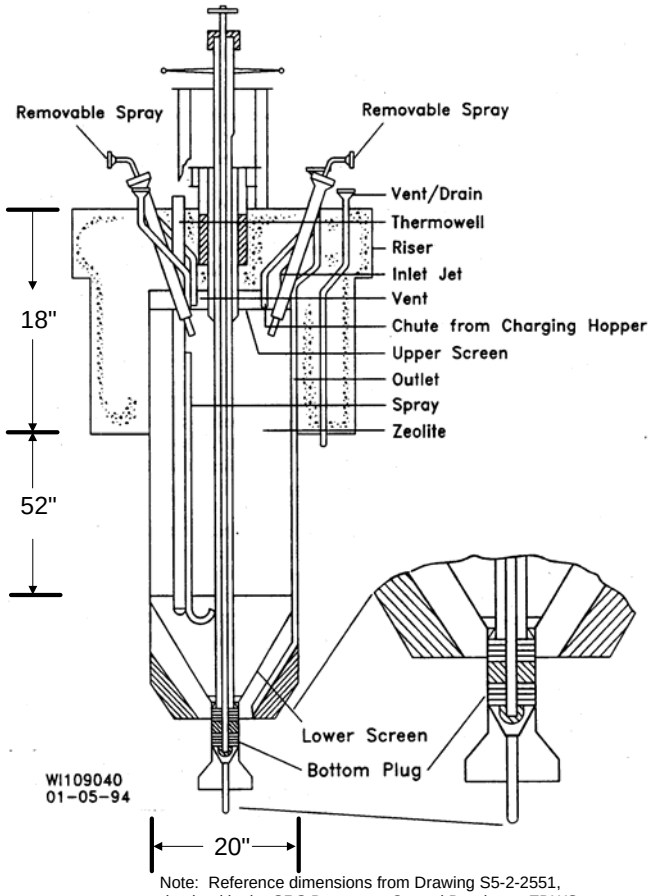
Figure 2-12: Tank 19 PUREX Waste Stream and 242-F Evaporator Diagram



2.3.1.3 Zeolite Resin from Ion-Exchange Column

Zeolite resin was received from the 242-F evaporation process via an ion-exchange column. During the concentration process depicted in Figure 2-12, the excess water that is removed during the evaporation process is known as "overheads". The overheads contain a minimal concentration of radioactive constituents from the evaporation process. The liquid overheads were treated with an ion exchange column that contained a resin commonly known as zeolite. The ion exchange column, also known as the Cesium Removal Column (CRC), was located on the Tank 19 Northeast Riser. The Tank 19 CRC was designed to remove Cs-137 from the overheads stream (Figure 2-13). The CRC contained a sixty-gallon batch of zeolite, and the zeolite batch was changed when flow (pressure drop) parameters were exceeded, or the zeolite became "loaded" with radionuclides to the point that the radiation exposure to workers exceeded established criteria for worker safety on the tank top. The spent zeolite resins and the captured Cs-137 were directly discarded into Tank 19 from 1964 until 1984. Approximately 13 kgal of "spent" zeolite from the CRC were discarded into Tank 19. [29]

Figure 2-13: Cesium Removal Column on Tank Top



Note: Reference dimensions from Drawing S5-2-2551, maintained in the SRS Document Control Database, EDWS

2.3.2 Tank 19 Waste Attributes (Primarily a Saltcake Bearing Tank)

The concentrated waste stream liquid (from the 242-F evaporator) and the unconcentrated waste stream liquid (from the F-Canyon PUREX process) were commingled in Tank 19. Both of these are liquid salt waste solutions. The soluble salts in the liquid are mainly nitrites and nitrates, and the liquid is alkaline as a result of final F-Canyon process adjustment. The liquid density depends on the concentration of soluble salts, and the most prevalent isotope in the liquid is Cs-137 and its daughter product Ba-137m. [5] These liquids also contain sludge constituents.

The sludge (insoluble solids) is comprised of fine particles of settled metal oxides and hydroxides, insoluble salts, and other inert materials. [3] The interstitial area (the space between the insoluble particles) is filled with liquid and gas. [13] As stated previously, sludge constituents are also entrained in the saltcake, but their contribution to the total sludge inventory is minor. [3] When the saltcake was removed from the system, these entrained sludge particles eventually settled on the tank bottom adding to the sludge inventory. The solids collectively behave as a non-ideal fluid and generally have a consistency similar to peanut butter. Thus, a significant amount of mixing energy is required to suspend the sludge in solution to transfer it out of a tank. The radionuclide inventory found in the insoluble solids in Tank 19 is addressed in

detail in Section 2.5. The predominant radionuclide present in the salt waste is Cs-137. Because of the high solubility of the Cs-137, approximately 95% of it is present in the concentrated supernate solution and not in the saltcake. [3] The resin used for the CRC column in Tank 19 was made of crystalline hydrated aluminosilicates. "Resin" is the term used for the solid media used in ion exchange. There are many different types of resins, but the kind used in Tank 19 was zeolite, which is a group of naturally-occurring minerals with specific ion exchange properties known for their ability to capture and retain cesium. [29] When zeolite is aged in a typical SRS waste tank environment it densifies and hardens. [30] When the resin could no longer absorb contaminants (i.e., reached its capacity), the resin was discarded to Tank 19.

The zeolite is difficult to remove from a waste tank; it is resistant to removal using standard mixer pumps due to the significantly higher settling velocities than that of the metal oxides and insoluble salts. It has also been resistant to dissolving with acids. [29] The cesium atoms captured by the zeolite resin become chemically part of the zeolite molecule. The zeolite molecules are insoluble, hence the cesium component is insoluble. Ion exchange reactions are typically reversible, but in this case, a reaction with a very strong acid is needed to release the tightly affixed cesium atoms. Weaker acids such as oxalic acid (OA) have little effect in releasing the cesium. Therefore, washing with water has little to no effect on the removal of cesium from zeolite. The zeolite in Tank 19 held on to the cesium cation (which includes Cs-137) despite being vigorously mixed and washed with 2,300 kgal of inhibited water (IW). IW is SRS well water that has been treated with corrosion inhibitors during salt dissolution. [31] The Cs-137 will continue to decay while chemically bonded to the zeolite and has a half-life of approximately 30 years, i.e., half the Cs-137 will be gone after 30 years, three-quarters of the Cs-137 after 60 years, etc.

2.3.3 Waste Sources In Tank 18

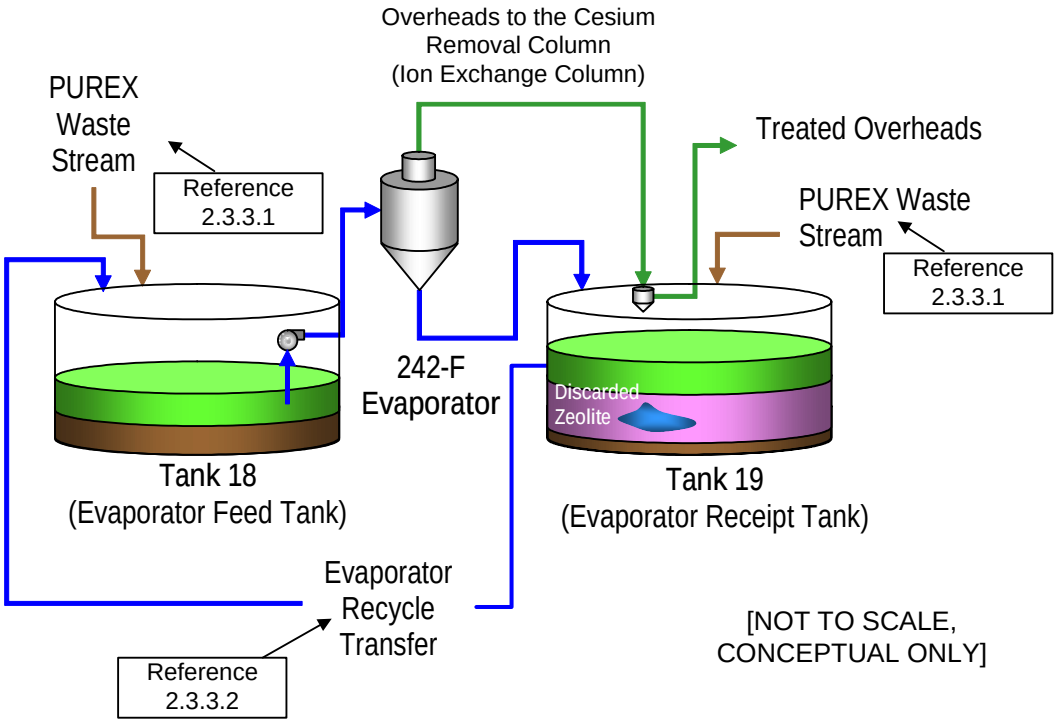
There were four sources of waste sent to Tank 18 over its service life as a waste receipt tank. [26] The maximum historical operating volume (tank waste volume) as a result of these waste streams received into Tank 18 was approximately 1,350 kgal in 1961. [27] The four sources are described below.

2.3.3.1 The F-Canyon PUREX Process

Waste streams were received from the F-Canyon PUREX process (Figure 2-14). These waste streams were the same as those received into Tank 19 from the F-Canyon PUREX process. These waste streams contained sludge constituents that settled on the bottom of the tank. Approximately 33,600 kgal of unconcentrated waste was originally received in Tank 18, from 1960 through 1977.[4] [26]

[4] Volume of material received into a tank are accumulative volumes that have been received during the lifetime of the tank. During tank operation material is removed from the tank and stored in another waste tank to accommodate additional material receipts.

Figure 2-14: Tank 18 Operational Diagram



2.3.3.2 The 242-F Evaporator System

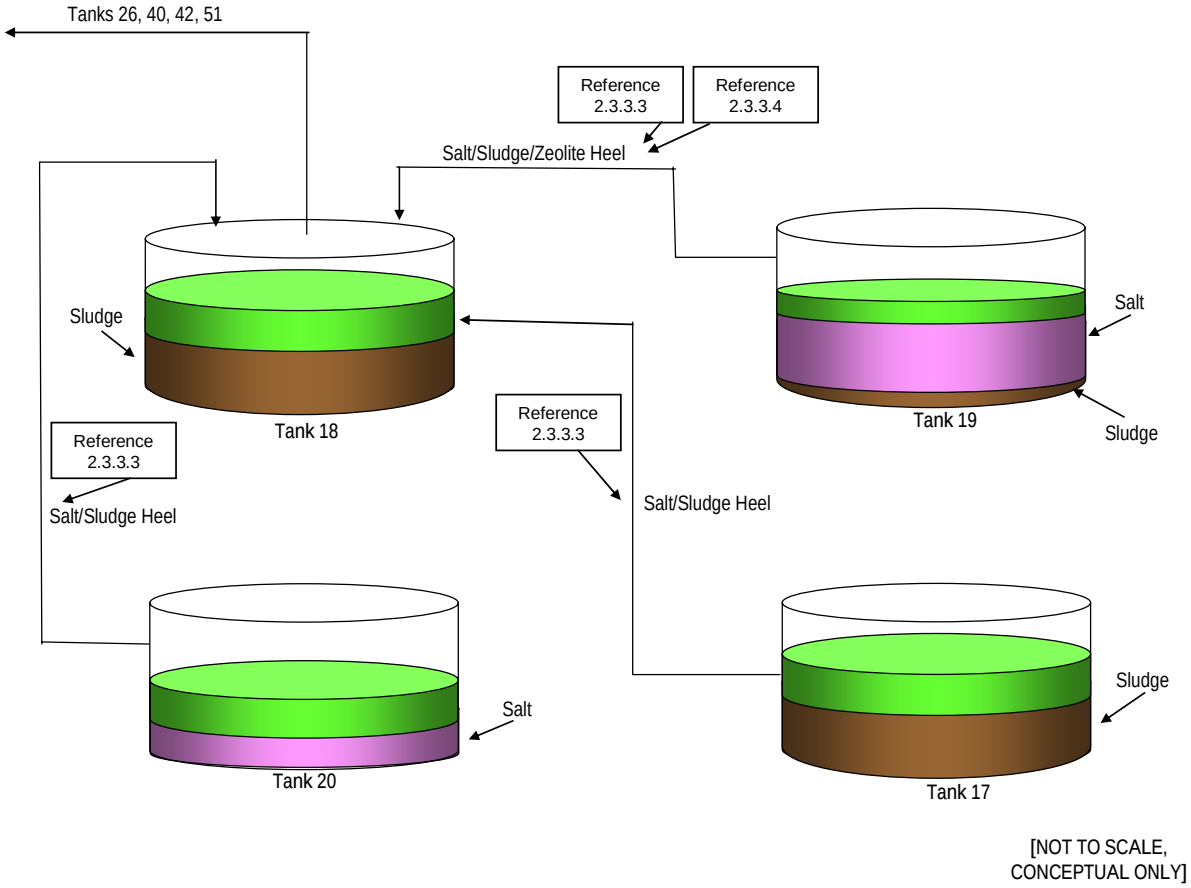
The 242-F evaporator concentrated liquid waste, forming saltcake and concentrated liquid in three receipt tanks (Tank 17, Tank 19, and Tank 20). The concentrated PUREX liquid waste was subsequently transferred back (recycled) to Tank 18 for further processing (Figure 2-14). [10] Approximately 24,000 kgal[5] of concentrated waste were received into Tank 18 from 1960 to 1985. [26]

2.3.3.3 Sludge Removal from Tanks 17, Tank 20, and Tank 19

Settled sludge and insoluble solids from the PUREX process were transferred to Tank 18 via waste removal activity from Tank 17, Tank 19, and Tank 20 (Figure 2-15). [32] Approximately 560 kgal of settled sludge was moved into and out of Tank 18 over its service life. [33] Tank 18 was designed to be the only conventional transfer route out of the Tank 17 through Tank 20 "four pack" area and, therefore, received essentially all of the sludge transferred into this area.

[5] Volume of material received into a tank are accumulative volumes that have been received during the lifetime of the tank. During tank operation material is removed from the tank and stored in another waste tank to accommodate additional material receipts.

Figure 2-15: Tank 18 Waste Removal Diagram



2.3.3.4 Heel Removal which contained Zeolite

Approximately 18 kgal of solids, which included 2.5 kgal of zeolite, was transferred into Tank 18 from the Tank 19 heel removal process, illustrated in Figure 2-15. [33],[29]

2.3.4 Tank 18 Waste Attributes (Primarily a Sludge Bearing Tank)

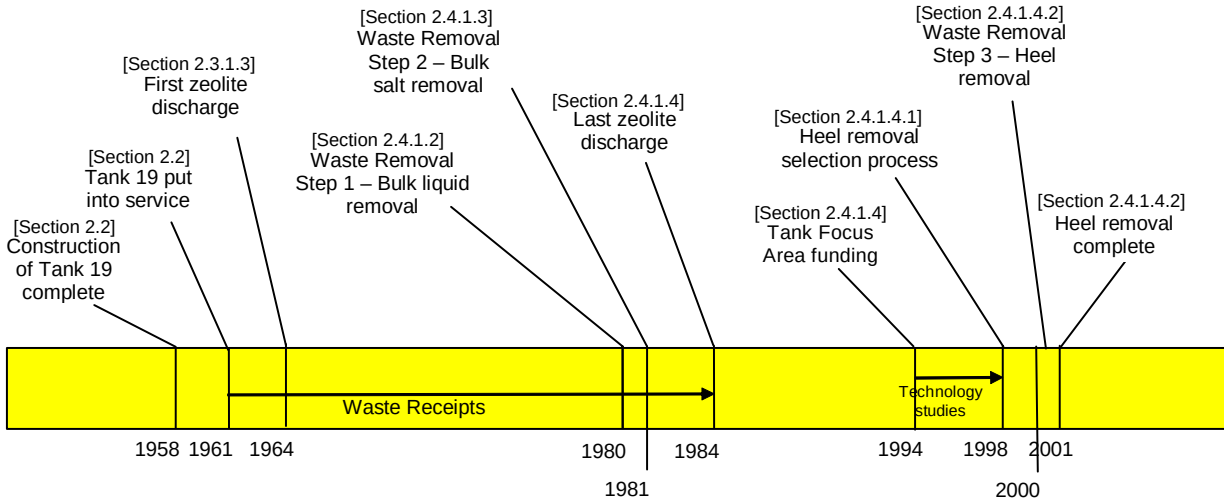
The Tank 18 waste attributes are identical to those of Tank 19, described above in Section 2.3.2 with the exception that Tank 18 had no saltcake.

2.4 Tank 19 and Tank 18 Waste Removal

2.4.1 Tank 19 Waste Removal

Beginning in January 1980, DOE began a two-phase, three-step plan to remove waste from Tank 19. [31] Phase One of the plan, bulk waste removal, comprised steps one and two, removal of the concentrated liquid waste and then removal of the bulk saltcake. Phase Two of the plan comprised step three, heel removal. See Figure 2-16 for the Tank 19 historical timeline which includes waste removal activities.

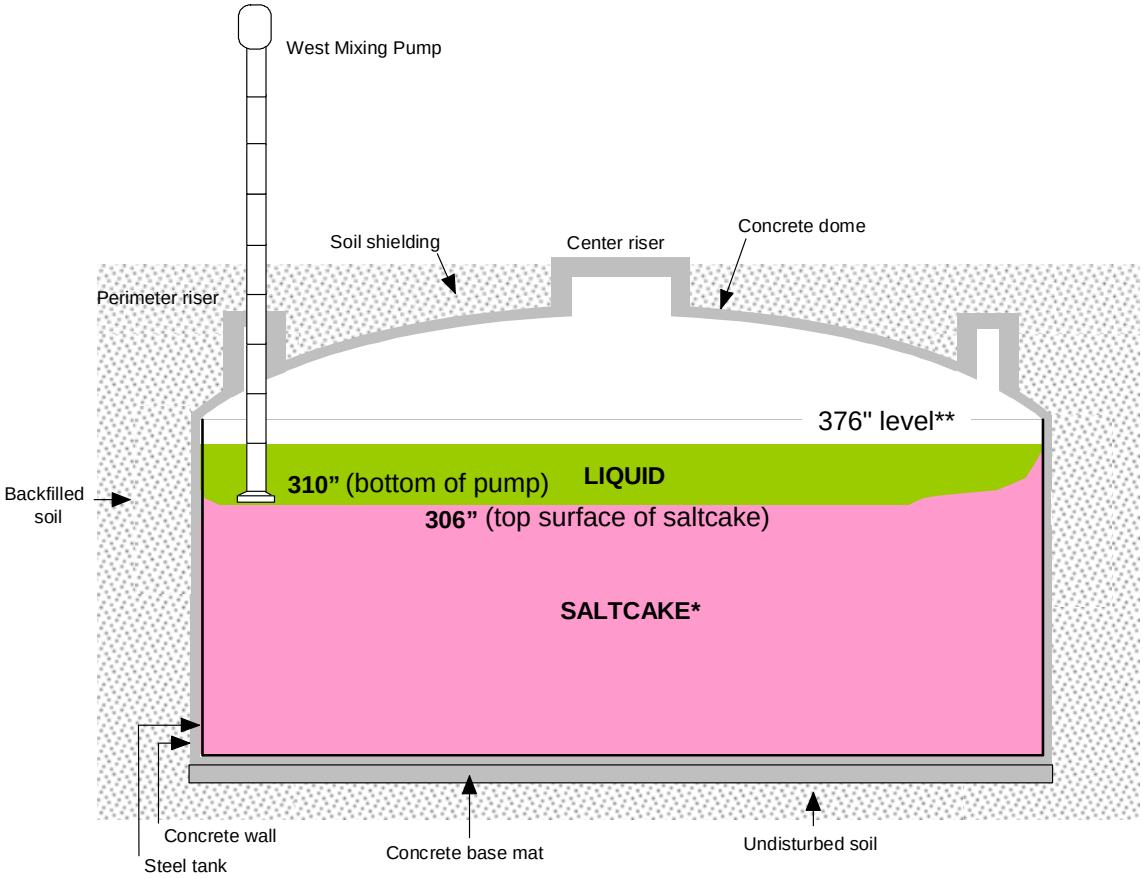
Figure 2-16: Tank 19 Historical Timeline



2.4.1.1 Tank 19 Initial Condition Prior to Waste Removal

The Tank 19 contents prior to beginning the waste removal campaign were approximately 1,300 kgal of total waste. Approximately 1,100 kgal of this waste was in a wet solids form (saltcake, zeolite, insoluble solids). Approximately 400 kgal of the remaining portion of the waste was either “free-standing” (liquid above the saltcake) or interstitial liquid (liquid in the saltcake) (Figure 2-17). [31],[11]

Figure 2-17: Tank 19 Initial Condition Prior to Waste Removal [31]



* Saltcake includes crystallized salts, interstitial liquid, insoluble solids and zeolite solids.
**Numbers in inches indicate distance above the tank bottom.

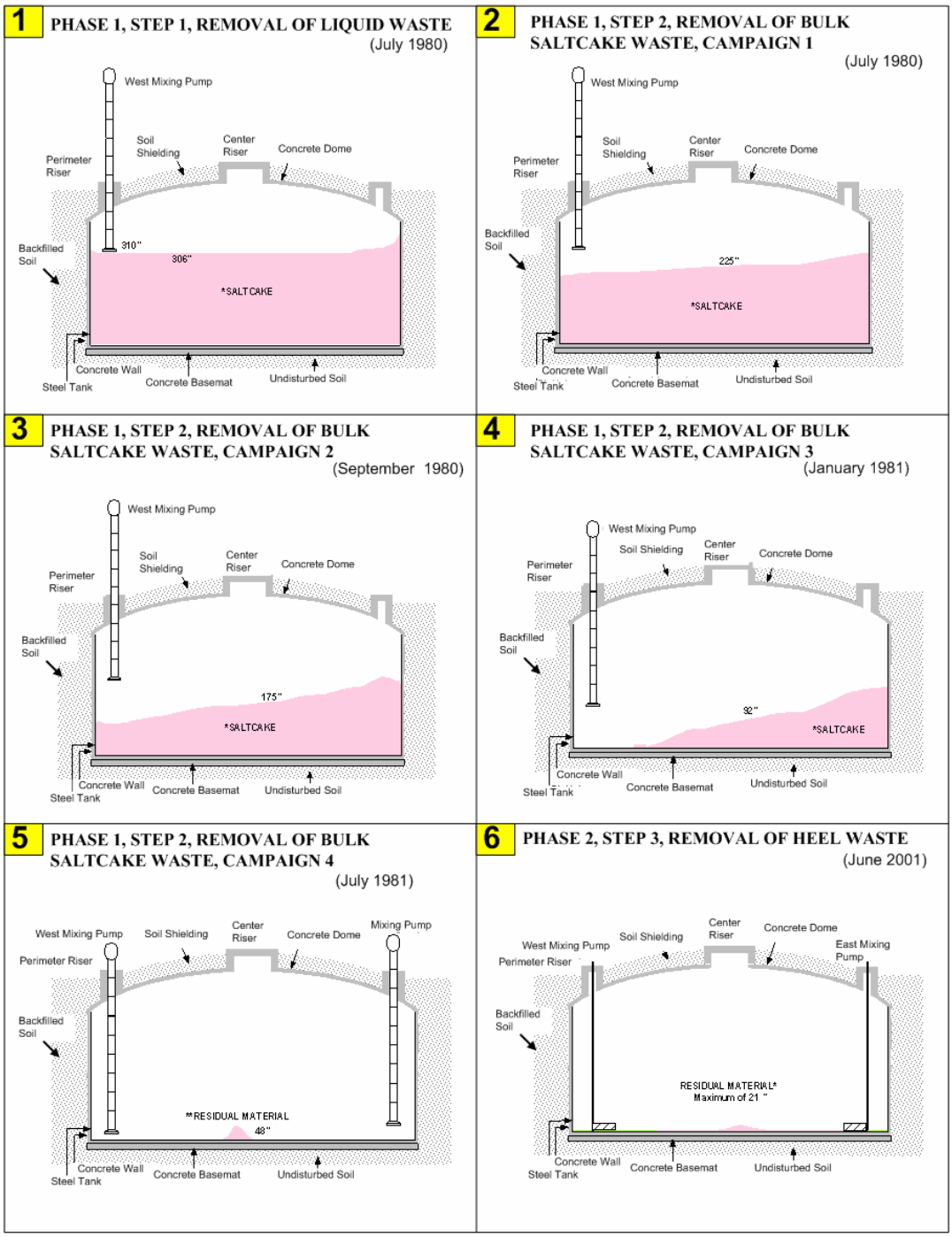
(January 1980)

[NOT TO SCALE]

2.4.1.2 Phase 1, Step 1 - Removal of Liquid Waste

During this step, the concentrated liquid waste, both free standing and some interstitial was removed in three transfers from January 1980 to May 1980, the end of Step-1 is indicated in Figure 2-18. The liquid waste was removed passively, i.e., with no slurry pump operation. A fixed-length transfer jet was used as the prime mover of this waste stream**.** A total of 277 kgal was removed and the liquid waste stream was transferred to Tank 18 for future processing/storage throughout FTF. After the desired volume of liquid waste was removed, a saltcake formation with an estimated volume of approximately 1,100 kgal remained (Figure 2-18, Panel 1). [11]

Figure 2-18: Tank 19 Salt Removal Campaign (July 1980 – June 2001) [31]



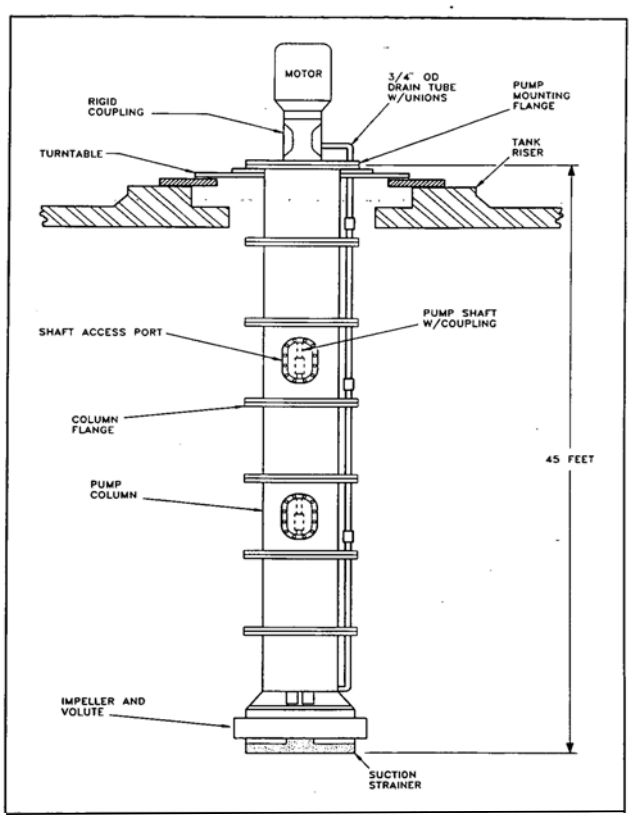
*Saltcake includes crystallized salts, interstitial liquid, insoluble solids and zeolite solids.
**Residual material includes free liquid, interstitial liquid, insoluble solids and zeolite solids.
***Each panel represents condition at the end of each step.

NOTE: The numbers related to saltcake height represent approximate average height, except for panels 5 and 6, where the numbers represent maximum height.

2.4.1.3 Phase 1, Step 2 - Removal of Bulk Saltcake Waste

During this second step, SRS installed two standard mixer pumps (Figure 2-19) which required installation of new infrastructure (steel trusses) to prepare Tank 19 for saltcake removal. The two mixer pumps were mounted on the new supporting truss work. Each mixer pump had a 150-horsepower (hp) motor and ran at 1,800 revolutions per minute. Each pump had a 45-foot shaft with cylindrical support columns that were filled with clean water. The column was pressurized to prevent cross-contamination between the waste and the top part of the tank. IW was added to the tank and the mixer pumps exerted a sweeping liquid jet action on the saltcake to promote its dissolving back into solution. In addition to the new mixer pumps, service upgrades were performed on the Tank 19 heating and ventilation system, IW addition system, and BW support system. A telescoping transfer jet was employed to move the dissolved salt solution out of Tank 19 into Tank 18. This salt solution was eventually transferred to other tanks in FTF where it was stored while awaiting further processing by an evaporator (i.e., reconstitution of the saltcake in a different waste tank). The ultimate disposition of the salt solution will be at either the Saltstone Disposal Facility vaults or the Federal Repository via the Salt Waste Processing Facility (SWPF) and then DWPF. [9]

Figure 2-19: Typical Standard Mixer Pump [34]



[NOT TO SCALE]

To begin the phase 1, step 2 process, non-radioactive IW was added to Tank 19 to dissolve saltcake to form a salt solution. IW was added to Tank 19 in four different campaigns from July 1980 to July 1981. The saltcake was washed using the mixing pumps to promote salt dissolution and removal of soluble radionuclides (Table 2-1). The mixer pump(s) were lowered farther into the tank in each successive run. During each campaign the solids and zeolite resins that were entrained within the saltcake were extensively washed, stripping out the soluble Cs-137 and other soluble species and leaving behind Cs-137 chemically bound to zeolite and insoluble particles. The insoluble particulate matter entrained in the salt and interstitial liquid settled to the tank bottom along with an existing sludge layer. As expected, no measurable amount of zeolite was removed during this step because its insoluble, hardened, and fast-settling characteristics make it resistant to this salt removal method. [31]

Table 2-1: Tank 19 Saltcake Dissolution Campaigns [31]

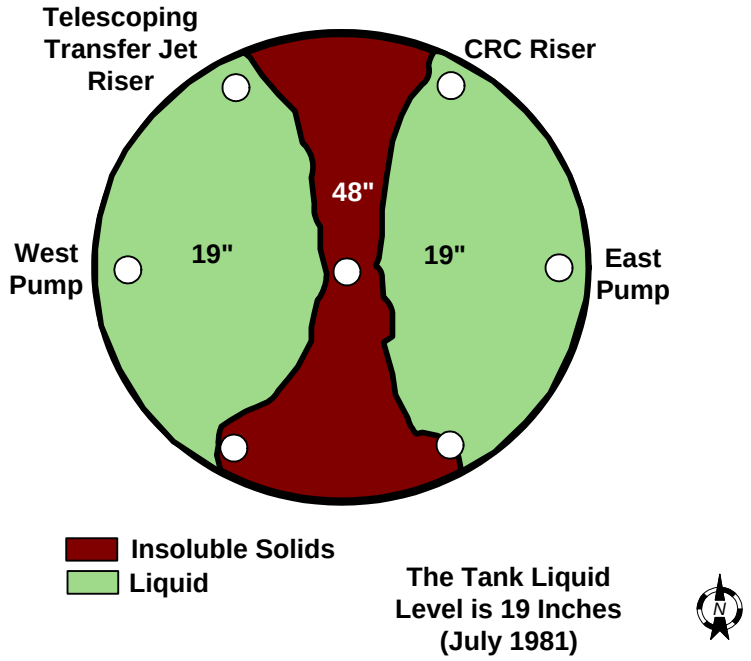
Dissolution Campaigns	Saltcake Removed (kgal)	Water Removed (kgal)	Figure 2-18
1st (7/80)	172	236	Panel 2
2nd (9/80)	281	442	Panel 3
3rd (1/81)	271	529	Panel 4
4th (7/81)	309	1,175	Panel 5
Overall Results	1,033	2,382	

During the four saltcake dissolution campaigns approximately 1,100 kgal of saltcake was dissolved using about 2,300 kgal of IW. The final salt solution transfer was completed July 1981 (Figure 2-18, Panel 5).

Following the final salt solution transfer, the tank residual material was sampled and mapped. There were approximately 33 kgal of solids remaining in Tank 19 in an hour-glass shaped mound (Figure 2-20) that was attributed to the location of the slurry pumps in the east and west risers. The mound appeared to be continuous with the highest point approximately 48 inches tall. This material was known as the heel that will be addressed in Phase 2, step 3. Sample analysis revealed that the approximate composition of the remaining solids was:

- 20 kgal of insoluble solids (fine particles of settled metal oxides, insoluble salts and other inert materials)
- 13 kgal of zeolite.

Figure 2-20: Hourglass Shaped Landscape following Bulk Salt Removal [31]



2.4.1.4 Phase 2, Step 3 - Removal of Heel Waste

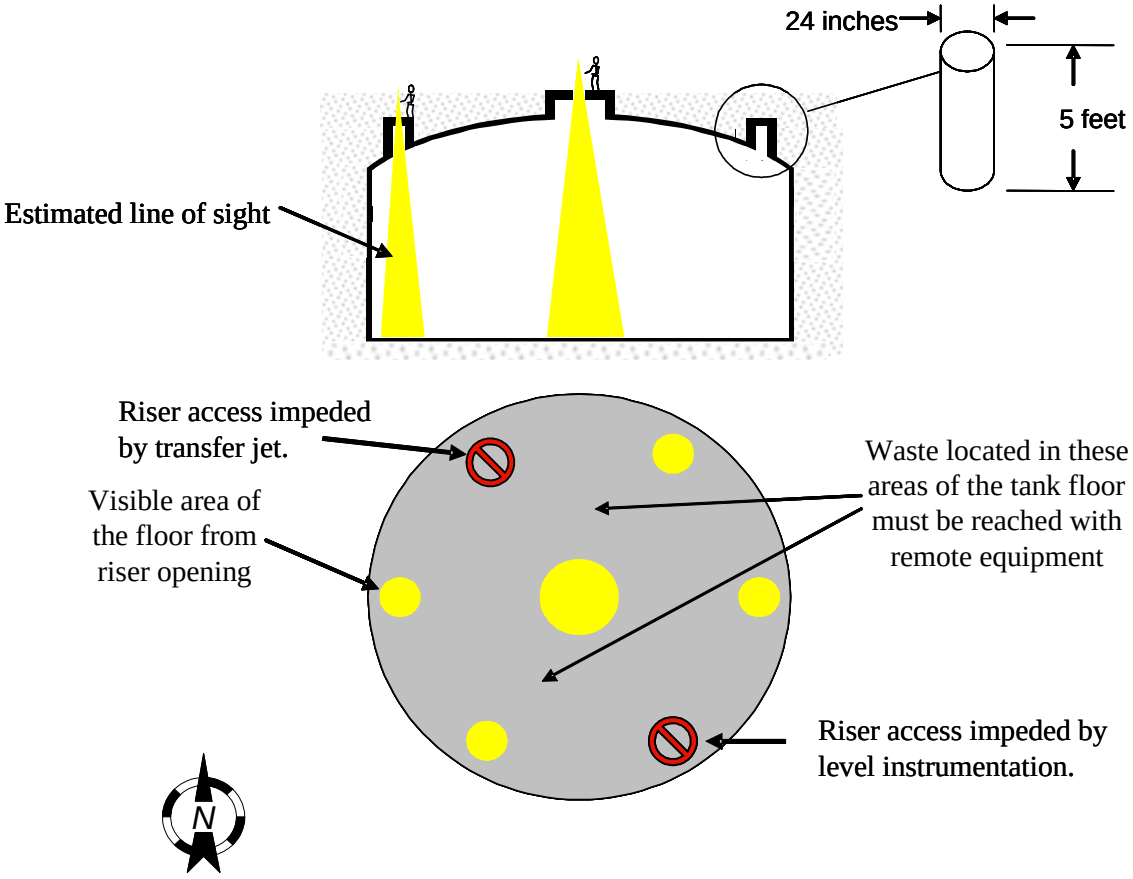
After completion of the bulk saltcake removal in July 1981, Tank 19 remained inactive (except for routine surveillance and maintenance) until June 2000. However, beginning in 1984, studies were conducted to develop or identify the best new and innovative technologies for heel removal. Requirements and factors impacting technology selection included physical configuration of the tank, the physical properties of the zeolite, FFA regulatory commitments (discussed previously), limited storage space in the tank farm system, and occupational radiological exposure risks. [29],[35],[36]

Physical configuration factors impacting selection of the best technology for heel removal in Tank 19 included the carbon steel tank construction material, the large size of the tank, tank floor design, and limited visual and equipment manipulation access in the tank interior. The carbon steel tank wall and floor material are incompatible with nearly all acidic cleaning solutions due to corrosion concerns. The large size of the tank (85-feet in diameter by 45-feet high at the top of the domed roof) also influences the types of equipment that can be successfully deployed. In addition to its large size, the floor of the tank is essentially level with no low spot where material could collect. Rather residual material would collect in areas along the floor dependent on fluid movement within the tank. The design configuration of tank risers (access openings) provides very limited access to the tank interior. The tank has one center riser and six perimeter risers. Each perimeter riser is a cylinder nominally 24 inches in diameter and approximately five-feet long. The cylindrical shape of the risers restricts the ability to view and manipulate equipment inside the tank. Two small (10-inch diameter) openings are installed in the center riser. Existing tank equipment, including a transfer jet and level instrumentation, are installed in two of the perimeter risers. Therefore, these two risers were undesirable locations for heel removal

equipment, as change out of equipment in these risers would result in significant occupational radiation exposure and generation of additional radiological waste. Furthermore, riser configuration above the tank top limits direct equipment access and allows only a restricted view of the tank floor (Figure 2-21). [19],[20],[21],[22],[23]

In addition, the size of the access ports limited the manipulation of long-handled mechanical tools. Creating additional access ports had the potential to weaken the tank top, increasing the risk of tank top collapse. Ensuring tank-top integrity added significant cost to the waste removal efforts. Finally, due to internal obstructions and access port geometry, choices were very limited as to the kinds of remote equipment that could be successfully deployed.

Figure 2-21: Diagram of Tank 19 Access Area for Heel Removal Equipment [19],[20],[21],[22],[23]



As discussed in Section 2.3.1.3, zeolite was used to remove cesium from the evaporator system. When the zeolite became fully loaded with cesium, it was discharged into Tank 19. Zeolite becomes more dense and hardens over time, making it difficult to slurry for removal from a waste tank. [30] An estimated 13 kgal of zeolite were placed in Tank 19 from 1964 to 1984. During this 20-year period, numerous batches of zeolite were directly discarded onto various layers of saltcake formed from waste transferred into Tank 19. Bulk waste removal efforts were unsuccessful in removing the bulk of the inert zeolite resins with the associated short-lived Cs-137. [31]

Because of the lack of available tank space, as discussed in Section 2.1.1, an important criterion for the choice of the heel removal technique is the amount of additional liquid waste that the removal technique would generate.

As with all work in the tank farm radiological environment and consistent with the use of ALARA requirements, radiation exposure risks to workers was a primary consideration during technology selection.

In 1994, DOE authorized a Tank Focus Area (TFA) initiative (including national and international groups) that provided funding to study and design/build new waste retrieval devices. [35] Additional information describing these and other waste retrieval equipment options are contained in Appendix C.

2.4.1.4.1 Technologies Selection Process and Testing for Tank 19 Heel Removal

2.4.1.4.1.1 Selection Process

In 1998 SRS used a systematic selection process which was documented in a System Engineering Evaluation (SEE). [36] This study investigated twenty-one options, including those technologies included in the TFA studies that were broken down into mechanical and chemical retrieval methods. The study graded the options on ten selection criteria ranging from cost and schedule to technical maturity. Technical maturity is a judgment regarding the proven field applicability of an option being investigated with considerations given to the waste characteristics and probability of success. Options that were in the conceptual design phase (many of the TFA options fall into this category) would tend to get lower scores versus options that had a proven field record. Knowledgeable tank farm operations and engineering personnel identified potential heel removal technology candidates based on research and experience. The technologies were evaluated based on the following criteria: [36]

- Safety – The degree to which the technology can be constructed and operated with regard to protection of the occupational workers and the public in the area of industrial safety and radiological controls.
- Effectiveness/Probability of Success – The degree of confidence that the technology will perform the function for which it was proposed.
- Complexity – The degree of complexity of the technology with regard to design, construction, testing and operation.
- Technical Maturity – The degree to which the technology has been developed and/or has been demonstrated in a radioactive waste removal application.

- Authorization Basis Impact – The degree of changes required in the tank farm facility safety basis documentation to implement the technology.
- System Integration – The degree to which the technology is compatible with existing regulatory programs, processes, and infrastructure.
- Reliability – The degree of confidence as to how well the technology equipment will perform the needed function without failure.

The selection process investigated a broad range of technologies that can be grouped into three general categories: (1) mechanical agitation equipment, (2) mechanical retrieval equipment, and (3) chemical treatment. [36]

Examples of mechanical agitation options include mixer pumps (standard and advanced design), pulse tube agitators, submersible agitators, and blade mixers as described below.

- Standard Mixer Pumps - This standard mixer pump, previously used in waste removal projects, consists of a submerged pump assembly driven by a top-mounted 150-hp electric motor. The pump draws suction from the tank bottom and immediately shoots out opposing jet streams in the horizontal plane. The jetting action as a result cleans and scours a circular area. This mixer pump scored low against the criteria of effectiveness because it was determined to have an inadequate cleaning area.
- Advanced Design Mixer Pump (ADMP) – This mixer pump is similar to the standard mixer pump; however, it is powered by a 300-hp motor (versus a 150-hp motor for the standard mixer pump).
- Pulse Tube Agitator – This agitator uses a pulse jet pump to provide a charge vessel with either vacuum or pressure. Under vacuum, the charge vessel pulls in liquid. Pressurized air discharges liquid from nozzles to mix solids. The pulse tube agitator scored low on the criteria of authorization basis impact and effectiveness.
- Jet Mixing Pump – This pump consists of a submersible motor and propeller that discharges liquid through a shroud to mix solids. This technology, when deployed with the capability to oscillate, scored high for the criteria of probability of success and was the selected technology for Tank 19 heel removal.

Examples of mechanical retrieval options included robotics, vacuum conveying systems, mechanical conveyance systems, robotic arm systems, and alternative pumping systems.

- Robotics – Various configurations of remotely operated platform were evaluated. These platforms are used to move retrieval equipment in the tank. This technology scored relatively low on the criteria of technical maturity because of a potential lack of mobility due to obstacles and hose management issues.
- Vacuum conveying systems – This technology retrieves the heel by using displaced air to create a vacuum that lifts and transfers the material. This technology scored relatively low on the criteria of technical maturity and complexity because of limited history in a large radioactive waste tank and design features required for potentially contaminated air handling equipment.

- Robotic arm systems – These remotely controlled systems have the capability to extend vertically and horizontally to position cleaning equipment in the tank. This technology scored relatively low on the criteria of technical maturity and reliability because of the complex equipment associated with large robotic arm systems required to reach the expansive interior areas of the tanks.
- Alternative pumping systems – These mechanical retrieval systems consist of pumps that operate on the principles of positive displacement, centrifugal force, or vacuum. Evaluation of these various pumping systems resulted in the selection of submersible centrifugal pumps for transferring the residual heel out of Tank 19 because of high scores for system integration, authorization basis impact, reliability, and complexity.

Chemical treatment with OA was evaluated as an option for Tank 19 heel removal during the technology selection process. OA cleaning had been performed at SRS with varying levels of success.

Tank 16 waste removal experience showed repeated OA rinses were effective in removing sludge but the acid had to be of sufficient strength and heated to be effective. Later research showed OA was ineffective in dissolving insoluble sodium alumina-silicate compounds (such as that in the annulus of Tank 16). [37] Also, in 1984, two OA washes were completed in Tank 24 on the zeolite heel left behind after bulk waste removal activities were completed. It was believed that two slurry treatments with OA would dissolve up to 70 wt% of the insoluble solids into a finely divided solid that would be easier to mix. The results were less than expected and only approximately 45% of the solids were removed. [29] Approximately 10,000 gallons of zeolite remained in Tank 24 after the chemical cleaning campaign. [38] Tank 24 experience showed OA helped remove some zeolite but not as effectively as it had removed the sludge in Tank 16.

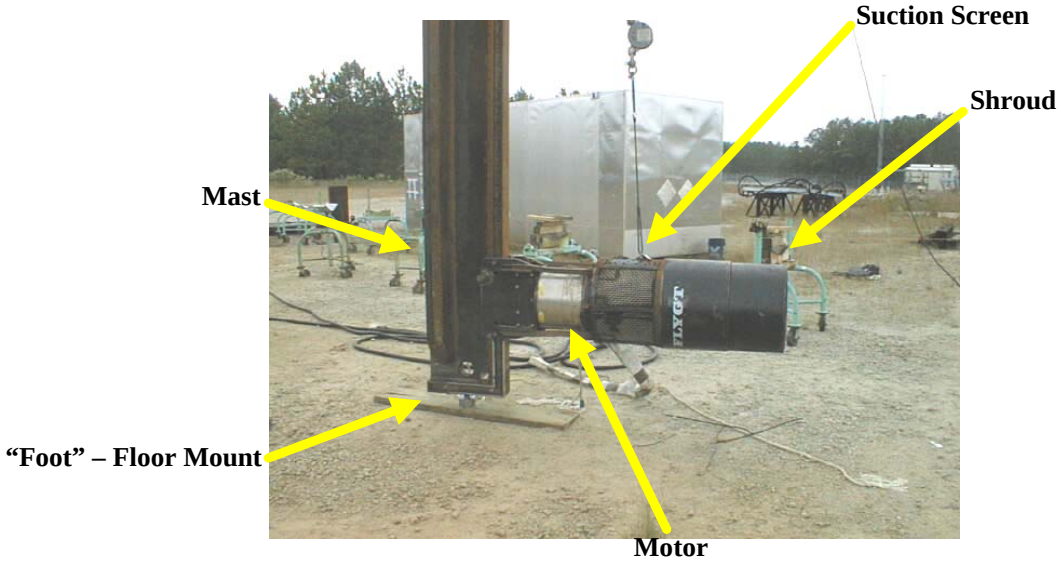
The Tank 19 technology selection evaluation rejected chemical cleaning as a heel removal option because of concerns about down-stream process impacts (system integration) and authorization basis impacts. The system integration impacts included, but were not limited to, tank space requirements to neutralize the acid and DWPF processing issues with neutralization products (oxalates and glass qualifications). Safety Basis[6] impacts currently preclude use of acids because of criticality concerns, tank degradation issues, and hydrogen production.[7] [39]

The study concluded that use of three 50-hp rotating submersible jet mixer pumps (Figure 2-22) along with a recycle liquid waste stream was the preferred heel removal method. This option was chosen because its ability to keep the aged zeolite (which is the type present in the waste tanks) suspended longer because it is faster settling and denser than fresh zeolite. [30] Traditional mixer pumps would quickly lift the zeolite particles into suspension, but once the jetting action of the pump stream passes by, the particles would just as quickly settle. Therefore, removing the zeolite would require a mixing regimen that promoted overall tank agitation, rather than the localized mixing provided by traditional equipment. The recycle stream was also

[6] The Safety Basis consists of specified parameters and controls to ensure the safe operation of DOE facilities and protect the public, workers, and the environment.
[7] The use of acids for tank cleaning is not analyzed in the current Concentration, Storage, and Transfer Facilities Documented Safety Analysis (DSA). [39]

necessary because the studies revealed that a significant number of mixing campaigns and transfers would be required to achieve desired results. [40] This recycle stream also would conserve overall tank farm space by reusing the waste stream as a transfer medium.

Figure 2-22: Prototype Mixer Pumps



2.4.1.4.1.2 Testing

The selected technology for Tank 19 was extensively tested prior to installation and operation in the radioactive waste tank. The testing of Flygt mixers for Tank 19 was performed at SRS and other facilities (both partial and full scale). Scale testing revealed the 50-hp mixers imparted the tank agitation characteristics desired for zeolite removal, and the overall tank agitation option was consequently selected as the preferred method. [41] Mixer pump testing was conducted at three locations; ITT Flygt Corporate Headquarters located in Trumbull, Connecticut, Pacific Northwest National Laboratory (PNNL), Richland, Washington, and, at SRS in Aiken, South Carolina. Testing indicated three oscillating free jet flow mixers would be effective for Tank 19. [42] The testing results were used to determine optimum mixer placement, shroud enhancement, run speed optimization, durability, vibration analysis, and failure/recovery analysis. Based on the results of these tests, the optimum placements for the three mixer pumps were identified. Three mixer pumps were purchased, and functionally tested at SRS, and installation began in July 2000. The mixer pumps were placed on the bottom of the tank in the locations that the tests indicated would be effective. Each pump was attached to a 45-foot rotating five-ton mast. Each mixer was flow rated at 17.5 kgal per minute. [42]

2.4.1.4.2 Tank 19 Heel Waste Removal Operation and Results

In addition to the use of the selected technology, a different technique was used to remove the heel from Tank 19. In the past, fresh water had been added to a tank and mixer pumps had stirred the contents. The suspension had then been transferred out as a slurry mixture. However, the zeolite solids in Tank 19 were fast settling, and it was unlikely that a single transfer would

clean the tank. In fact, early studies suggested that it would take dozens of water additions and transfer sequences to transfer a significant amount of heel to another tank. [40] To complicate matters, the tank farm system did not have the available space to handle clean water additions in the volumes needed. Therefore, a technique was devised to transfer the solids to another tank but reuse existing tank liquid as the transport medium. This was accomplished by making an initial addition of 280 kgal of existing tank farm liquid (comprised of tank ballast IW additions plus addition of sodium hydroxide to raise pH and control corrosion from 1982 to 2000), stirring the contents with submersible mixers, and then transferring the slurry to Tank 18. The fast-settling solids quickly dropped to the bottom of the receiving tank (Tank 18), allowing that liquid to be used as the transfer medium again. The liquid was then decanted (removed from the top portion of Tank 18) back to Tank 19 to start the process over again. The waste solution transfer cycles were continued until they were no longer effective (i.e., no significant amount of residual heel continued to be removed). See Figures 2-23 and 2-24. Heel removal operations stopped after using forty-six such cycles, which would have been equivalent to adding over 10 Mgal of waste to the tank farm system. [41] As discussed in Section 2.2.1, this volume would have required more space than the total available working inventory of the tank farm at any time during the four years evaluated. In summary, the “reuse” method adopted resulted in significant tank space savings during execution of this step.

After the final heel removal transfer from Tank 19 to Tank 18 was completed, the walls of Tank 19 were washed with water from a spray nozzle installed in the center riser. [41] This final wall wash was done in preparation for closure. The spray nozzle had pan and tilt capability and an operating capacity of 65 gpm (gallons per minute) at a pressure of 175 pounds per square inch (psi). This spray wash system was used for washing the tank walls from top to bottom and for focused washing in areas with deposits. Approximately 3.2 kgal of additional waste was created during this effort and transferred to Tank 18. Approximately 100 gallons of insoluble solids could not be removed, and were left on the three stiffener bands (bands that protrude four to five inches forming a ledge around the circumference of the tank). This heel removal campaign reduced the wet solids volume from approximately 33 kgal to approximately 15 kgal of residual waste. (Figure 2-18, Panel 6). [5],[43]

Figure 2-23: Tank 19 Heel Removal Rate [41]

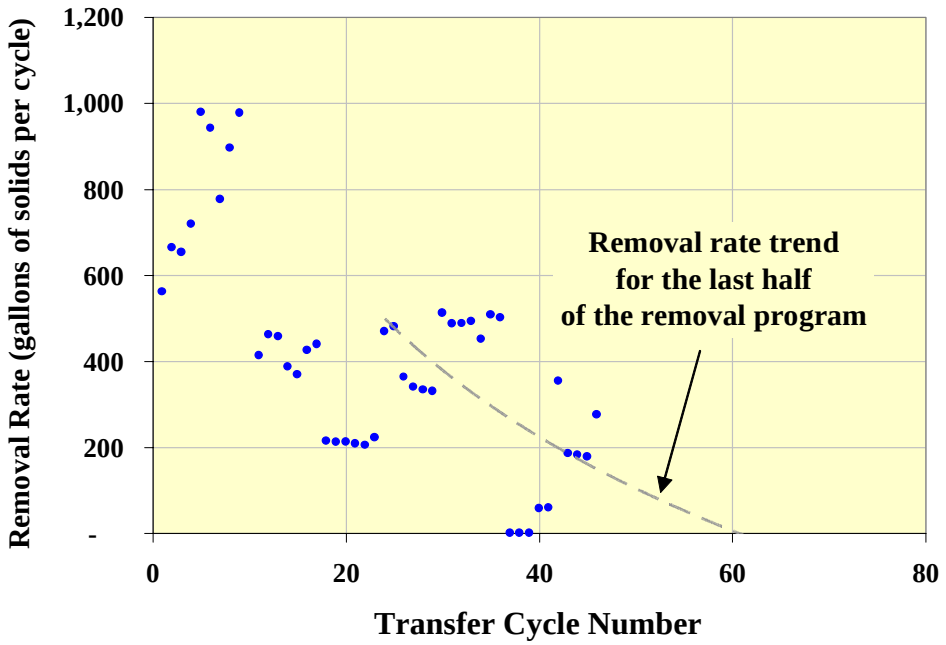
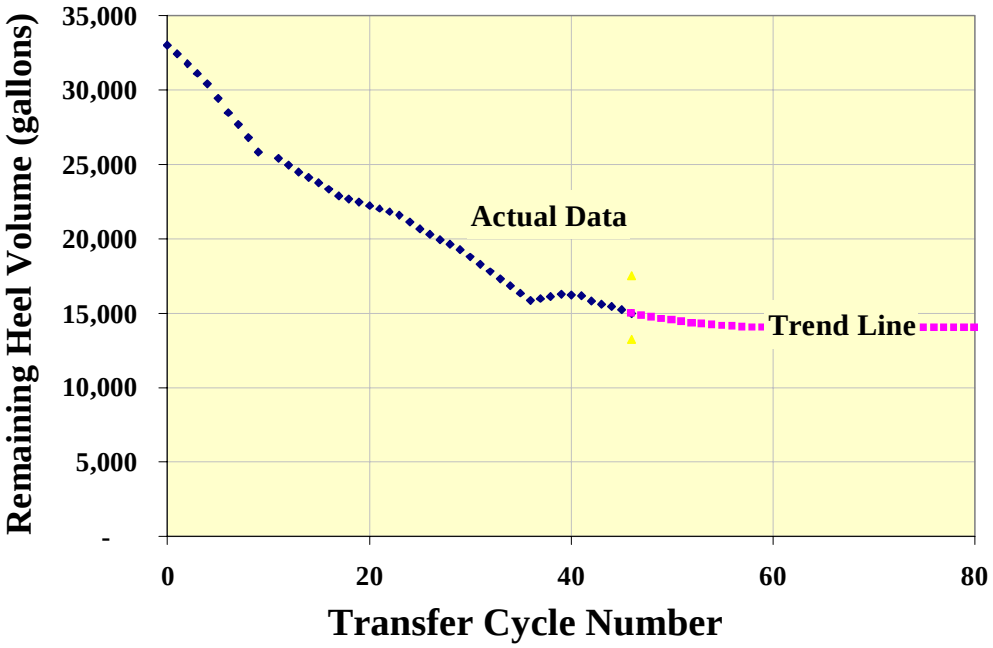
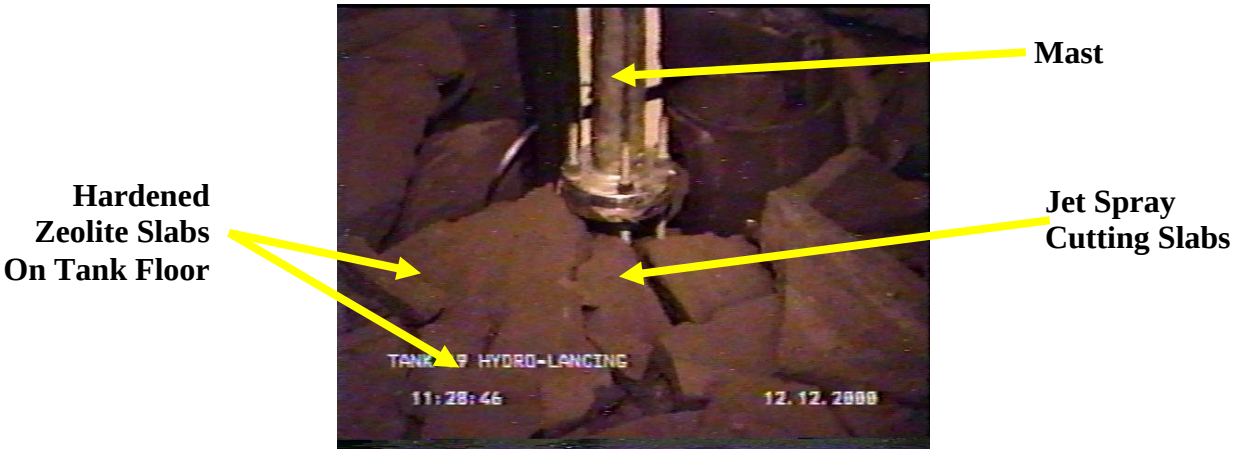


Figure 2-24: Tank 19 Remaining Heel Volume [41]



The principal difficulty of the Tank 19 heel removal efforts was the characteristics of zeolite. Zeolite has proven to be resistant to removal using standard mixer pumps because of its significantly higher settling velocity when compared to that of the metal oxides and insoluble salts (i.e., traditional sludge waste). Zeolite is also resistant to dissolving with acids. [30] A tool was designed to assist the mixer pumps in breaking up the hardened zeolite mound. The tool, called a hydro-lance, was designed to apply a 10,000 psi, 32 gpm water stream directly to the mound (Figure 2-25). Two separate hydro-lance campaigns were executed in Tank 19 using approximately 3 kgal of IW to successfully breakup the zeolite mounds**.** These hydro-lance campaigns cleared the way for the transfer pump mast to be fully seated at the bottom of the tank, thus allowing the transfer pump to pump from a lower level in the tank, cleared the transfer pump suction of zeolite "chunk" debris, and enabled the zeolite mass to be reduced for better mixing and washing capabilities. This process was not possible prior to use of the hydro-lance. [40]

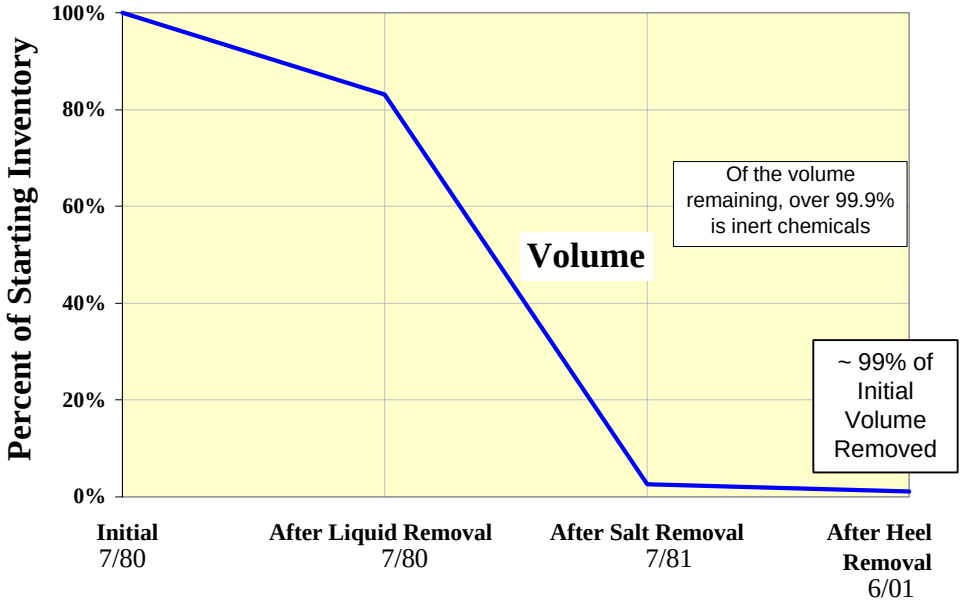
Figure 2-25: Hydro-Lance in Operation Breaking Up Mound



2.4.1.5 Tank 19 Waste Removal Summary

Figure 2-26 shows the volume in Tank 19 after each waste removal step. Of the starting waste volume in 1980 of 1,300 Kgal in Tank 19, as shown in Figure 2-17, only approximately 15 kgal of wet solids remains inside the tank. Of the remaining material in Tank 19, greater than 99.9% is inert chemicals and approximately two gallons are radioactive isotopes. [44]

Figure 2-26: Tank 19 Waste Removal Effectiveness – Volume [31],[33]

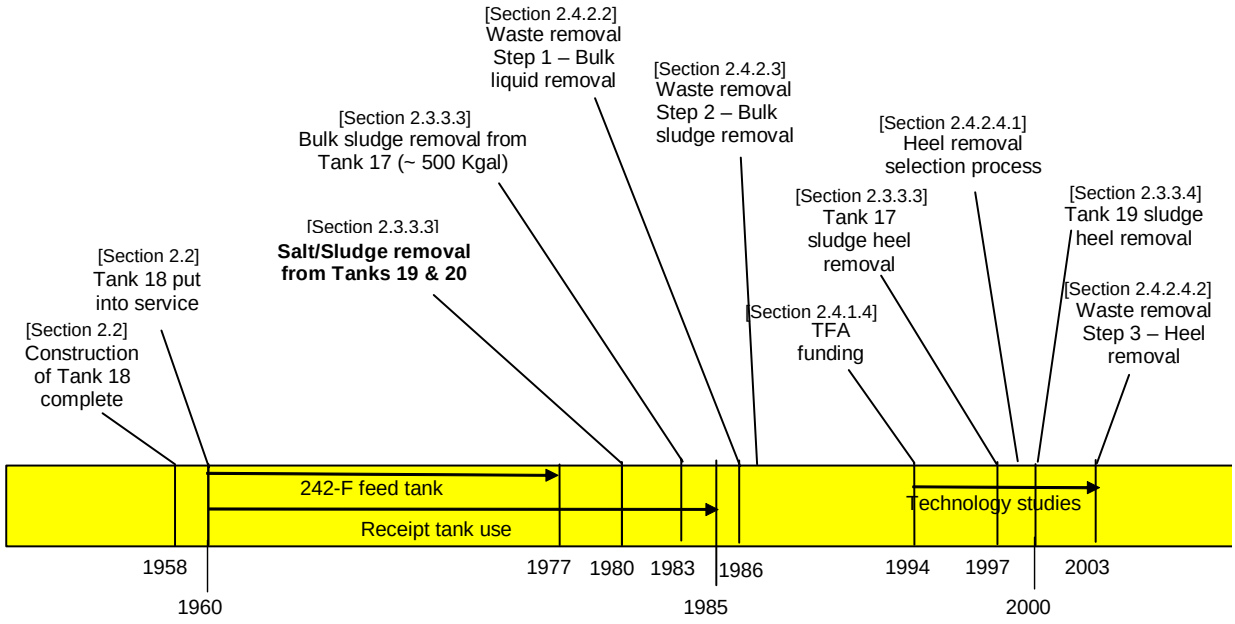


As discussed in Section 2.5.1, the two-phase, three-step waste removal process illustrated in Figure 2-18, removed approximately 96% of the curie inventory and approximately 99% of the volume in Tank 19 at the start of waste removal. Figure 2-18 shows the dates of the phases and steps in waste removal. Approximately 99% of the remaining curies are associated with the short-lived Cs-137/Ba-137m chemically bound to the zeolite.

2.4.2 Tank 18 Waste Removal

Beginning in January 1986, DOE began a two-phase, three-step plan to remove waste from Tank 18. Phase One of the plan, bulk waste removal, comprised steps one and two, removal of the concentrated liquid waste and then removal of the bulk sludge. Phase Two of the plan comprised step three, heel removal. See Figure 2-27 for the Tank 18 historical timeline which includes waste removal activities.

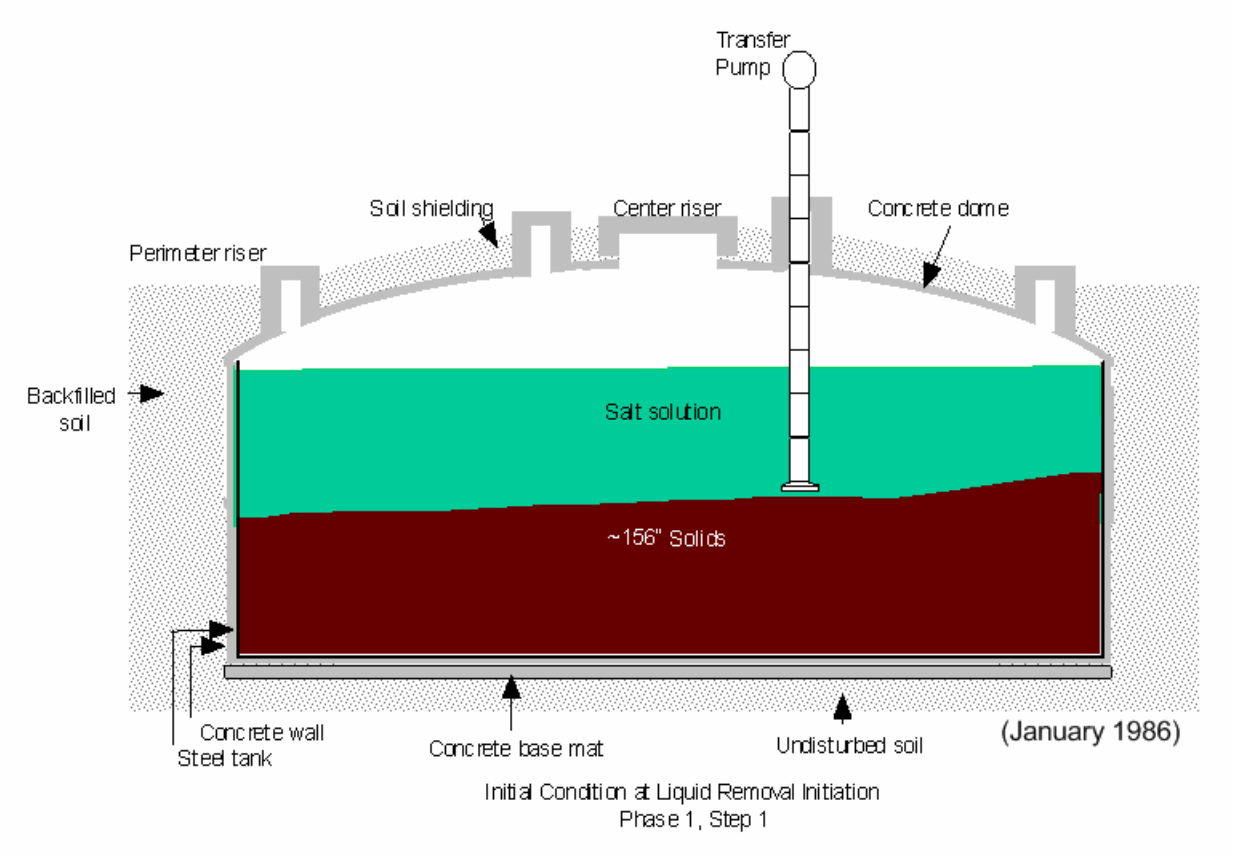
Figure 2-27: Tank 18 Historical Timeline



2.4.2.1 Initial Condition Prior to Waste Removal

The Tank 18 contents prior to beginning the waste removal campaign were approximately 1,300 kgal of total waste. Approximately 551 kgal of this waste was in a wet solids form (comprised of sludge and insoluble solids) with their associated interstitial liquid and approximately 750 kgal of freestanding liquid (salt solution) (Figure 2-28). [45]

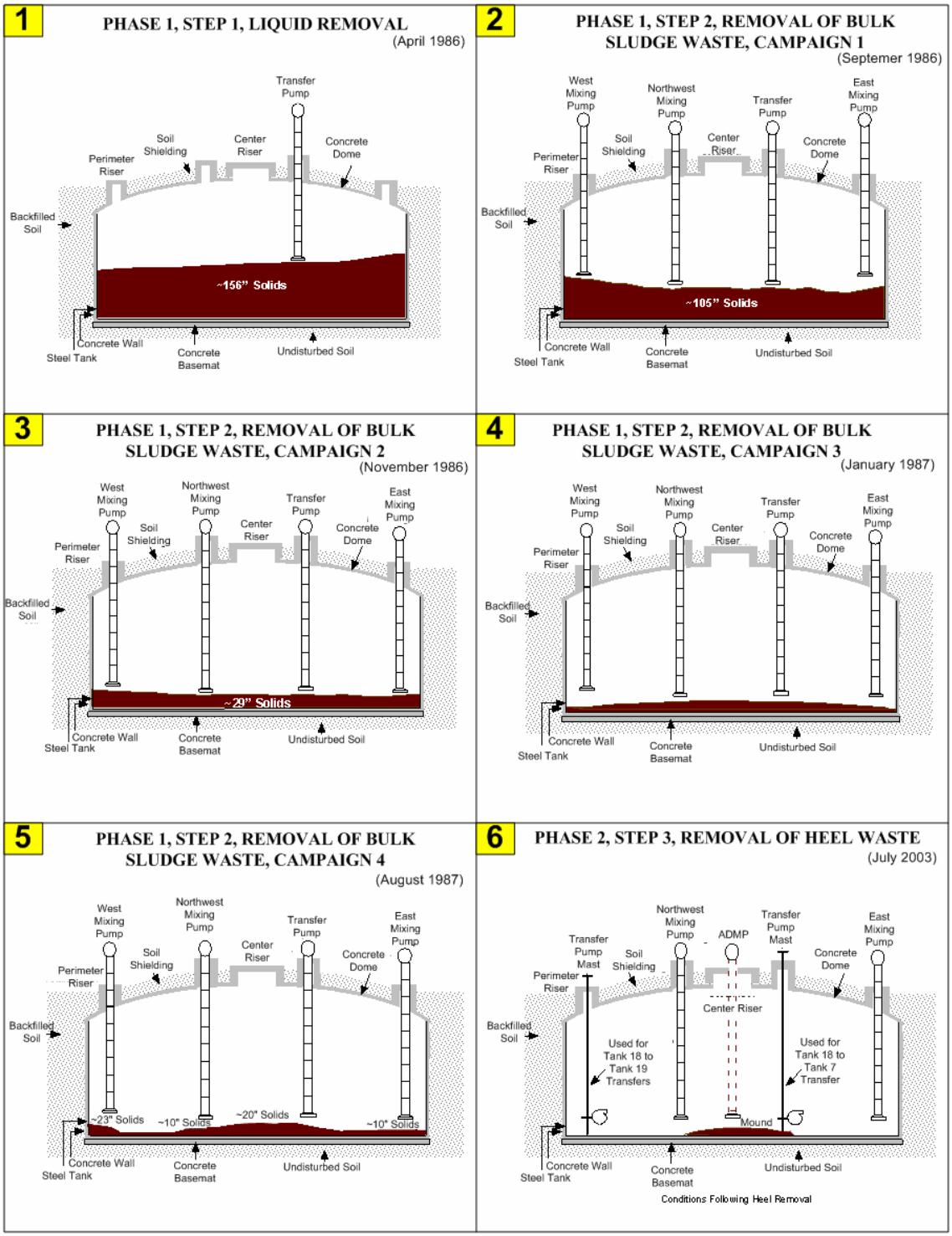
Figure 2-28: Tank 18 Initial Condition [45]



2.4.2.2 <u>Phase 1, Step 1- Removal of Liquid Waste</u>

During this step, the concentrated liquid waste both free standing (liquid above the sludge) and some interstitial (liquid in the sludge) was removed from January 1986 to April 1986. No slurry pump operations were conducted during this phase and a telescoping transfer pump (TTP) was used as the prime mover of this waste stream. A total of 850 kgal (some of which was interstitial) were moved and the liquid waste stream was transferred to Tank 26 for future processing/storage in FTF. This liquid waste stream was redistributed to other tanks in FTF and then evaporated again to further reduce the volume and transfer the saltcake to other tanks. After the liquid waste was removed an estimated volume of approximately 551 kgal of sludge remained (Figure 2-29, Panel 1). [45]

Figure 2-29: Tank 18 Sludge Removal Campaign (April 1986 – July 2003) [45]



The numbers related to saltcake height represent approximate average height, except for panels 5 and 6, where the numbers represent maximum height.

2.4.2.3 Phase 1, Step 2- Removal of Bulk Sludge Waste [45]

During Step 2, SRS installed three mixer pumps (Figure 2-19) which required installation of new infrastructure (steel trusses), to prepare Tank 18 for sludge removal. The three mixer pumps were mounted on the new supporting truss work. Each mixer pump had a 150-hp motor and ran at 2,200 revolutions per minute. Each pump had a 45-foot shaft with cylindrical support columns that were filled with clean water and pressurized to prevent cross contamination between the waste and the top part of the tank. IW was added to the tank, and the mixer pumps exerted a sweeping liquid jet action on the sludge to promote its mixing and allow the particles to be suspended for transferring. In addition to the new mixer pumps, service upgrades were performed on the Tank 18 heating and ventilation system, IW addition system, and BW support system. A TTP was used to move the sludge slurry solution out of Tank 18 into the tank farm system. This sludge has already been processed through the DWPF and resides in canisters (stored in the Glass Waste Storage Building) at DWPF, awaiting approval for shipment to the Federal Depository located in Yucca Mountain, Nevada when it has been licensed approved for operation. [10]

To begin the phase 1, step 2 process, both radioactive salt solution and non-radioactive IW were added to the tank to improve hydraulic properties and optimize mixing and washing to allow transferring of the sludge slurry. From July 1986 to August 1987, seventeen different sludge slurry transfers combined into four campaigns were executed (Table 2-2). The mixer pumps were lowered periodically during the transfers. The zeolite was not yet present during this step of waste removal because the Tank 19 material had not been transferred into Tank 18 at that time. [45]

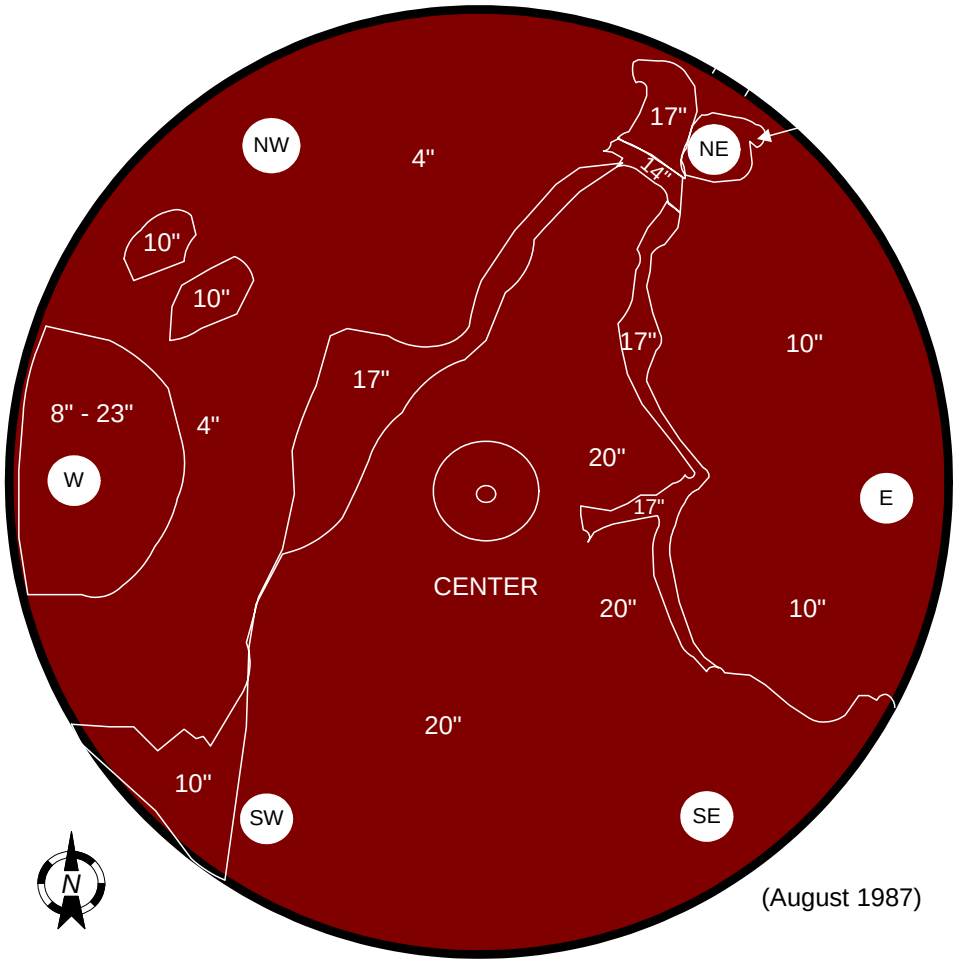
Table 2-2: Tank 18 Sludge Removal Campaigns [45]

Sludge Slurry Transfers (grouped in 4 campaigns)	IW/Salt Solution Added (kgals)	Sludge Removed (kgals)	Sludge Slurry Removed (kgals)	Figure 2-29
1st (transfers 1-5) (9/86)	1,300	180	1,300	Panel 2
2nd (transfers 6-10) (11/86)	400	270	700	Panel 3
3rd (transfers 11-15) (1/87)	290	60	420	Panel 4
4th (transfers 16-17) (8/87)	200	5	220	Panel 5
Overall Results	2,190	515	2,640	

During the four sludge slurry removal campaigns, approximately 515 kgal of sludge were slurried and removed using about 2,190 kgal of IW/salt solution. [45] The final sludge slurry transfer was completed August 1987 (Figure 2-29, Panel 5). Following the final sludge slurry transfer, approximately 37 kgal of solids remained in Tank 18. An additional 12.5 kgal of insoluble solids were later transferred into Tank 18 from the heel removal operations from Tank

17 and Tank 19. This included approximately 2.5 kgal of zeolite that was transferred into Tank 18 from Tank 19. The final sludge mapping indicated that several mounds existed but the majority of the material was in a connected formation along the centerline of the tank (Figure 2-30). The shape of these mounds and their locations was attributed to the location of the three slurry pumps in the northeast, northwest and southeast risers. The highest mound was approximately 23 inches tall. This residual material was known as the heel that will be addressed in Phase 2, step 3. [32]

Figure 2-30: Heel Map Following Bulk Sludge Removal [32]



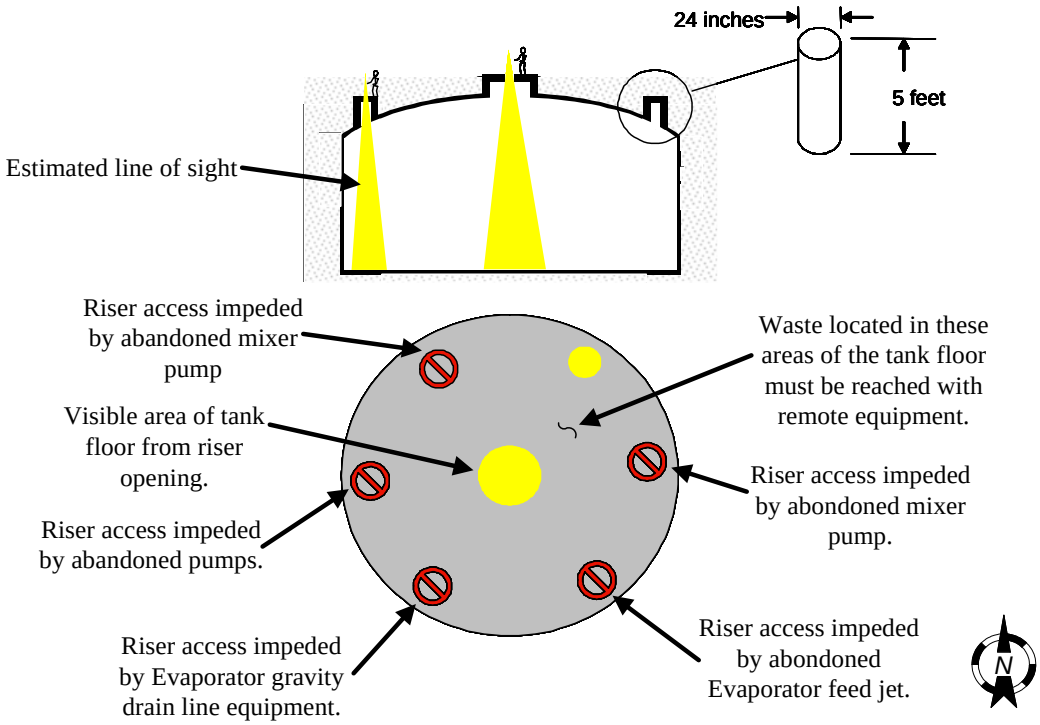
2.4.2.4 Phase 2, Step 3- Removal of Heel Waste

After Phase 1, Tank 18 remained inactive (except for routine surveillance and maintenance), until June 2000. Requirements and factors impacting selection of heel reduction technology included physical configuration of the tank, fast settling zeolite, FFA regulatory commitments, and limited storage space in the tank farm system.

The same physical configuration factors impacting heel removal technology selection for Tank 19 impacted the technology selection process for Tank 18. In addition to the tank's large size and floor design, the configuration of tank risers (access openings) provides very limited access to the tank interior. The tank has one center riser and six perimeter risers. Each perimeter riser has a nominal diameter of 24 inches. In Tank 18, an abandoned transfer pump, two abandoned mixer pumps, an evaporator feed jet and gravity drain line, are installed in five of the six perimeter risers. Therefore, these five risers were undesirable locations for heel removal equipment, as change out of equipment in these risers would result in significant occupational radiation exposure and the generation of additional radiological waste. Riser configuration above the tank top limits direct equipment access and allows only a restricted view of the tank floor (Figure 2-31).

As discussed in Section 2.3.3, an estimated 13 kgal of zeolite were discharged into Tank 19, and approximately 2.5 kgal of that zeolite were transferred into Tank 18 during waste removal activities from Tank 19. [29] The other factors impacting heel removal technology selection, FFA commitments and available waste tank space, discussed in Section 2.1.1, are also applicable to Tank 18.

Figure 2-31: Diagram of Tank 18 Access Area for Heel Removal Equipment [19],[20],[21],[22],[23]



2.4.2.4.1 Technologies Selection Process and Testing for Tank 18 Heel Removal

2.4.2.4.1.1 ***Selection Process***

A similar selection process that was used for the selection of the heel removal technology for Tank 19, described in Section 2.4.1.4, was also used to select a removal technology for the heel in Tank 18. [46] TFA studies conducted since the issuance of the Tank 19 SEE were included in the Tank 18 selection process. This selection process, documented in the SEE, concluded that the ADMP (Figure 2-32) was the preferred technology for Tank 18 heel removal. [46] The center-mounted ADMP was selected based on the limited access to the tank risers (the center riser was accessible) and previous testing of the ADMP via the TFA effort. As in the case for Tank 19, the selection process rejected chemical cleaning as a heel removal option because of concerns about down-stream process impacts (system integration) and safety authorization basis impacts. [46]

Figure 2-32: Advanced Design Mixer Pump (ADMP)

View of ADMP showing discharge nozzle.

View of ADMP showing suction screen (39 inch diameter).

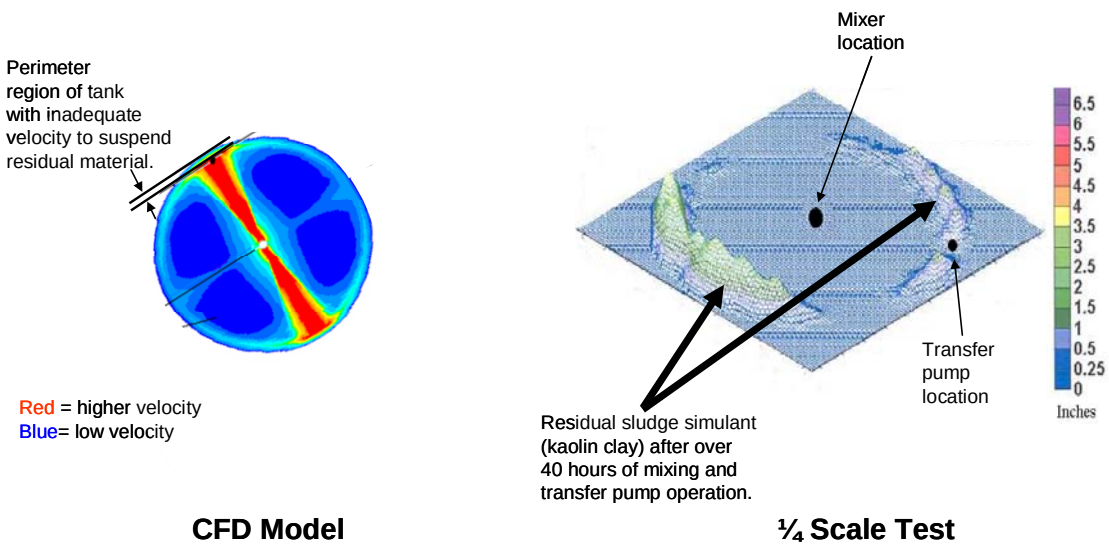
Installation of the ADMP in the center riser of Tank 18

2.4.2.4.1.2 ***Testing***

After the SEE was concluded, the ADMP was purchased and functionally tested at SRS. The ADMP successfully demonstrated its effectiveness with more than 4,200 hours of testing with kaolin clay that was significantly more viscous than the expected Tank 18 residual material. The ADMP also has a greater Effective Cleaning Radius (ECR) than the slurry pumps typically used for waste removal and would require only one tank entry versus three that are associated with standard waste removal technology (three slurry pumps). [47]

Computational fluid dynamics (CFD) modeling was also performed at the SRNL using kaolin clay as a simulant to the heel material (Figure 2-33). The modeling predicted that the ADMP was capable of suspending sludge particles to the waste tank wall with the ADMP placed in the center riser and operating at full speed (1,185 rpm). [48]

Figure 2-33: Modeling and Test Results for Tank 18 [48]



2.4.2.4.2 Heel Removal Operation and Results

The ADMP pump was placed in the center riser of the tank and attached to structural steel above the tank top in September 2002. The pump had a 55-foot vertical shaft that enabled the suction/discharge nozzles to reach the tank bottom. The pump had two discharge nozzles rated at 5.2 Kgal per minute each for a total of 10.4 Kgal per minute. During heel removal operations, the ADMP mixer, one centrifugal transfer pump, a dewatering pump and a transfer system completed over 1,000 hours of mixing and six transfers of material out of Tank 18 into Tank 7. [32]

The heel removal strategy consisted of adding well water (total of 800 Kgal), mixing with the ADMP, and transferring the sludge slurry to Tank 7. The ADMP was turned off when the tank level reached a level specified to prevent aerosolization of waste, a radiological release hazard. This sequence of adding well water, mixing and transferring, continued for six transfers until a point of diminishing return was reached, i.e., waste was no longer being effectively transferred. Figures 2-34 and 2-35 indicate the amount of heel removal and the remaining heel volume, respectively, as a function of the number of transfer cycles. [49] The amount of solids removed from Tank 18 during transfers was compared with the time of ADMP operation, as shown in Figure 2-36. Comparing the gallons removed per cycle with the time of mixing per cycle indicates that solids removal per hour of mixing decreased significantly in the final three cleaning cycles. [50] The 800 Kgal of well water added to the tank farm system during this heel removal step was transferred to Tank 51 in H-Area to wash sludge destined for disposal as vitrified glass via the DWPF. [32]

As a part of the heel removal process, the walls of Tank 18 were washed with water from a nozzle in the center riser. [32] The spray nozzle had pan and tilt capability, and an operating capacity of 65 gpm at 175 psig. This spray wash system was used for washing of the tank walls from top to bottom and for focused washing in areas with deposits including the stiffener bands (bands that protrude four to five inches forming a ledge around the circumference of the tank). Approximately 2.8 Kgal of water were used. Approximately 14 gallons of insoluble solids were left on the stiffener bands. The Phase 2, step 3 heel removal campaign reduced the wet solids volume in Tank 18 from approximately 49 Kgal to approximately 4.3 Kgal. [6]

Figure 2-34: Tank 18 Heel Removal Rate [49]

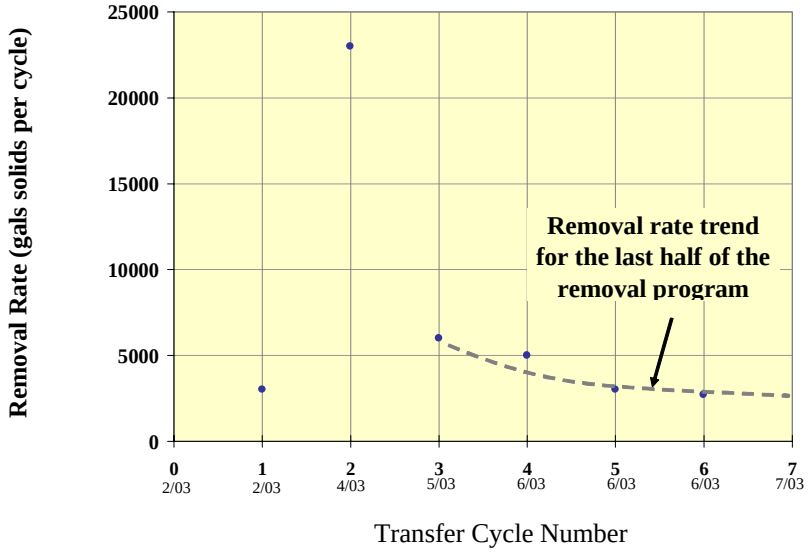


Figure 2-35: Tank 18 Remaining Heel Volume [49]

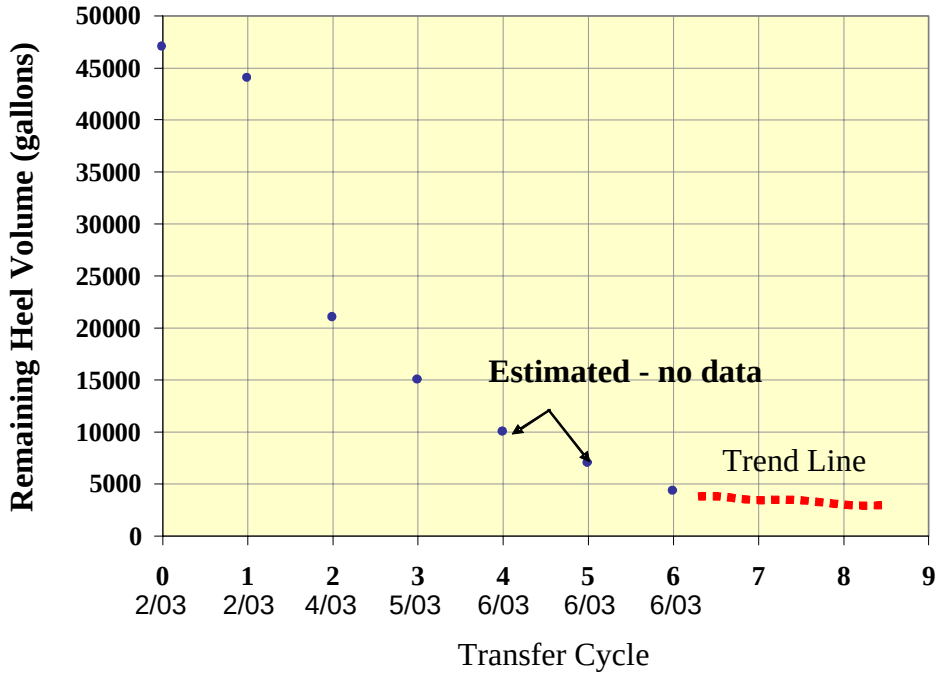
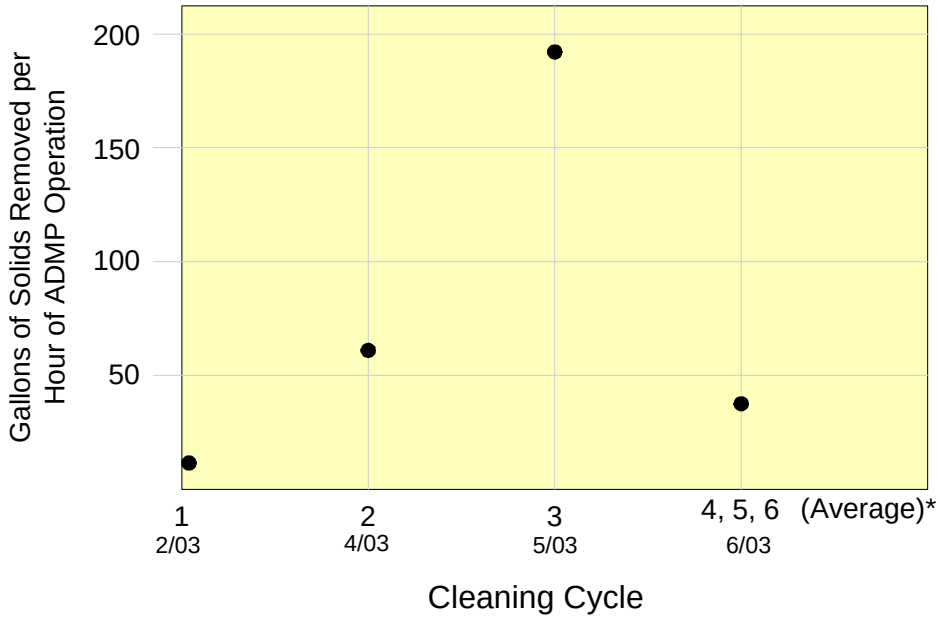


Figure 2-36: Tank 18 Heel Removal-Mixing Rate [50]



* The actual volumes of solids removed for cleaning cycles 4 and 5 were not calculated; however, the total volume of solids removed for cleaning cycles 4, 5, and 6 combined is known. Therefore, the average volume of solids removed per hour of ADMP operation for cleaning cycles 4, 5, and 6 is shown.

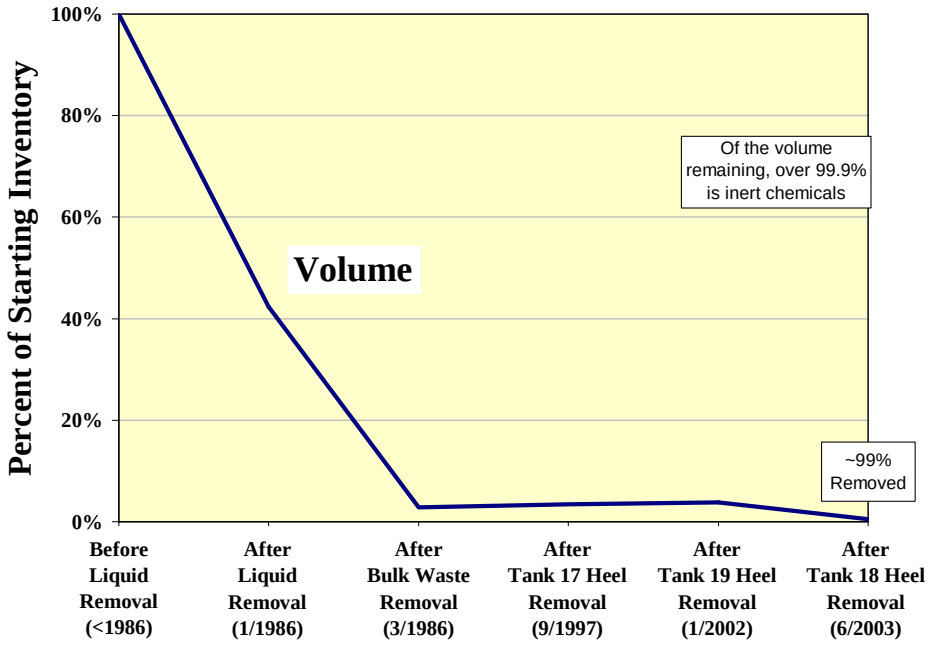
The principal difficulty of Tank 18 heel removal efforts was the removal of a hardened mound near the southwest riser. The southwest mound was a result of the bulk removal campaigns (Phase 1, steps 1 and 2). The location of the three slurry pumps and their ECR had left the southwest area of the tank vulnerable to "difficult to remove" hardened sludge. As previously stated, the southwest risers were considered inaccessible for waste removal efforts (because of the contaminated "pill boxes"), and therefore no slurry pumps had been installed at these locations (Figure 2-27). There was also a smaller mound remaining in the north region of the tank. In addition to the unexpected difficulty to remove the mounds, the flow characteristics of the Tank 18 residual material is believed to have impacted the ADMP's ECR and caused the pump's performance to be less than predicted. [32]

In summary, a number of obstacles prevented further residual waste removal during this last phase. The obstacles fell into four categories: (1) fast-settling zeolite resins, (2) general tank access limitations and obstructions, (3) hardened mounds of insolubles, and (4) unexpected material flow characteristics that adversely impacted ADMP performance.

2.4.2.5 Tank 18 Waste Removal Summary

Figure 2-37 shows the volume in Tank 18 after each waste removal step. Of the starting waste volume of 1,300 Kgal in Tank 18 only approximately 4.3 Kgal remains inside the tank. Of the remaining material in Tank 18, greater than 99.8% is inert chemicals and approximately 10 gallons are radioactive isotopes. [51]

Figure 2-37: Tank 18 Waste Removal Effectiveness – Sludge Volume [33]



As discussed in Section 2.5.2, the two-phase, three-step waste removal process removed approximately 99% of the curie inventory in Tank 18 at the start of waste removal. Approximately 86% of the remaining curies are associated with the short-lived Cs-137/Ba-137m bound to the zeolite.

2.5 Radionuclide Inventory

2.5.1 Tank 19 Residual Waste Characterization Process

Tank 19 residual waste characterization was a three-step process that followed heel removal. Step 1 consisted of sampling and analysis to determine the presence and concentration of radionuclides. Step 2 consisted of mapping and volume estimation to determine the amount of residual material. Step 3 consisted of characterizing the residual material based on analytical results, process knowledge, special analysis, and the amount of residual material. [5]

2.5.1.1 Step 1 – Determine Radionuclide Concentration

After the completion of heel removal, a plan for sampling the residual material in Tank 19 was developed. [52] A statistical study was performed by the Statistical Consulting Section of SRNL to aid in the planning of the sampling process. The statistical study supported the development of an appropriate sampling methodology in terms of the number of samples taken, locations of samples, and statistical approach for characterization as discussed in Section 2.5.1.3. This analysis demonstrated that fewer samples were needed for adequate characterization because the residual material was well-mixed following the 3,000-hour mixing campaign. Analytical results for the samples were similar providing indication that the solids were well-mixed. [53] After the final mixing/transfer cycle, the residual solids that remained in Tank 19 appeared visually similar. Furthermore, visual inspections during the heel removal process indicated that residual solids had been significantly distributed within the tank. [43]

Samples were obtained from Tank 19 using custom-designed tools (Figure 2-38) specifically developed to extract the solids samples from various locations in accordance with the sample plan. [52] Three solids samples were taken from different locations and various residual material heights in the tank. A solids sample (FTF-077) was taken through the southwest riser of the tank, and a solids sample (FTF-075) was taken from the center riser of the tank. The third solids sample (FTF-118) was taken from the northwest region of the tank through a new opening cut into the tank top. The new opening was needed to facilitate multi-level sampling of the northwest residual solids mound as required by the sample plan. The purpose of this sample was to characterize material underneath the surface of the mound. [52] In addition, one liquid sample was obtained. The single liquid sample was representative because of the 3,000-hour mixing campaign. Following waste removal, the liquid level in Tank 19 was very low. The liquid sample (representing the residual liquid in Tank 19) was obtained from Tank 18 because the liquid in Tank 18 was directly received from Tank 19.

Figure 2-38: Tank 19 Sample Tools [54]

Sample tool used to obtain samples FTF-075 and FTF-077

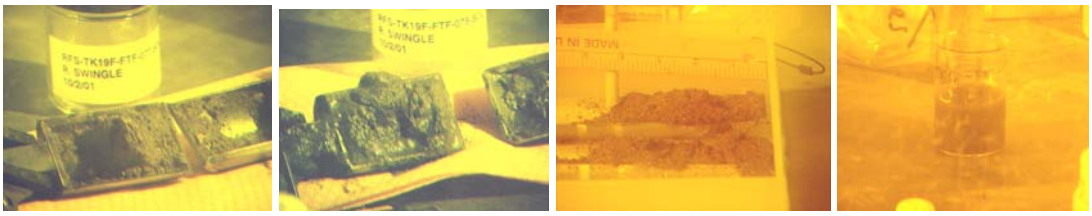
Sample Container
(2.5 inches long)

Sample Container
(8 inches long)

Sample tool used to obtain multilevel sample FTF-118

The three solids samples appeared similar in color and texture (Figure 2-39) providing additional evidence that the residual material was well mixed throughout Tank 19. [54]

Figure 2-39: Samples in the Laboratory [54]



FTF-077 FTF-075 FTF-118 Liquid Sample (from Tank 18 representing Tank 19 residual liquid)

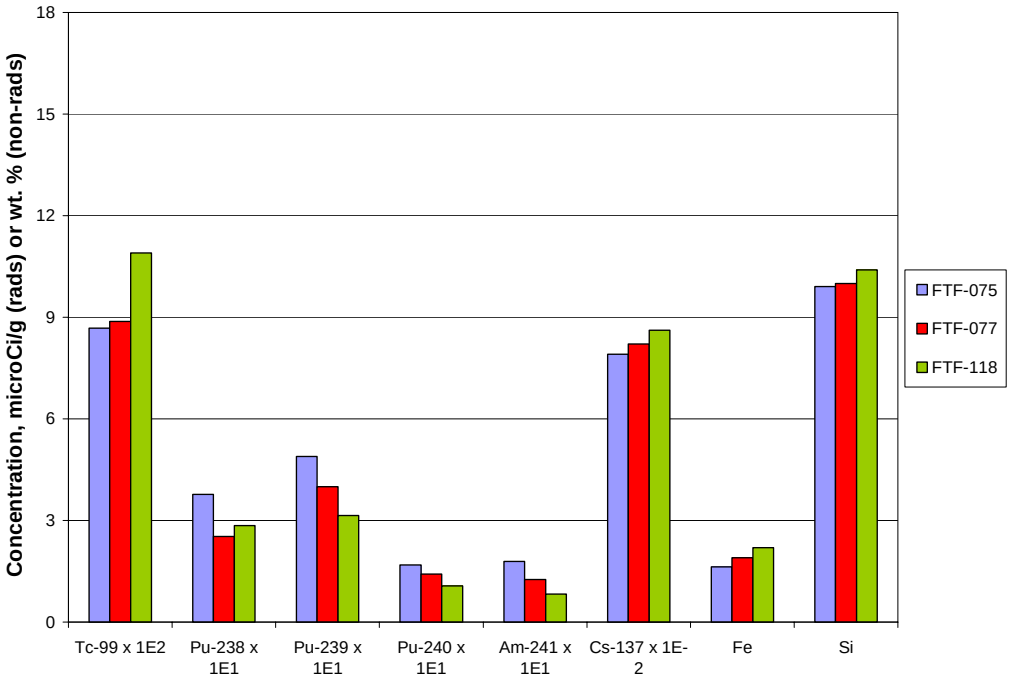
The samples were analyzed in the laboratory by SRNL technicians for radiological constituents important to the tank closure process (analytical techniques are further discussed in Section 2.5.1.3). Radiological constituents were selected for analysis based on their importance to the environmental modeling for the performance evaluation and 10 CFR 61.55 waste classification

determination. The total Tank 19 radionuclide inventory was estimated based on a combination of analytical data, process history, and special analysis. [55]

The analytical results from the laboratory analysis indicated that the radionuclide concentrations of the three solids samples were also similar, as shown in Figure 2-40. [54] The similarity of the three solids samples radionuclide concentrations further indicated that the solids were well mixed (Figure 2-40).

Obtaining, packaging, shipping and analysis of each sample resulted in personnel radiation exposure. Tank 19 sample dose rates were as high as 8,000 mrem per hour (mrem/hr) dose to the hands and 4,000 mrem/hr dose to the whole body. The total whole body worker dose received from the three solid samples was approximately 200 mrem. [56],[57],[58]

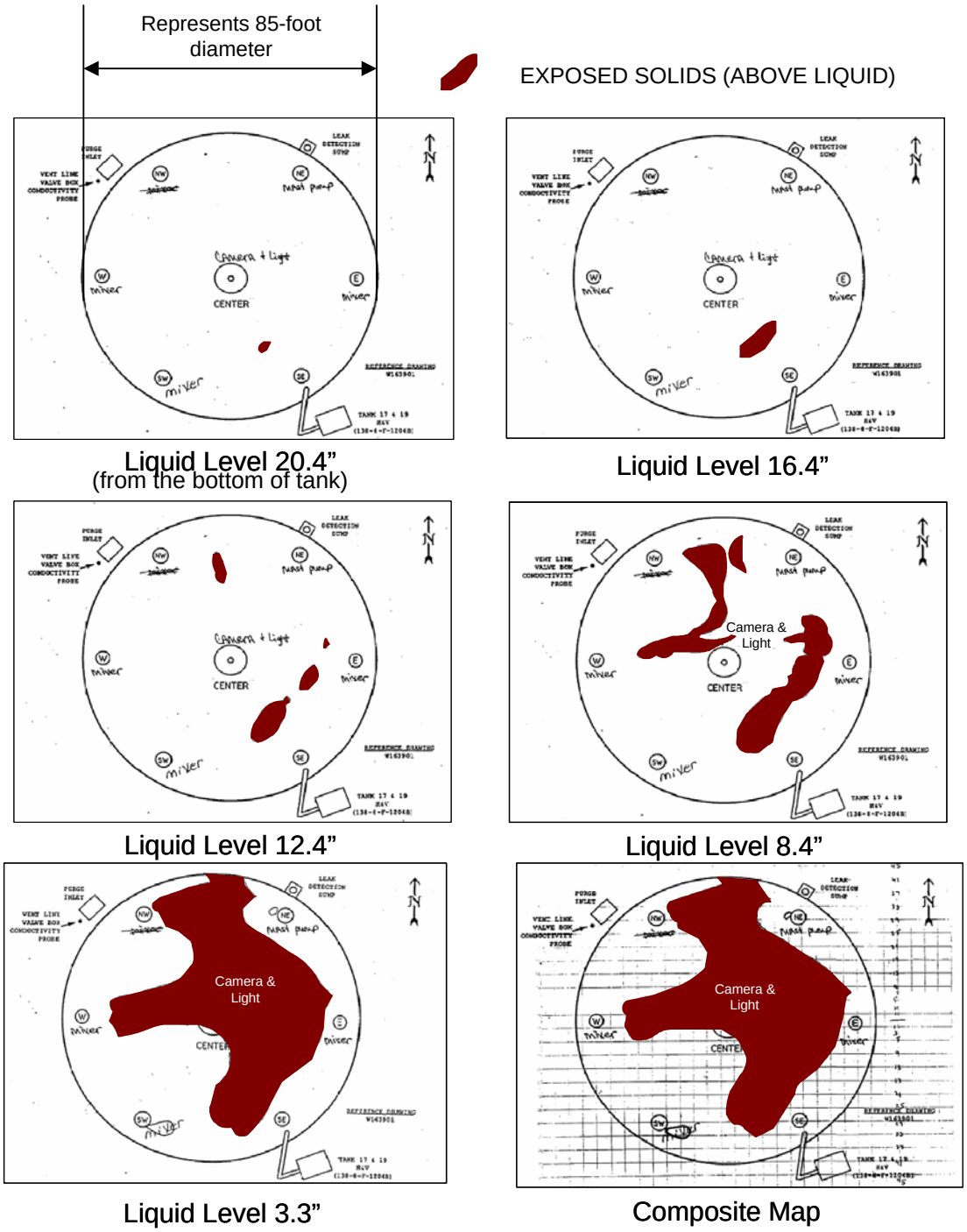
Figure 2-40: Similarity of Tank 19 Samples



2.5.1.2 Step 2 - Determine the Amount of Residual Material

During transfers from Tank 19 to Tank 18 near the end of the heel removal campaign, exposed solids were mapped corresponding to liquid levels based on liquid level instrumentation (reel tape) readings (Figure 2-41). Detailed camera inspections of the tank contents used various landmarks within the tank to determine the locations of the exposed solids. Exposed solids were mapped and residual volume was calculated a total of 14 times throughout the Tank 19 heel removal campaign. [43] The experience gained from these previous efforts provided the opportunity to continuously improve the process and minimize uncertainty in the final mapping and volume calculation. Additionally, camera technicians trained to perform tank inspections assisted in the mapping of exposed solids.

Figure 2-41: Tank 19 Exposed Solids Maps [43]



Liquid Level 20.4" (from the bottom of tank)

Liquid Level 16.4"

Liquid Level 12.4"

Liquid Level 8.4"

Liquid Level 3.3"

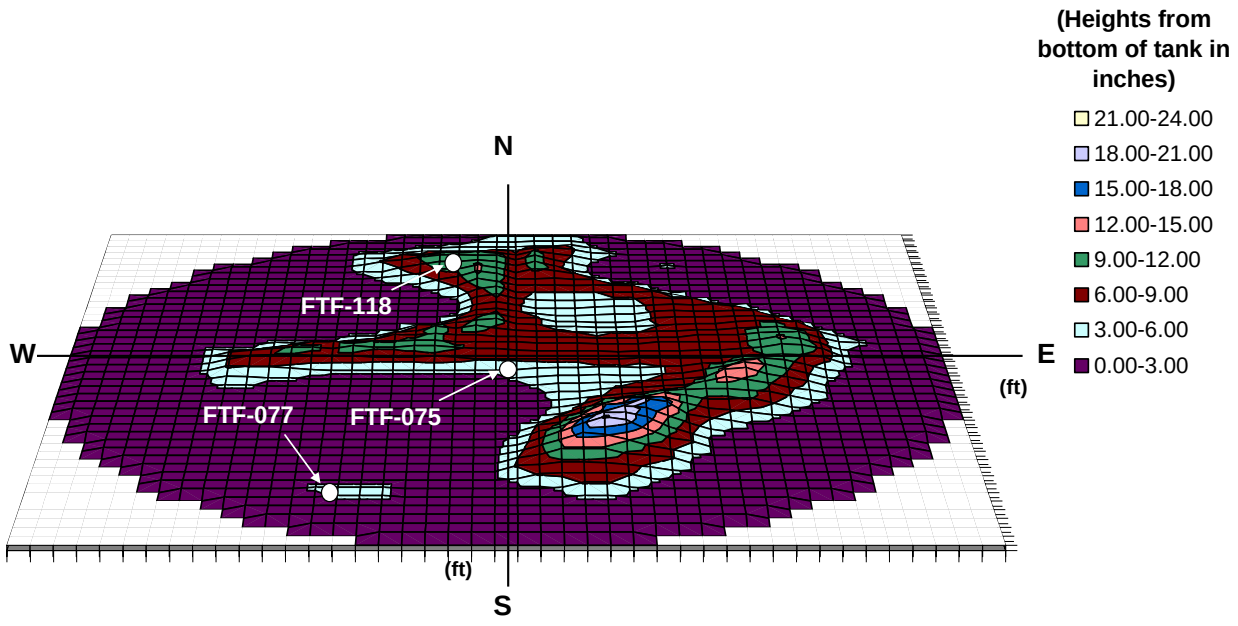
Composite Map

Solids maps at multiple liquid levels combined to produce a contour map of the residual solids (see Figure 2-41). The contour map was used with a tank grid as seen in Figure 2-42 to plot the height of residual material. The appropriate height of residual solids and liquid was assigned to the corresponding grid location to determine the residual solids volume and free liquid volume. [43]

Tank 19 and Tank 18 have three bands of steel angle around the inside circumference of the tank wall designed to "stiffen" and provide support to the steel tank liner. The top angle protrudes four inches from the tank wall, while the bottom two angles extend five inches from the tank wall. For Tank 19, historical photographs show piles of solids built up on these stiffening bands. Spray washing dislodged residual waste off the stiffening bands leaving approximately 100 gallons of solids on the bands in Tank 19 based on visual camera inspection. [5]

The residual material volume totals 16.9 kgal. Wet solids (including interstitial liquid) comprises 15.1 kgal, and the free liquid volume is 1.8 kgal. This represents an equivalent depth of wet solids of approximately four inches in the 34-foot high tank. [4] The configuration of the residual material is depicted in Figure 2-42. This volume includes the equivalent of approximately two gallons of radioactive constituents (<< 1% of the total volume of radioactive constituents originally in Tank 19). [44] The residual material volume consists of approximately 10 kgal of zeolite, and approximately 5 kgal of insoluble solids. [5],[29]

Figure 2-42: Tank 19 Residual Material Configuration and Sample Locations [5]



Note: Each square in the grid represents a 2 foot x 2 foot area of the tank bottom

Approximately 96% of the original curies in the tank at the start of removal have been removed. [59] Approximately 99% of the remaining curies are associated with Cs-137/Ba-137m bound to the zeolite (i.e., the cesium/barium atoms captured by the zeolite resin are chemically part of the zeolite molecule).

2.5.1.3 Step 3 - Characterize the Residual Material

The radioactive inventory of each radionuclide in the residual waste was determined by one of three means: (1) sample laboratory analysis, (2) special analysis, and (3) predictions based on the knowledge of the wastes that had been sent to the tank (process knowledge). Sample laboratory analysis was the preferred method for determining the radionuclide inventory. Table 2-3 shows the means by which each radionuclide inventory was determined.

Table 2-3: Tank 19 Radionuclide Inventory [5],[55]

Radionuclide	Tank 19 Total Inventory (Ci)	Percent of Total Curies (%)
H-3	3.78E-01	*
C-14	1.12E-01	*
Co-60	7.28E+00	*
Ni-59	8.26E-01	*
Ni-63	7.62E+01	0.08%
Se-79	5.89E-02	*
Sr-90	4.00E+01	0.042%
Y-90	4.00E+01	0.042%
Nb-94	3.63E-05	*
Tc-99	6.76E+00	*
Ru-106	2.70E-04	*
Rh-106	2.70E-04	*
Sn-126	1.27E-01	*
Sb-125	2.59E+00	*
Sb-126	1.78E-02	*
Sb-126m	1.27E-01	*
Te-125m	6.32E-01	*
I-129	4.95E-05	*
Cs-134	7.60E-03	*
Cs-135	7.91E-04	*
Cs-137	4.92E+04	51.27%
Ba-137m	4.64E+04	48.35%
Ce-144	3.71E-06	*
Pr-144	3.71E-06	*
Pm-147	4.15E+01	0.04%
Sm-151	1.30E+00	*
Eu-152	5.62E-03	*
Eu-154	1.47E+01	0.02%
Eu-155	7.57E-02	*
Ra-226	1.06E-07	*

Radionuclide	Tank 19 Total Inventory (Ci)	Percent of Total Curies (%)
Ra-228	0.00E+00	0
Ac-227	2.85E-07	*
Th-229	2.61E-03	*
Th-230	1.29E-05	*
Th-232	0.00E+00	0
Pa-231	7.90E-07	*
U-232	2.10E-04	*
U-233	9.83E-01	*
U-234	8.96E-02	*
U-235	1.36E-03	*
U-236	3.27E-03	*
U-238	4.54E-02	*
Np-237	1.12E-02	*
Pu-238	2.45E+01	0.03%
Pu-239	3.12E+01	0.03%
Pu-240	1.11E+01	0.01%
Pu-241	5.49E+01	0.06%
Pu-242	3.72E-01	*
Pu-244	1.66E-03	*
Am-241	1.17E+01	0.01%
Am-242m	1.93E+00	*
Am-243	1.75E-07	*
Cm-242	9.07E-22	*
Cm-243	2.70E-06	*
Cm-244	3.05E-03	*
Cm-245	1.97E-09	*
Cm-247	6.16E-20	*
Cm-248	1.42E-20	*
Bk-249	5.36E-30	*
Cf-249	3.96E-22	*

*	<0.01% of Total Curies
	Sample Analysis
	WCS
	Special Calculation

Total Curies	9.6E+04

For radionuclides that were not analyzed in the laboratory, the WCS was used to determine radionuclide inventory based on process knowledge. See Appendix D for a description of WCS. [55]

Special analyses calculations were performed to determine radionuclide inventories that had not been historically tracked in WCS. Examples of special analyses include fission yield ratios to other radionuclides and equations for progeny radionuclides in terms of the appropriate parent radionuclide. For example, the inventory of Sm-151 is based on fission yield activity ratio to the Sr-90 solids sample inventory, and the inventory of Th-229, a daughter of U-233, is based on the solids sample inventory of U-233. [55]

The samples were analyzed in the laboratory by SRNL technicians for radiological constituents important to the tank closure process. [54] Portions of the samples were dried and digested by two different methods. Each of these digestions was completed in triplicate aliquots (three portions) for analysis. Therefore, each sample underwent six analyses. Confidence in the radioanalytical data is high, based on the following attributes. The radioanalytical techniques were optimized for high level waste matrices. The analyses were performed using accepted quality assurance practices (standards, blanks, replicates, recovery determinations, etc.). All calibrations were based on National Institute of Standards and Technology (NIST) traceable standards or standards calibrated against NIST traceable standards. Multiple independent measurement techniques provided a basis for estimating data accuracy. For example, some radionuclides were analyzed independently by both mass spectroscopy and radiation counting methods. [54]

The statistical study provided the approach to produce the expected upper-bound uncertainty at the 95% confidence level. The 95% upper-confidence limit value of the average concentration of the six analyses was used for each analyzed radionuclide. (The true value is likely to be at or below the 95% upper confidence limit value used for the radionuclide inventory. By comparison, there is a 50% chance that the true value is higher (or lower) than the average value). The radionuclide inventory values in Table 2-3 were used in Sections 5, 6, and 7 of this document. The results of the characterization of the residual material on the tank floor are included in the radionuclide inventory report. [55]

In addition to the residual material on the tank floor, an analysis of the radionuclide inventory of corrosion products on the interior of the tank was performed. [60] Tank 19 and Tank 18 construction includes a carbon steel liner that is subject to corrosion (rusting) over time. A visual examination of video footage of the interior of Tank 19 concluded that some corrosion of the carbon steel walls had occurred. The surface of the tank appeared as expected for carbon steel exposed to either an alkaline aqueous environment or an ambient vapor space environment. Using general corrosion rates from laboratory and field test data and historical ultrasonic tank thickness measurements, SRNL estimated there are approximately 500 pounds of corrosion product (rust) on the Tank 19 walls. [61] Liquid constituents are in contact with corrosion products in the carbon steel tank. Some radionuclides partition from the liquid to the corrosion products. The amount of contamination on the tank interior surfaces was estimated using information on the liquid radionuclide concentration, the partition coefficient (K_d), and the mass of the corrosion products. SRNL estimated the corrosion products for Tank 19 represent approximately 200 curies (~0.2% of the total residual curies). For Tank 19, approximately 98%

of the corrosion product curies is from Cs-137. Radionuclides that have potentially been held on the corrosion products on the Tank 19 interior surfaces are included in the Tank 19 total radionuclide inventory reported in Table 2-3. [5]

2.5.2 Tank 18 Residual Waste Characterization Process

As with Tank 19, the Tank 18 residual waste characterization process followed heel removal and consisted of (1) sampling and analysis to determine the presence and concentration of radionuclides, (2) mapping and volume estimation to determine the amount of residual material, and (3) characterizing the residual material based on analysis, process knowledge, and quantity of material. [6]

2.5.2.1 Step 1 – Determine Radionuclide Concentration

The statistical study performed by the Statistical Consulting Section of SRNL was used to aid in the planning of the sampling process for Tank 18. As discussed in Section 2.5.1.1, the statistical study was used to help determine the number of samples required, locations of samples, and a statistical approach for characterization, in order to provide a representative characterization of the Tank 18 residual material. [53] A plan for sampling the residual material in Tank 18 was developed. [62]

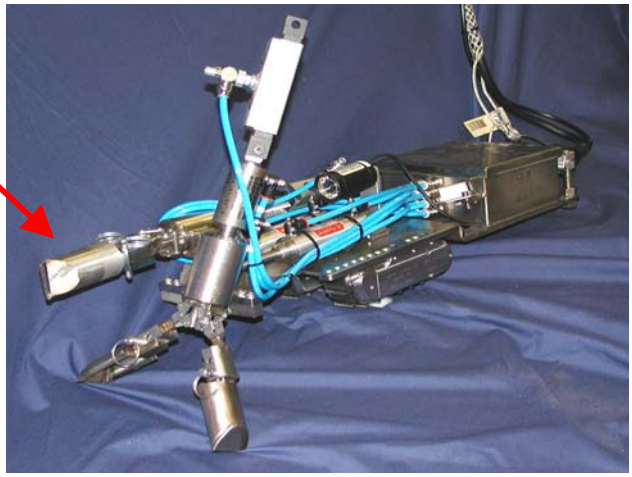
The residual solids were determined to be well-mixed except for a portion of the south region because of the >900-hour mixing campaign discussed in Section 2. Analytical results for five of six samples were similar providing indication that the solids were well-mixed, except in a portion of the south region. After the final mixing/transfer cycle, the residual solids appeared visually similar except for a portion of the south region. Furthermore, visual inspections during the heel removal process indicated that residual solids had been moved within the tank except for a portion of the south region. [6] Based on the statistical study, mixing campaign, and visual inspections it was determined that three samples were needed from the north region and three samples were needed from the south region to adequately characterize the residual waste. [62]

Samples were obtained from Tank 18 using custom-designed tools (Figure 2-43) specifically developed to extract the samples from various locations in accordance with the sample plan. Six solids samples were taken from different locations and various residual waste heights in the tank. Three samples were obtained from the north region of Tank 18 (samples FTF-213, FTF-214, and FTF-216). Three samples were obtained from the south region of Tank 18 using a remotely operated crawler (FTF-228, FTF-229, and FTF-230). In addition, one liquid sample was obtained. The single liquid sample was representative because of the >900-hour mixing campaign.

Figure 2-43: Tank 18 Sample Tools [63]

Sample container (2.5 inches long)

Sample tool used to obtain samples FTF-213, FTF-214, and FTF-216.



Remotely Operated Crawler used to obtain samples FTF-228, FTF-229, and FTF-230.

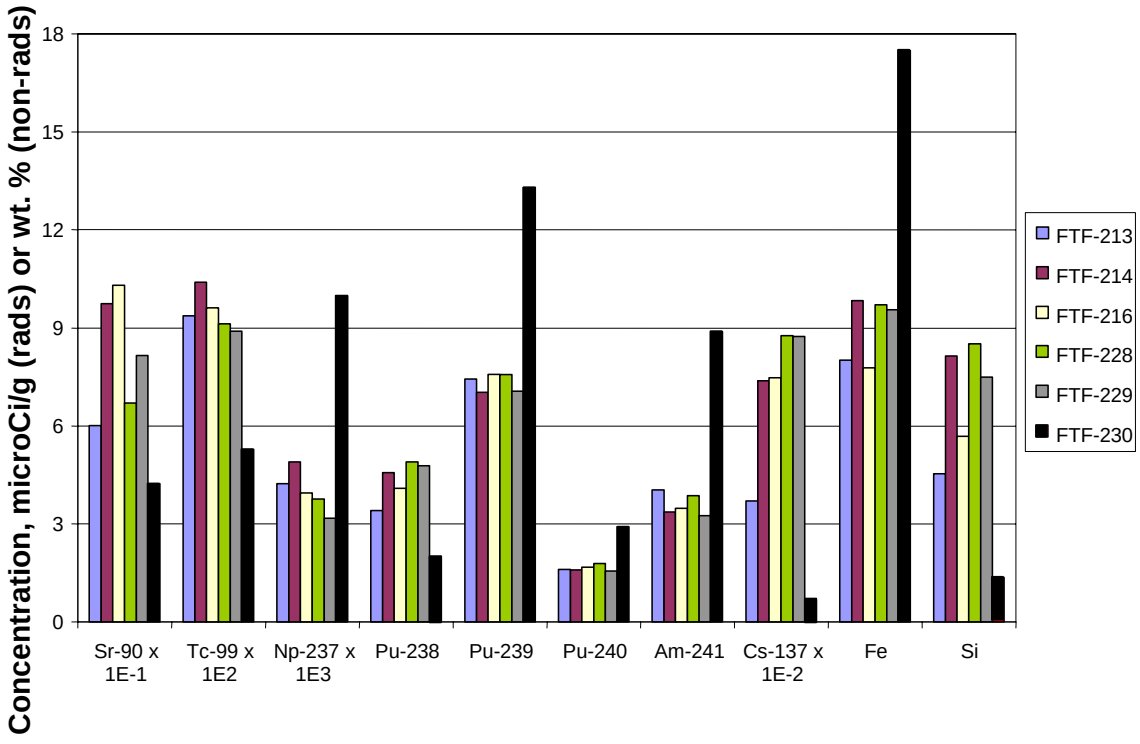
Figure 2-44: Samples in the Laboratory [63]

FTF-216 FTF-228 FTF-230 Liquid Sample from Tank 18

The samples were analyzed in the laboratory by SRNL technicians for radiological constituents important to the tank closure process (analytical techniques are further discussed in Section 2.5.2.3). Radiological constituents were selected for analysis based on their importance to the environmental modeling for the performance evaluation and 10 CFR 61.55 waste classification determination. The total Tank 18 radionuclide inventory was estimated based on a combination of analytical data, process history, and special analysis. [64]

The analytical results of two solids samples from the south region were similar to the analytical results of the three solids samples from the north region. One of the three solids samples from the south region (FTF-230) had different analytical results (Figure 2-45). As shown in Figure 2-45, the Np-237 and Pu-239 concentration from the sample in the south region was approximately twice as high as the other samples. However, the concentration of Tc-99, Sr-90, and Cs-137 was much lower than in the other five samples. [63] The similarity of the analytical results of five samples indicated that solids were well mixed except in a portion of the south region.

Figure 2-45: Variation Among Tank 18 Samples

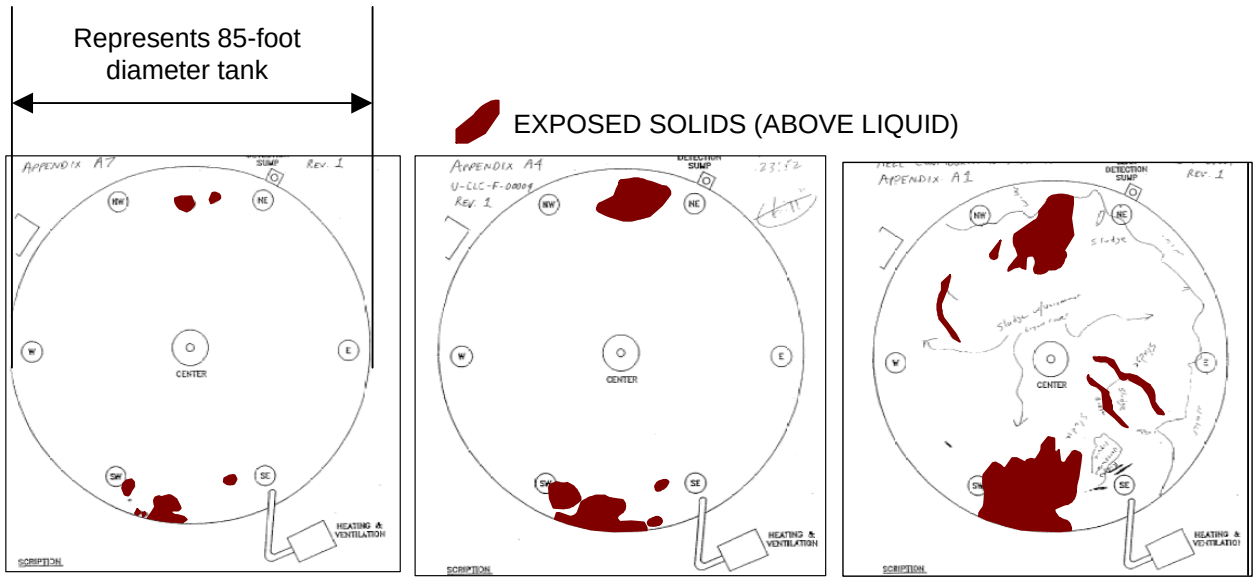


Obtaining, packaging, shipping and analysis of each sample resulted in personnel radiation exposure. Tank 18 sample dose rates were as high as 150 mrem/hr dose to the whole body. The total whole body worker dose received from the six solid samples and the one liquid sample was approximately 120 mrem. [65],[66],[67]

2.5.2.2 Step 2 – Determine the Amount of Residual Material

During transfers from Tank 18 to Tank 7 near the end of the heel removal campaign, exposed solids were mapped corresponding to liquid levels based on liquid level instrumentation (reel tape) readings and landmarks (Figure 2-46). Detailed camera inspections of the tank contents also used various landmarks within the tank to determine the locations of exposed solids. Exposed solids were mapped and residual volume was calculated 14 times in Tank 19 and three times during the Tank 18 heel removal campaign. The experience gained from these previous efforts provided the opportunity to continuously improve the process and minimize uncertainty in the final Tank 18 mapping and volume calculation. Additionally, camera technicians trained to perform tank inspections assisted in the mapping of exposed solids.

Figure 2-46: Tank 18 Exposed Solids Maps [68]



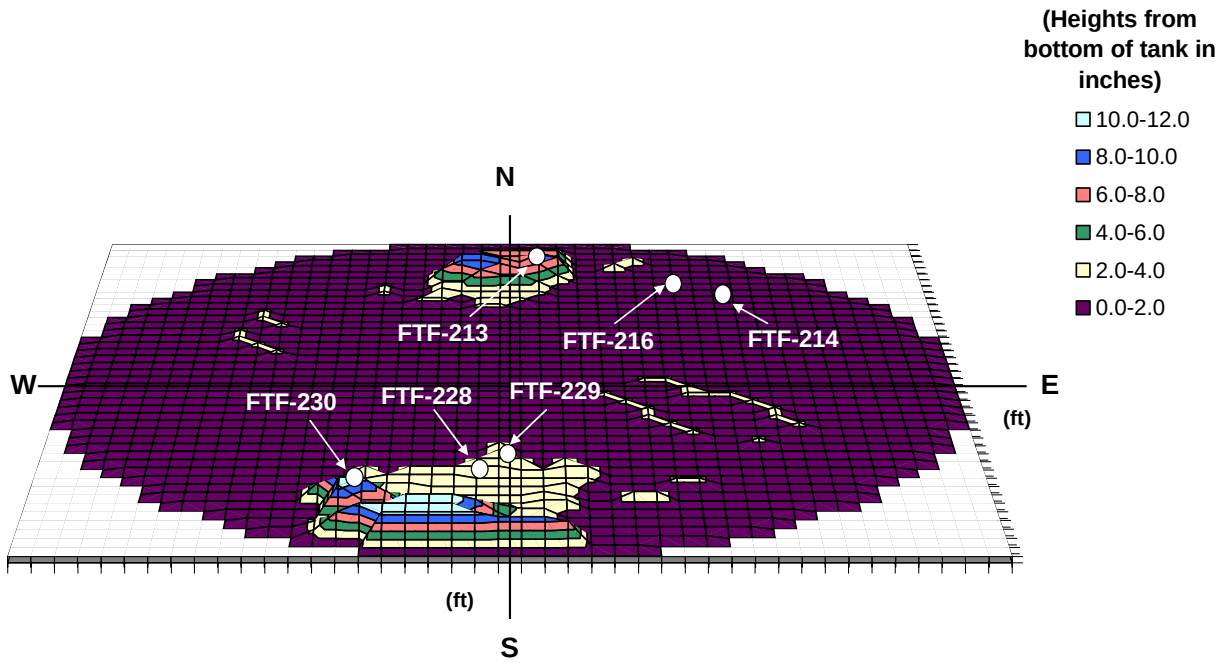
Liquid level 8.7 inches (from the bottom of tank) | **Liquid level 4.5 inches** | **Liquid level 1.5 inches**

Solids maps at multiple liquid levels combined to produce a contour map of the residual solids (Figure 2-35). The contour map was used with a tank grid as seen in Figure 2-47 to plot the height of residual material. The appropriate height of residual solids and liquid was assigned to the corresponding grid location to determine the residual solids volume and free liquid volume. [68]

Tank 19 and Tank 18 have three bands of steel angle around the inside circumference of the tank wall designed to "stiffen" and provide support to the steel tank liner. The top angle protrudes four inches from the tank wall, while the bottom two angles extend five inches from the tank wall. For Waste Tank 18, historical photographs show piles of solids built up on these stiffening bands. For Tank 18, water washing dislodged almost all of the waste present on the stiffening bands, leaving approximately 14 gallons on the lowest stiffening band in the southwest of the tank based on visual camera inspection. [6]

The residual material volumes are 6.7 kgal. Wet solids (including interstitial liquid) comprises 4.3 kgal. This represents an equivalent depth of approximately 1.2 inches in the 34-foot high tank. The free liquid comprises 2.4 kgal. The configuration of the residual material is depicted in Figure 2-47. This volume includes the equivalent of approximately 10 gallons of radioactive constituents (<< 1% of the total volume of radioactive constituents originally in Tank 18). [51] The residual material volume consists of approximately 2 kgal of zeolite and approximately 2.3 kgal of insoluble solids. [6],[29

Figure 2-47: Tank 18 Residual Material Configuration and Sample Locations [6]



Note: Each square in the grid represents a 2 foot x 2 foot area of the tank bottom

Approximately 99% of the original curies in the tank at the start of removal has been removed. [59] Approximately 86% of the remaining curies are associated with Cs-137/Ba-137m bound to the zeolite (i.e., the cesium/barium atoms captured by the zeolite resin are chemically part of the zeolite molecule).

2.5.2.3 Step 3 – Characterize the Residual Material

As with Tank 19, the radioactive inventory of each radionuclide in the residual waste in Tank 18 was determined by one of three means: (1) sample laboratory analysis, (2) special analysis, and (3) predictions based on process knowledge. As with Tank 19, the preferred bases for radionuclide inventory was laboratory analysis. Table 2-4 shows the means by which each radionuclide inventory was determined.

Table 2-4: Tank 18 Radionuclide Inventory [6],[64]

Radionuclide	Tank 18 Total Inventory (Ci)	Percent of Total Curies (%)
H-3	6.16E-02	*
C-14	5.61E-02	*
Co-60	4.35E+00	0.02%
Ni-59	9.12E-01	*
Ni-63	8.04E+01	0.29%
Se-79	5.05E-02	*
Sr-90	1.41E+03	5.04%
Y-90	1.41E+03	5.04%
Nb-94	4.01E-05	*
Tc-99	1.48E+00	*
Ru-106	9.72E-05	*
Rh-106	9.72E-05	*
Sn-126	1.40E-01	*
Sb-125	1.17E+00	*
Sb-126	1.96E-02	*
Sb-126m	1.40E-01	*
Te-125m	2.87E-01	*
I-129	6.42E-06	*
Cs-134	3.29E-03	*
Cs-135	8.74E-04	*
Cs-137	1.26E+04	45.07%
Ba-137m	1.16E+04	41.49%
Ce-144	1.44E-06	*
Pr-144	1.44E-06	*
Pm-147	1.85E+01	0.07%
Sm-151	4.58E+01	0.16%
Eu-152	1.98E-01	*
Eu-154	1.06E+01	0.04%
Eu-155	2.67E+00	*
Ra-226	4.90E-07	*

Radionuclide	Tank 18 Total Inventory (Ci)	Percent of Total Curies (%)
Ra-228	0.00E+00	0
Ac-227	1.64E-06	*
Th-229	2.27E-03	*
Th-230	6.00E-05	*
Th-232	0.00E+00	0
Pa-231	4.57E-06	*
U-232	2.18E-04	*
U-233	4.41E+00	0.02%
U-234	2.53E+00	*
U-235	9.31E-03	*
U-236	3.15E-02	*
U-238	2.37E-01	*
Np-237	1.18E-01	*
Pu-238	7.01E+01	0.25%
Pu-239	1.40E+02	0.50%
Pu-240	5.81E+01	0.21%
Pu-241	2.52E+02	0.90%
Pu-242	5.60E-01	*
Pu-244	3.39E-04	*
Am-241	7.41E+01	0.26%
Am-242m	8.03E-01	*
Am-243	6.15E-06	*
Cm-242	3.19E-20	*
Cm-243	9.52E-05	*
Cm-244	1.69E+02	0.60%
Cm-245	2.17E-09	*
Cm-247	2.17E-18	*
Cm-248	5.02E-19	*
Bk-249	1.89E-28	*
Cf-249	1.39E-20	*

*	<0.01% of Total Curies
	Sample Analysis
	WCS
	Special Calculation

Total Curies	2.8E+04

For radionuclides that were not analyzed in the laboratory, the WCS was used to determine radionuclide inventory based on process knowledge. See Appendix C for a description of WCS. [64]

Special analyses were performed to determine radionuclide inventories that had not been historically tracked in WCS. Examples of special analyses include fission yield ratios to other radionuclides and equations for progeny radionuclides in terms of the appropriate parent radionuclide. [64]

The samples were analyzed in the laboratory by SRNL technicians for selected radiological constituents important to the tank closure process. [63] Following the same analysis process performed for the Tank 19 samples (described in subsection 2.5.1.3), each sample underwent six analyses. As discussed in Section 2.5.1.3, the attributes of the radioanalytical techniques provided high confidence in the radioanalytical data for Tank 18 samples. The 95% upper-confidence limit value of the average concentration of the six analyses was used for each analyzed radionuclide. The radionuclide inventory values in Table 2-4 were used for the classification determination described in Section 6 and the performance evaluation described in Section 7 of this document. The results of the characterization of the residual material on the tank floor are included in the radionuclide inventory report. [64]

In addition to the residual material on the tank floor, an analysis of the radionuclide inventory of corrosion products on the interior of the tank was performed (following the same process as performed for Tank 19, as described in subsection 2.5.1.3). [69] SRNL estimated that there were approximately 700 pounds of corrosion product (rust) in Tank 18. [70] It has been estimated that the corrosion products for Tank 18 represent approximately 400 curies (~1.4% of the tank's total residual waste curies). For Tank 18, approximately 88% of the corrosion product curies is from Cs-137. Radionuclides that have potentially been held on the corrosion products on the Tank 18 interior surfaces are included in the Tank 18 total radionuclide inventory reported in Table 2-4. [6]

FTF waste tank residual material inventory in shown in Appendix E.

2.6 Tank Stabilization

In May 2002, DOE issued an EIS on tank cleaning and stabilization alternatives. [2] DOE studied four alternatives: (1) fill with grout, (2) fill tank with sand, (3) fill tank with saltstone, and (4) clean and remove tanks. The EIS concluded the Fill with Grout option under the Stabilize Tanks Alternative was preferred. DOE also issued a ROD selecting the Fill with Grout alternative for SRS tank closure. [8]

Evaluations described in the EIS showed the Fill With Grout alternative to be the best approach to minimize human health and safety risks associated with closure of the tanks. [2] This alternative offers several advantages over the other alternatives evaluated such as:

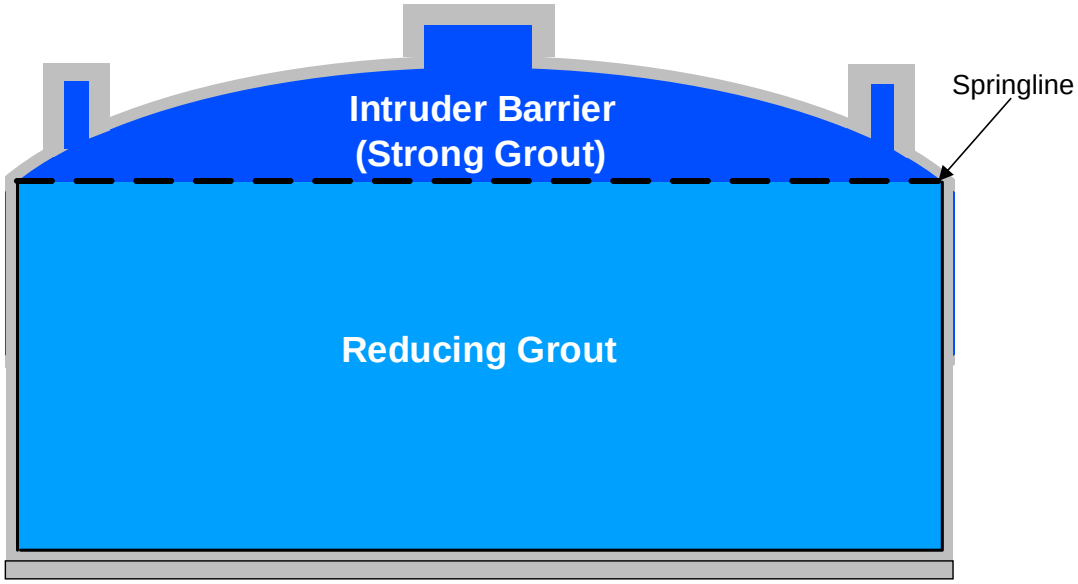
- Provides greater long-term stability of the tanks and their residual waste than the sand-fill approach;
- Provides for retaining radionuclides within the tanks by use of reducing agents in a fashion that the sand-fill would not;

- Avoids the technical complexities and additional worker radiation exposure that the fill-with-saltstone approach would entail;
- Produces smaller impacts due to radiological contaminant transport than the sand- and saltstone-fill alternatives,
- Avoids the excessive personnel radiation exposure and greater occupational safety impact that would be associated with the clean and remove alternative.[8] [2]

2.6.1 Tank 19 and Tank 18 Stabilization

DOE plans to use two kinds of grout for closure of Tank 19 and Tank 18 (Figure 2-48). The first grout is a reducing grout, selected from the results of laboratory and field testing of various mixes to provide structural stability to the waste as mandated by 10 CFR 61.56. The reducing grout is composed primarily of sand, water, fly ash, slag, cement, and sodium thiosulfate. [71] The reducing grout is pumpable through pipe lines up to 1,500-feet in distance from the grout plant, and will be used to fill the tank from the bottom to approximately the beginning of the dome (commonly referred to as the springline). [72] The solidified reducing grout will also provide a long-term chemical environment to minimize leaching of radionuclides from the residual waste. [72] The reducing grout also provides compressive strength with minimal bleeding. [72] Bleeding is the migration of water to the top surface of freshly placed grout caused by settling of the solid material. Minimal bleeding encourages aspiration of residual liquid. [72]

Figure 2-48: Typical Tank 19 and Tank 18 Grout Pour



[NOT TO SCALE]

[8] Personnel radiation exposure and occupational safety impacts are shown in Table 4.1.8-1, Table 4.1.8-2, and Table 4.1.8-3 in the EIS.

A second type of grout will be used to fill the tank from the springline up to the bottom of the tank top riser ports. This will be a "strong" grout poured on top of the reducing grout. This "strong" grout is designed to have a compressive strength in the normal concrete range of 2,000 to 3,000 psi. In addition, this "strong" grout will prevent subsidence of the tank dome and discourage an intruder from inadvertently accessing the waste. This layer will vary from approximately one-foot at the tank perimeter to approximately 11-feet at the center riser location. The intruder barrier grout consists mainly of cement and sand. This grout is designed to provide the greater strength needed to deter inadvertent intrusion. [73]

The reducing grout and intruder barrier grout provide sufficient strength to support a future overbearing cap, thereby preventing subsidence. Both grout mixes have low viscosity and free-flowing properties so that they will be self-leveling across the 85-foot diameter tank and will effectively fill void spaces. Properties of the Tank 19 and Tank 18 reducing grout and intruder barrier grout are described in Table 2-5. [74],[75]

Table 2-5: Tank 19 and Tank 18 Grout Properties [71]

Purpose	Reducing Grout Property	Intruder Barrier Grout Property
Prevent subsidence	Minimum compressive strength >50 psi.	Minimum compressive strength >2000 psi.
Provide deterrence to inadvertent intrusion	NA	Minimum compressive strength >2000 psi. Recognizable and different than surrounding soil.
Provide durability	Quartz sand is cemented together by binder material (hydrated cement, pozzolans, slag, and fly ash) resulting in a cohesive solidified mass.	Quartz sand is cemented together by binder material (hydrated cement, pozzolans, slag, and fly ash) resulting in a cohesive solidified mass.
Pourable from a single location and fill void spaces	Low viscosity and self-leveling.	Low viscosity and self-leveling.
Provide chemical reducing properties	Eh <0mV pH >9.5	NA

The components of the grout mixes selected for closure of Tank 19 and Tank 18 are listed in Table 2-6. [71]

Table 2-6: Tank 19 and Tank 18 Grout Components [73]

Component	Reducing Grout	Intruder Barrier Grout
Portland Cement	X	X
Sand	X	X
Slag	X	
Water	X	X
Fly Ash	X	
Sodium Thiosulfate	X	
High Performance Zero Bleed Admixture	X	X

In addition to the residual waste solids at the bottom of Tank 19 and Tank 18, residual contamination will remain on some equipment within the tank system as well as on the tank walls and dome. This equipment consists of a transfer jet, thermowells, level instrumentation, a leak detection system, transfer piping out of the tank, and equipment directly used in tank closure operations (such as submersible mixers and pumps, cables, temporary transfer hoses and backfill transfer piping). In addition, steel tapes and other miscellaneous debris will remain in the tank after closure. These components will be entombed in the grout as part of the closure process. As the tank is filled, grout will migrate into abandoned tank and riser penetrations (including the "pill box" vaults), thereby sealing and effectively isolating the abandoned lines. The closing and grouting of external transfer lines associated with Tank 17 through Tank 20 are not addressed in this waste determination. [76]

The process for grouting the tank entails pouring the reducing grout into the bulk tank area, stopping to apply a dry grout "top dressing" to absorb free liquid on an as-needed basis; continuing to pour the reducing grout pour until it reaches the beginning of the tank's dome (the springline, see Figure 2-48); and, finally, filling the concrete dome and associated risers with the intruder barrier grout. Grout placement will be observed and monitored by in-tank cameras. [76]

The grout installation system will consist of a grout plant, a grout transfer pipeline, and the tank grout downcomer. The grout plant will be located approximately 1000-feet from Tank 19 and Tank 18 and will be equipped with feed hoppers and silos that feed basic materials into a pump hopper. Grout will be pumped from the hopper through a transfer pipeline routed directly to the downcomer located at the tank riser. The downcomer, a flexible delivery pipe, directs the grout to the proper location inside the tank. [73]

2.6.2 Tank 19 and Tank 18 Post Stabilization

Following stabilization, Tank 19 and Tank 18 will ultimately be backfilled with soil at a depth that will provide at least 10-feet of soil above the highest tank elevation. Optionally, a low-permeability cover may also be installed.

The SRS lies within the Aiken Plateau, and this Plateau slopes to the southeast approximately five-feet per mile. The Plateau is bounded by the Savannah and Congaree Rivers, and extends from the fall line to the Orangeburg Excarpment. The highly dissected surface of the Aiken Plateau is characterized by broad interfluvial areas with narrow, steep-sided valleys.

The topography of the general area of FTF has a high level of approximately 295-feet to a low level of approximately 288-feet above mean sea level (MSL). Local relief can be as much as 300-feet. The ground surface elevation in the depression around Tank 19 and Tank 18 is approximately 270-feet above MSL based on ground surface elevation from the groundwater monitoring wells closest to the tank. Tank 19 and Tank 18 were placed below the original site grade to facilitate gravity feed of waste. The soils in the vicinity of Tank 19 and Tank 18 are well-drained soils that formed in heterogeneous materials, which are the spoil or refuse from excavations and major construction operations. The area around the tanks was backfilled with clay and topsoil to the top of the tanks. The base elevation is 228-feet MSL for Tank 19 and 229-feet MSL for Tank 18. For comparison, neighboring Tank 25 and Tank 28, located above the depression, have base elevations of 249-feet MSL. The 500-year and 100-year flood plains are located at 153.1-feet above MSL (74.9-feet below the bottom of Tank 19) and 151.7-feet above MSL (76.3-feet below the bottom of Tank 19), respectively. [77] The nearest surface water of significance flows in Fourmile Branch, which runs approximately one mile to the south. Upper Three Runs runs approximately two miles to the north.

Seismic activity associated within the entire SRS is closely related to the South Carolina Piedmont-type activity. This activity is characterized by occasional small, shallow seismic events associated with strain release near small scale faults, intrusive bodies, and the edges of metamorphic belts, however, the Charleston area is the most significant seismic zone affecting the SRS. Seismic history for the southeast covers a period of approximately 300 years, with the most significant earthquake event being that of Charleston 1886. Two major earthquakes have occurred within 100 miles of SRS. The largest known earthquake to affect SRS was the Charleston earthquake of 1886, with an epicenter approximately 90 miles from SRS and a magnitude of 6.6 on the Richter Scale. It is estimated that an earthquake of this magnitude would result in a peak ground acceleration of 0.10g at SRS. [78] The second earthquake occurred approximately 90 – 100 miles from SRS, with an estimated magnitude of 4.5.

The most recent event occurring within a 50 mile radius of SRS was the August 8, 1993, Couchton earthquake. No damage to the SRS was recorded. However, there have been three recorded earthquakes with epicentral locations within SRS. They occurred on June 9, 1985, August 5, 1998, and May 17, 1997. No Strong Motion Accelerometers (SMA) were triggered as a result of these earthquakes.

The Performance Objective Determination Document (PODD) includes analysis of the duration of reduction capacity with cracks in the grouted tank, as well as sensitivity analysis that quantitatively evaluated different parameters associated with the grouted tank. [4] Sensitivity analysis can be used to understand the consequences resulting from earthquakes. For example, sensitivity analysis were conducted on the time of hydraulic failure, infiltration rates, and movement of contaminants in the soil. More detailed information on sensitivity analysis and results is discussed in Section 5 of the PODD. [4]

3.0 SECTION 3116 OF THE RONALD W. REAGAN NATIONAL DEFENSE AUTHORIZATION ACT FOR FISCAL YEAR 2005

Section 3116 (a) of the NDAA provides:

> In General – Notwithstanding the provisions of the Nuclear Waste Policy Act of 1982, the requirements of Section 202 of the Energy Reorganization Act of 1974, and other laws that define classes of radioactive waste, with respect to material stored at a DOE site at which activities are regulated by a covered State pursuant to approved closure plans or permits issued by the State, the term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy (in this section referred to as the "Secretary"), in consultation with the Nuclear Regulatory Commission (in this section referred to as the "Commission"), determines -
>
> (1) does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste;
>
> (2) has had highly radioactive radionuclides removed to the maximum extent practical; and
>
> (3) (A) does not exceed concentration limits for Class C low-level waste as set out in Section 61.55 of title 10, Code of Federal Regulations, and will be disposed of -
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or
>
> (B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will be disposed of -
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations, and
>
> (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; and
>
> (iii) pursuant to plans developed by the Secretary in consultation with the Commission.

4.0 WASTE DOES NOT REQUIRE PERMANENT ISOLATION IN A DEEP GEOLOGIC REPOSITORY FOR SPENT FUEL OR HIGH-LEVEL RADIOACTIVE WASTE

Section 3116 (a) of the NDAA provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy…, in consultation with the Nuclear Regulatory Commission…, determines –
>
> (1) does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste.

The purpose of Section 3116 is to clarify the Secretary's authority, in consultation with the NRC, to determine that certain waste from reprocessing that meets the criteria set out in the section should not be classified as "high-level radioactive waste." Section 3116(a) sets out two specific criteria for making this determination in clauses (2) and (3). Clause (2) requires DOE to remove highly radioactive radionuclides to the maximum extent practical. Clause (3) generally mirrors the criteria that the NRC has established for determining whether waste qualifies for land disposal as low-level-waste (10 CFR. 61.55). This clause provides that disposal of the waste must meet the NRC performance objectives of 10 CFR 61, Subpart C, and that the waste must not exceed the concentration levels for Class C waste in 10 CFR 61.55 or the Secretary must consult with the NRC concerning DOE's disposal plans.

Clause (1) noted above, is a broader criterion for the Secretary, in consultation with the NRC, to consider whether, notwithstanding that waste from reprocessing meets the other two criteria, there are other considerations that, in the Secretary's judgment, require its disposal in a deep geologic repository. Generally, such considerations would be an unusual case because waste that meets the third criterion would be waste that will be disposed of in a manner that meets the 10 CFR 61, Subpart C performance objectives and either falls within one of the classes set out in 10 CFR 61.55 that the NRC has specified are considered "generally acceptable for near-surface disposal" or for which the Secretary has consulted with NRC concerning DOE's disposal plans. As the NRC explained in *In the Matter of Louisiana Energy Services, L.P. (National Enrichment Services)* (CLI-05-05, 2005), the 10 CFR 61, Subpart C performance objectives in turn "set forth the ultimate standards and radiation limits for (1) protection of the general population from releases of radioactivity, (2) protection of individuals from inadvertent intrusion, (3) protection of individuals during operations, and (4) stability of the disposal site after closure." It follows that if disposal of a waste stream in a facility that is not a deep geologic repository will meet these objectives, in the ordinary case that waste stream does not "required disposal in a deep geologic repository" because non-repository disposal will be protective of public health and safety.

It is possible that in rare circumstances, a waste stream that meets the third criterion might have some other unique radiological characteristic or may raise unique policy considerations that warrant its disposal in a deep geologic repository. Clause (1) is an acknowledgment by Congress of that possibility. For example, the waste stream could contain material that, while not

presenting a health and safety danger if disposed of at near- or intermediate-surface, nevertheless presents non-proliferation risks that the Secretary concludes cannot be adequately guarded against absent deep geologic disposal. Clause (1) gives the Secretary, in consultation with the NRC, the authority to consider such factors in determining whether waste that meets the other two criteria nonetheless requires disposal in a deep geologic repository in light of such considerations.

That is not the case here. As demonstrated later in this draft 3116 Determination, disposal of the Tank 19 and Tank 18 solidified residual waste will meet the performance objectives of 10 CFR 61, Subpart C so as to provide for the protection of the public health and the environment. Accordingly, the waste does not require disposal in a deep geologic repository due to risk to public health and safety. Furthermore, disposal of the solidified residual waste in Tank 19 and Tank 18 does not raise any unique considerations that, notwithstanding these demonstrations, nevertheless require its permanent isolation in a deep geologic repository. Accordingly, the Tank 19 and Tank 18 solidified residual waste qualifies for classification as other than high-level waste under Clause (1) of Section 3116(a).

5.0 WASTE HAS HAD HIGHLY RADIOACTIVE RADIONUCLIDES REMOVED TO THE MAXIMUM EXTENT PRACTICAL

Section Purpose

The purpose of this section is to demonstrate that highly radioactive radionuclides have been removed from the waste to the maximum extent practical.

Section Contents

Section 5.1 identifies highly radioactive radionuclides for the purpose of this determination. Section 5.2 explains how these radionuclides were removed to the maximum extent practical, providing tables showing removal effectiveness. Section 5.3 summarizes a cost-benefit analysis performed to evaluate potential benefits, costs, and risks that would be associated with further efforts to remove residual waste.

Key Points

- The list of highly radioactive radionuclides for Tank 19 and Tank 18 describes the radionuclides that could reasonably be expected to exist in the waste and that, using a risk-informed approach, contribute significantly to the radiological risk to workers, the public, and the environment, taking into account scientific and health physics principles, knowledge, and expertise.
- Over 99% of the remaining radioactivity in the residual material, after waste removal (2002) inside Tank 19, and approximately 86% of remaining radioactivity in the residual material, after waste removal (2003) inside Tank 18, are associated with Cs-137 and its progeny Ba-137m that are chemically bound to zeolite remaining in the tank.
- Risk benefit analyses considering a variety of waste removal methods showed that complete further of residual waste in the tanks would not result in significant dose savings. Risk to workers who would be involved in such an effort would be greater than potential risk reduction to the public from waste removal.

Section 3116(a) of the NDAA provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy …, in consultation with the Nuclear Regulatory Commission …, determines— …
> (2) has had highly radioactive radionuclides removed to the maximum extent practical.

5.1 Highly Radioactive Radionuclides

5.1.1 Approach

Based on consultations with NRC, DOE views "highly radioactive radionuclides" to be those radionuclides that, using a risk-informed approach, contribute most significantly to radiological risk to workers, the public, and the environment. Table 5-1 lists the radionuclides for the grouted (stabilized) Tank 19 and Tank 18 residual waste at closure that DOE has determined, on the basis of a risk-informed approach, contribute significantly to radiological risk to workers, the public, and the environment, taking into account scientific and health physics principles, knowledge, and expertise.

The list of highly radioactive radionuclides, for the purpose of this Determination, was developed, beginning with the entire inventory of radionuclides stored in Tank 19 and Tank 18 prior to waste removal.[9] The DOE reviewed this inventory of radionuclides and identified those radionuclides in Tables 1 and 2 of 10 CFR 61.55,[10] as well as any additional radionuclides that may be important to meeting the performance objectives in 10 CFR 61, Subpart C because they contribute to the dose to workers, the public, and/or the inadvertent intruder in the expected and degraded cases, using sensitivity analyses. In DOE's view, this approach results in a risk-informed list of highly radioactive radionuclides that includes: those short-lived radionuclides that may present risk because they produce radiation emissions that, without shielding or controls, may harm humans simply by proximity to humans without inhalation or ingestion; and those long-lived radionuclides that persist well into the future, may be mobile in the environment, or may pose a risk to humans if inhaled or ingested. This list includes the short-lived fission products Cs-137 and Sr-90 and their equilibrium daughters, Ba-137m and Y-90, that are responsible for the bulk of activity in the Tank 19 and Tank 18 inventory. Based on sample analysis [5],[6], Cs-137 and Sr-90 and their equilibrium daughters, Ba-137m and Y-90, account for approximately 99% of the activity in Tank 19 after waste removal (2002) and approximately 97% of the activity in Tank 18, after waste removal (2003). This list also includes the long-lived actinides (Np-237, the plutonium isotopes, Am-241, Am-243, Am-242m, Cm-244, Cm-245, Cm-246, Cm-247, Cm-248, Cf-249) that are responsible for approximately 0.14% of the activity in Tank 19 after waste removal (2002), and approximately 2.73% of the activity in Tank 18 after waste removal (2003). The other radionuclides included in the list are C-14, H-3, I-129, Nb-94, Ni-59, Co-60, Ni-63, Tc-99, and Cm-242, which contribute approximately 0.95% of the activity in Tank 19 and approximately 0.31% of the activity in Tank 18 after waste removal (2003). The list of highly radioactive radionuclides (shown in Table 5-1) account for approximately 99% of the radioactivity in Tank 19, as of 2002, and Tank 18, as of 2003, based on process and sampling knowledge.

[9] The DOE has reviewed the inventory of 68 radionuclides identified as being present prior to waste removal in Tank 19 and Tank 18. [11],[87]

[10] Although Tables 1 and 2 in 10 CFR 61.55 specify concentration limits for certain radionuclides in the form of activated metal, DOE has included such radionuclides, if present in the waste, in the list of "highly radioactive radionuclides" as it exists in the waste, without regard to whether such radionuclides are in the form of activated metal. Consistent with Table 1, DOE has excluded alpha-emitting transuranic nuclides with half-lives of 5 years or less from the list of highly radioactive radionuclides. Some of the radionuclides listed as highly radioactive radionuclides in this draft 3116 Determination may not be listed in other 3116 Determinations if such radionuclides are not present in the waste or do not contribute to dose to the worker, the public, or the intruder.

5.1.2 Radionuclides from Performance Objective Demonstration

As explained above, DOE has included in the list of highly radioactive radionuclides (shown in Table 5-1) those radionuclides that may be important to meeting the performance objectives of 10 CFR 61, Subpart C because they contribute to the dose to workers, the public, and/or the inadvertent intruder based on the Tank 19 and Tank 18 PODD, which includes sensitivity and uncertainty analyses. [4] The DOE used the following information from the PODD to focus on those radionuclides that contribute to the dose for various pathways.

5.1.2.1 Groundwater Radionuclide Screening

Numerical modeling and dose assessment was considered when developing the list of highly radioactive radionuclides. The Tank 19 and Tank 18 inventories include radionuclides that are significant in terms of projected receptor doses from the groundwater pathway. The Tank 19 and Tank 18 groundwater analysis included all radionuclides identified in the residual material.

The groundwater analysis considered that active institutional control will be maintained over the disposal site for 100 years after facility closure and, furthermore, that the performance of the facility will be monitored to detect and/or prevent significant releases of radionuclides to the environment throughout the period of active institutional control. This consideration eliminates from concern any radionuclide with a half-life of less than five years because the inventory in the waste at 100 years after facility closure will be reduced to innocuous levels by radioactive decay. However, radionuclides with half-lives of less than five years cannot be neglected if the radionuclide appears in a decay chain. This is because its activity may increase with time due to decay of a parent radionuclide, unless the parent is also short-lived.

Using a five-year half-life screening criteria results in the elimination of the following radionuclides from further consideration: Cs-134, Eu-155, Na-22, and Th-228. In addition, the following radionuclides in short decay chains were eliminated from further analysis since the parent and progeny each have half-lives of less than 5 years: Ru-106/Rh-106, Sb-125/Te-125m, Sb-126m, and Ce-144/Pr-144.

Table 5-1: List of Highly Radioactive Radionuclides

Radionuclide	Radionuclide Half-Life (yr)	Long-Term Radiation Hazards	Short-Term Radiation Hazards
Am-241[b]	4.32E+02	X	
Am-242m[b]	1.41E+02	X	
Am-243[b]	7.37E+03	X	
C-14[b]	5.73E+03	X	
Cm-242[b,c]	4.50E-01	X	
Cm-244[b]	1.81E+01		X
Cm-245[b]	8.50E+03	X	
Cm-246[b]	4.76E+03	X	
Cm-247[b]	1.56E+07	X	
Cm-248[b]	3.48E+05	X	
Co-60[d]	5.28E+00		X
Cs-137[a,d]	3.01E+01		X
Ba-137m[a]	4.85E-06		X
H-3[d]	1.23E+01		X
I-129[b]	1.57E+07	X	
Nb-94[b]	2.03E+04	X	
Ni-59[b]	7.60E+04	X	
Ni-63[d]	1.00E+02		X
Np-237[a,b]	2.14E+06	X	
Pu-238[a,b]	8.77E+01	X	
Pu-239[b]	2.41E+04	X	
Pu-240[b]	6.56E+03	X	
Pu-241[b,c]	1.43E+01	X	
Pu-242[b]	3.73E+05	X	
Pu-244[b]	8.00E+07	X	
Sr-90[a,d]	2.88E+01		X
Y-90[a]	7.30E-03		X
Tc-99[a,b]	2.11E+05	X	
Cf-249[b]	3.51E+02	X	

a. Highly radioactive radionuclides based on dose assessment results from the 2005 PODD. [4]
b. Taken from Table 1 of 10 CFR 61.55. [79]
c. Daughter product of long-lived radionuclide.
d. Taken from Table 2 of 10 CFR 61.55. [79]

The next step involved evaluation of the release of radionuclides from the waste form and the resulting groundwater concentrations. Releases and groundwater concentrations were previously analyzed by modeling. The results are presented in Appendix C of the PODD. [4] Based on these analyses, only Tc-99 was found to result in concentrations in groundwater that contribute significantly to doses for Tank 19 (at least 99% of the all-pathways dose is attributable to this radionuclide). Only Np-237 was found to result in concentrations in groundwater that contribute significantly to doses for Tank 18 (at least 99% of the all-pathways dose is attributable to this radionuclide). All other radionuclides that did not contribute to the dose were not included in the list of highly radioactive radionuclides for the purpose of this Determination.

The *Savannah River Site High-Level Waste Tank Closure Final Environmental Impact Statement* also conducted screening and groundwater transport analyses for FTF, which showed that only Tc-99 and Np-237 were dominant dose contributors from the groundwater pathway. [8] Therefore, based on previous modeling of the FTF presented in the EIS, radionuclides (i.e., Tc-99 and Np-237) were determined to result in appreciable groundwater concentrations and doses. Therefore, the groundwater analysis included these two radionuclides, and they were included in the list of highly radioactive radionuclides for this rationale and their presence in Table 1, 10 CFR 61.55, *"Waste Classification"* for the purpose of this Determination.

5.1.2.2 Intruder Pathway Radionuclide Screening

The intruder analysis considers that active institutional control will be maintained over the disposal site for at least 100 years after facility closure. This consideration eliminates from concern any radionuclide with a half-life of less than five years.

The radionuclide inventory associated with a FTF transfer line (See Appendix E) is used for the intruder scenario. The radionuclide inventory basis for the FTF transfer line is explained in Section 3 of the PODD. [4] A review of the PODD intruder analysis shows that approximately 89% of the total maximum dose was due to Sr-90. Another approximately 8% of the total dose was due to Cs-137 and approximately 2% was due to Pu-238. The remaining radionuclides contributed less than 1% of the dose.

5.2 Radionuclide Removal to the Maximum Extent Practical

Section 3116(a) of the NDAA provides that certain waste resulting from reprocessing is not HLW if the Secretary, in consultation with the NRC, determines, among other things, that the waste has had highly radioactive radionuclides removed "to the maximum extent practical." This subsection discusses the basis on which the Secretary may conclude that DOE's approach for removing the highly radioactive radionuclides listed above meets this criterion.

Removal to the maximum extent "practical" is not removal to the extent theoretically "possible." Rather, a "practical" approach to removal is one that is "adapted to actual conditions" [80]; "adapted or designed for actual use" [81]; "useful" [81]; selected "mindful of the results, usefulness, advantages or disadvantages, etc., of [the] action or procedure" [81]; fitted to "the needs of a particular situation in a helpful way" [82] ; "effective or suitable". [82] Therefore, the determination as to whether a particular highly radioactive radionuclide will be removed to the maximum extent practical will vary from situation to situation, not only on the available technologies but also on the overall costs and benefits[11] of deploying a technology with respect to a particular waste stream. The "maximum extent practical" standard contemplates room for exercising expert judgment in weighing several factors. Such factors may include environmental, health, timing, or other exigencies; the risks and benefits to public health, safety, and the environment arising from further radionuclide removal as compared with countervailing considerations that may ensue from not removing or delaying removal; the reasonable availability of proven technologies; the usefulness of such technologies; and the sensibleness of

[11] While prior NRC and DOE requirements for waste determinations called for removal "to the maximum extent *technically* and *economically* practical" (Bernero 1993; DOE M 435.1-1, 2001), Section 3116 omits these adverbs, thereby suggesting that a broad range of considerations, including but not limited to technical and economic practicalities, may appropriately be taken into account in determining the extent of removal that is practical.

using such technologies. What may be removal to the maximum extent practical in a particular situation or at one point in time may not be that which, on balance, is practical, feasible, or sensible in another situation or at a later point in time.

Although not an issue here, DOE notes that it may not be practical to undertake any further removal of certain radionuclides because further removal is not sensible or useful in light of the overall benefit to human health or the environment. As a general matter, such a situation may arise if certain radionuclides are present in such extremely low quantities that they make an insignificant contribution to potential doses to workers, the public, and the hypothetical human intruder.[12] [83]

Radionuclides have been removed from Tank 19 and Tank 18 to the maximum extent practical by implementing a two-phase, three-step plan to remove waste (described in Subsection 2.4). To demonstrate compliance with Section 3116(a)(2), the list of highly radioactive radionuclides identified in Subsection 5.1, the total inventory of radionuclides in Tank 19 and Tank 18 prior to waste removal, and the residual waste inventory at closure and disposal are used to determine the effectiveness of radionuclide removal. These radionuclides in the list of highly radioactive radionuclides in Table 5-1 are listed or referred to in 10 CFR 61.55, "Waste Classification," are significant contributors to the Performance Objective Determination dose calculations, and contribute to the radionuclide inventory. [59] The Tank 19 and Tank 18 cleaning activities resulted in the removal of radionuclides to the maximum extent practical. Further removal of the residual waste inventory is not practical, as discussed below.

Tank 19 Waste Removal

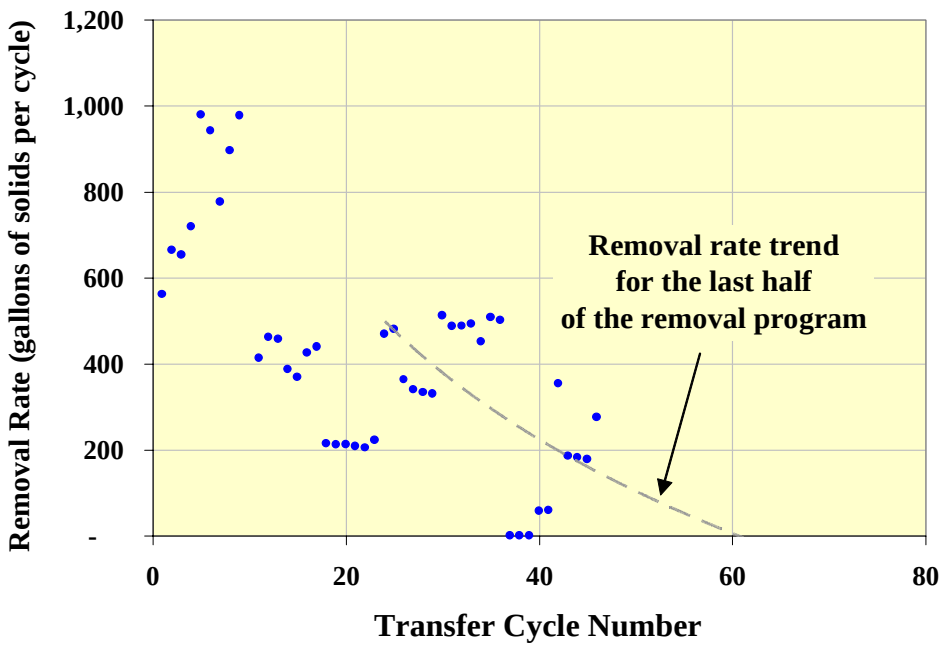
As discussed in Subsection 2.4.1, bulk waste removal efforts removed approximately 277 kgal of liquid. [84] After the liquid waste was removed, 1,033 kgal of saltcake was dissolved and removed. Insoluble, hardened, and fast-settling zeolite was resistant to the bulk waste removal method employed to remove the saltcake. However, zeolite, which was entrained within the saltcake, underwent washing cycles that effectively stripped out the soluble Cs-137, leaving behind the Cs-137 that is chemically bound to the zeolite. [33]

Following the completion of bulk waste removal, a selection process identified rotating submersible mixer pumps as the preferred technology for the Tank 19 heel removal system (as discussed in Subsection 2.4.1.4). [36] The heel removal campaign removed a large portion of

[12] The DOE normally would view radionuclides as making an insignificant contribution if the contribution to dose from those radionuclides, in both the expected case and fully degraded case using sensitivity analysis, does not exceed any of the following: (1) 10% of the 25-mrem/yr all-pathways annual dose to the public, (2) 10% of the DOE 100-mrem annual dose limit to the intruder (under all reasonable intruder scenarios), (3) 10% of the DOE 500-mrem acute dose limit to the intruder (under all intruder scenarios), and (4) 10% of the annual worker dose in the relevant provisions of 10 CFR 20. For perspective, DOE would also consider the extent to which such radionuclides contribute to an annual dose of 4 mrem to the receptor from the groundwater pathway. This methodology is based on NRC consultation and is intended to be consistent with the guidance and general approach in Volume 2 of NUREG-1757, *Consolidated NMSS Decommissioning Guidance* [83], which explains that "NRC staff considers radionuclides and exposure pathways that contribute no greater than 10% of the dose criteria to be insignificant contributors." The above-reference NUREG, which applies to NRC licensees, is being used only as general guidance, and DOE's use of this NUREG as guidance should not be construed to suggest that it is a requirement under Section 3116 of the NDAA or that either the NUREG or 10 CFR 20, Subpart E is applicable in the 3116 context.

the heel. [43] After forty-six mixing/transfer cycles, Cs-137 was no longer being washed out. The remaining chemically-bound Cs-137 that could not be removed by these techniques will likewise not be readily mobilized by water infiltration after closure. The heel removal rate declined as the transfer cycles progressed until the heel removal campaign was stopped (Figure 5-1). Of the entire 1,300 kgal of material in Tank 19, the overall waste removal program removed all but 15 kgal of material, of which approximately two gallons are radioactive isotopes. [44]

Figure 5-1: Tank 19 Heel Removal Rate [41]



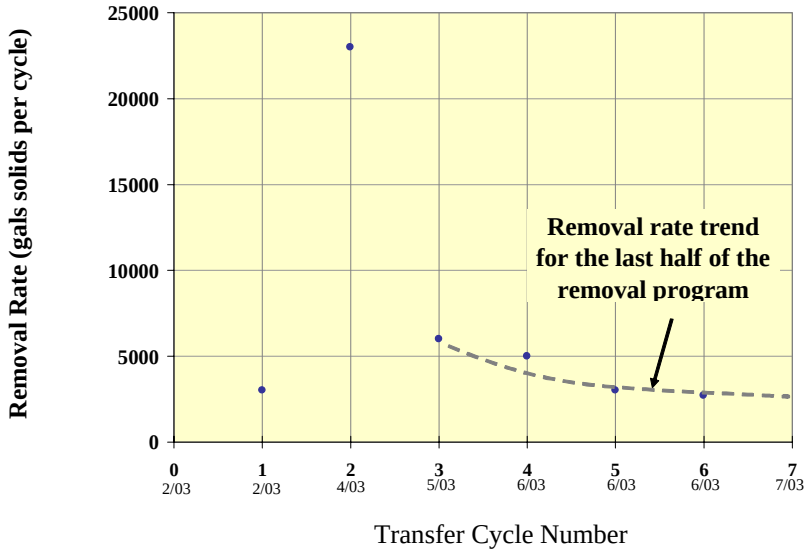
As discussed in Section 2.4, a number of obstacles prevented further residual waste removal. The obstacles fell into four categories: (1) fast-settling zeolite resins, (2) general tank access limitations and obstructions, and (3) mounds of insolubles. In addition to those obstacles, FFA requirements for closure of Tank 19 and Tank 18, as well as tank space constraints were additional factors that prevented further residual waste removal. Removal of highly radioactive radionuclides, for the purpose of this Determination, is discussed in Section 2.5.1. Risk-benefit of additional removal of radionuclides is discussed in Section 5.3.

Tank 18 Waste Removal

As discussed in Subsection 2.4.2, bulk waste removal efforts removed approximately 850 kgal of liquid. After the liquid waste was removed, a sludge formation with an estimated volume of approximately 551 kgal remained. Bulk waste removal efforts removed approximately 515 kgal of sludge.

Following the completion of bulk waste removal the ADMP was selected as the preferred technology for heel removal (as discussed in Subsection 2.4.2.4.1). [46] The heel removal campaign removed a large portion of the heel. [68] After washing the residual material with approximately 800 kgal of water, Cs-137 was no longer being washed out. As discussed in Section 2.4, the remaining chemically bound Cs-137 that could not be removed by these techniques will likewise not be readily mobilized by water infiltration after closure. Waste removal activities continued until no longer effective. The heel removal rate declined as the transfer cycles progressed until the heel removal campaign was stopped (Figure 5-2). [49] Of the entire 1,300 kgal of material in Tank 18, the overall waste removal program removed all but 4.3 kgal of material, of which approximately 10 gallons are radioactive isotopes. [51]

Figure 5-2: Tank 18 Heel Removal Rate [49]



As discussed in Section 2.4, a number of obstacles prevented further residual waste removal. The obstacles fell into four categories: (1) fast-settling zeolite resins, (2) general tank access limitations and obstructions, (3) hardened mounds of insolubles; and 4) unexpected material flow characteristics that adversely impacted ADMP performance. In addition to those obstacles, FFA requirements for closure of Tank 18, as well as tank space constraints, were additional factors that prevented further residual waste removal. Removal of high radioactive radionuclides is discussed in Section 2.5.1. Risk-benefit of additional removal of radionuclides is discussed in Section 5.3.

The cleaning processes for Tank 19 and Tank 18 were frequently reviewed during cleaning activities to maximize effectiveness. Detailed camera inspections of the tank contents used features and mounds within the tank to determine remaining heel volumes and locations of exposed solids after cleaning cycles. Engineers evaluated mixer operating strategies and transfer pump operation from previous cleaning cycles and the resulting heel removal rates and relocation of exposed solids remaining in the tanks. These evaluations resulted in new mixing/transfer strategies to optimize mixer orientation, mixer speed, and transfer pump operation to maximize effectiveness of radionuclide removal during subsequent cleaning cycles. [85],[86]

As discussed in the following subsections, further removal of this residual material is not practical, and any efforts to remove more of the small quantity of remaining residual material would not significantly reduce the potential risk to the public. Various cleaning technologies were considered for heel removal in Tank 19 and Tank 18 with the goal of protecting public and occupational health and safety at the tank closure site. Technologies were evaluated by comparing information on effectiveness, maturity, and cost balanced against minimizing occupational radiation dose. As shown in Appendix C, potential cleaning technologies were identified in coordination with the DOE TFA. The TFA was commissioned by the DOE to coordinate technology needs for DOE complex-wide radioactive tank waste remediation problems. As discussed in Subsection 2.4, the technology selection teams for Tank 19 and Tank 18 considered chemical treatment, mechanical agitation equipment, and mechanical retrieval equipment for cleaning tanks. As discussed in Section 2.4.2.4.1.1, DOE determined that chemical processes were not practical because of down stream process impacts and authorization basis impacts. [36],[46] The selected tank cleaning technologies for Tank 19 and Tank 18 were tested at SRS and other facilities (both partial and full scale).

5.2.1 Percentage of Radionuclides Removed

To quantitatively address how effective the waste removal techniques have been, a mass balance approach was developed. As discussed in Subsection 2.3, Tank 19 and Tank 18 had various sources of waste including: (1) waste streams from the F-Canyon PUREX process, (2) concentrated PUREX waste streams from the 242-F Evaporator, and (3) zeolite resin from the 242-F evaporation process. The total inventory of radionuclides in Tank 19 and Tank 18 prior to waste removal has been identified. Evaluation of historical waste transfer records, the WCS (discussed in Appendix D), and other special analyses were used to arrive at the total radioactivity present in Tank 19 and Tank 18 prior to waste removal. Reference dates prior to waste removal are 1980 and 1986 for Tank 19 and Tank 18, respectively. Prior to waste

removal, Tank 19 contained approximately 2.2 million curies, and Tank 18 contained approximately 2.5 million curies. [11],[87]

Table 5-2 shows the initial inventory, as of 1980, of waste in Tank 19 (decayed to 2020), and the percentage of radioactivity removed by waste removal operations. Table 5-3 shows the initial inventory, as of 1986, of waste in Tank 18 (decayed to 2020), and the percentage of radioactivity removed by waste removal operations. The percentages are determined by comparing the inventory of the radioactivity (Ci) that was present prior to waste removal (decayed to 2020) and the estimated residual waste inventory at closure (decayed to 2020). The residual waste inventory at closure is based on the characterization process discussed in Subsection 2.5.

Table 5-2: Percentage of Highly Radioactive Radionuclides Removed from Tank 19

Isotope	Inventory Prior to Waste Removal (Ci)[a]	Residual in Tank at Closure (Ci)[b]	% Removed at Closure
Am-241	3.24E+02	1.24E+01	96.17%
Am-242m	6.00E+01	1.77E+00	97.05%
Am-243	1.37E-04	1.75E-07	99.87%
C-14	2.81E+01	1.12E-01	99.60%
Cm-242	4.94E+01	1.45E+00	97.06%
Cm-244	1.39E-02	1.53E-03	88.99%
Cm-245	2.15E-08	1.97E-09	90.84%
Cm-246	1.60E-10	2.01E-13	99.87%
Cm-247	1.40E-15	6.16E-20	99.99%
Cm-248	1.07E-17	1.42E-20	99.87%
Co-60	3.95E+00	6.82E-01	82.73%
Cs-137[c]	2.58E+05	3.25E+04	87.40%
Ba-137m[c]	2.44E+05	3.07E+04	87.42%
H-3	2.20E+01	1.37E-01	99.38%
I-129	5.73E-02	4.95E-05	99.91%
Nb-94	3.93E-04	3.63E-05	90.76%
Ni-59	8.95E+00	8.26E-01	90.77%
Ni-63	7.06E+02	6.73E+01	90.47%
Np-237[c]	3.05E-01	1.13E-02	96.30%
Pu-238[c]	3.86E+03	2.14E+01	99.45%
Pu-239	7.40E+02	3.12E+01	95.78%
Pu-240	1.62E+02	1.11E+01	93.15%
Pu-242	8.81E-02	3.72E-01	[e]
Pu-244	3.99E-04	1.66e-03	[e]
Pu-241	1.03E+03	2.30E+01	97.77%
Sr-90[c]	1.64E+05	2.57E+01	99.98%
Y-90[c]	1.64E+05	2.57E+01	99.98%
Tc-99[c]	4.46E+02	6.76E+00	98.48%
Cf-249	2.93E-19	3.82E-22	99.87%
Total Ci[d]	8.36E+05	6.34E+04	92.43%

[a] Inventory values prior to waste removal are 1980 curie values decayed to 2020.

[b] Total Ci at closure includes Ba-137m, Y-90, decayed to 2020. As discussed in Section 2.5, inventories are based on: (1) heel residuals that are estimated using remote video inspection of cleaned tank internals to map out estimates of depth of remaining residual solids and liquids across tank bottoms using tank internal reference points (features) of known height and liquid levels based on instrumentation (reel tape) readings, (2) best estimated radionuclide concentrations from past and recent closure samples, WCS, and special analysis [5] , and (3) radioactive decay to 2020. [59]

[c] Radionuclides that are significant contributors to dose calculations in the PODD. [4]

[d] Radionuclides shown are contributors to the dose calculations or regulated by concentration limits in 10 CFR 61.55. The totals are based on the entire inventory of radionuclides.

[e] The Pu-242 and Pu-244 percentage (%) removed values are not valid due to use of minimum detection limits by the laboratory analysis for Pu-242. The Pu-242 concentrations for each Tank 19 sample were below the laboratory analytical minimum detection limits. Therefore, less than detectable values were used to estimate the Pu-242 inventory in the Tank 19 residual material, which drives the estimated inventory higher. This higher value for Pu-242 impacts the percentage removed value when compared to the Pu-242 inventory estimated in Tank 19 prior to waste removal. [11],[59] The Pu-244 inventory is based on the Pu-242 inventory in the Tank 19 residual material, and is therefore also a higher estimate of residual inventory. [55] Pu-242 and Pu-244 make an insignificant contribution to dose. Additionally, one would expect these isotopes to have been removed at the same rate as other Pu isotopes.

Table 5-3: Percentage of Highly Radioactive Radionuclides Removed from Tank 18

Isotope	Inventory Prior to Waste Removal (Ci)[a]	Residual in Tank at Closure (Ci)[b]	% Removed at Closure
Am-241	1.38E+03	7.68E+01	94.43%
Am-242m	8.80E+01	7.39E-01	99.16%
Am-243	6.96E-04	6.14E-06	99.12%
C-14	5.83E+00	5.60E-02	99.04%
Cm-242	7.24E+01	6.08E-01	99.16%
Cm-244	3.18E+03	8.83E+01	97.22%
Cm-245	8.64E-08	2.17E-09	97.49%
Cm-246	8.12E-10	7.09E-12	99.13%
Cm-247	7.09E-15	2.17E-18	99.97%
Cm-248	5.43E-17	5.02E-19	99.08%
Co-60	2.32E+01	4.65E-01	98.00%
Cs-137[c]	4.35E+05	8.52E+03	98.04%
Ba-137m[c]	4.11E+05	8.05E+03	98.04%
H-3	4.41E+01	2.37E-02	99.95%
I-129	9.88E-02	6.42E-06	99.99%
Nb-94	1.59E-03	4.01E-05	97.48%
Ni-59	3.61E+01	9.12E-01	97.47%
Ni-63	2.88E+03	7.16E+01	97.51%
Np-237[c]	2.41E+00	1.19E-01	95.06%
Pu-238[c]	1.33E+03	6.14E+01	95.38%
Pu-239	3.24E+03	1.40E+02	95.68%
Pu-240	8.46E+02	5.81E+01	93.13%
Pu-242	9.65E-01	5.60E-01	[e]
Pu-244	4.40E-03	3.39E-04	[e]
Pu-241	4.45E+03	1.11E+02	97.51%
Sr-90[c]	9.39E+04	9.28E+02	99.01%
Y-90[c]	9.40E+04	9.28E+02	99.01%
Tc-99[c]	2.35E+02	1.48E+00	99.37%
Cf-249	1.45E-18	1.34E-20	99.08%
Total Ci[d]	1.05E+06	1.90E+04	98.19%

[a] Inventory prior to waste removal are 1986 curie values decayed to 2020.

[b] Total Ci at closure includes Ba-137m and Y-90 and decayed to 2020. As discussed in Section 2.5, inventories are based on: (1) heel residuals that are estimated using remote video inspection of cleaned tank internals to map out estimates of depth of remaining residual solids and liquids across tank bottoms using tank internal reference points (features) of known height and liquid levels based on instrumentation (reel tape) readings, (2) best estimated radionuclide concentrations from past and recent closure samples, WCS, and special analysis [6], and (3) radioactive decay to 2020. [59]

[c] Radionuclides that are significant contributors to dose calculations in the PODD [4]

[d] Radionuclides shown are contributors to the dose calculations or regulated by concentration limits in 10 CFR 61.55. The totals are based on the entire inventory of radionuclides.

[e] The Pu-242 and Pu-244 percentage (%) removed values are not valid because of the Pu-242 curies that were estimated to be in the Tank 18 corrosion products. A large percentage (86%) of the Pu-242 residual inventory in the Tank 18 residual material is attributed to corrosion products. [6] The Pu-242 inventory associated with corrosion products is not included in the estimation of Pu-242 curies in Tank 18 prior to waste removal. [87] The Pu-244 inventory is based on the Pu-242 inventory in the Tank 18 residual material. [64] Pu-242 and Pu-244 make an insignificant contribution to dose. Additionally, one would expect these isotopes to have been removed at the same rate as other Pu isotopes.

The values in Table 5-2 and Table 5-3 have been calculated by decaying the radionuclides to 2020, which is the projected end of closure activities for FTF. However, for perspective, if the inventory were to be calculated by decaying the radionuclides to the present (2005), the total Ci remaining at closure would be somewhat higher, on the order of 96,000 Ci for Tank 19 and 28,000 Ci for Tank 18. The change is caused by Cs-137 and Sr-90 and their daughter products since these radionuclides account for the majority of the radioactivity, and they decay relatively quickly. However, if tables similar to Table 5-2 and Table 5-3 were constructed to show percent removed values for current (2005) decay dates, the percent removed values would remain the same, because the relatively short-lived radionuclides would also increase in the tank inventory at the same rate as the inventory prior to waste removal based on the half-life of each radionuclide.

Of the approximately 2.2 million Ci in 1.3 kgal of waste stored in Tank 19 prior to waste removal (1980), approximately 96,000 Ci are estimated to remain in the tank following waste removal (2002) representing about 4% of the tank waste inventory prior to waste removal. [11],[5] Approximately 99% of the total tank waste residual inventory[13] at closure is represented by Cs-137/Ba-137m bound to the zeolite (i.e., the cesium/barium atoms captured by the zeolite resin are chemically part of the zeolite molecule). The half-life of Cs-137 is 30.17 years.

Of the approximately 2.5 million Ci in 1.3 kgal of waste stored in Tank 18 prior to waste removal (1986), approximately 28,000 Ci are estimated to remain in the tank following waste removal (2003), representing about 1% of the tank waste inventory prior to waste removal. The calculation of 1% of the inventory is conservative, because it does not account for additions of radionuclides to Tank 18 from Tank 17 and Tank 19 heel removal, which would reduce the percentage removed even further. [6],[87] Approximately 86% of the total tank waste residual inventory[14] is represented by Cs-137/Ba-137m bound to the zeolite, and approximately 10% of the inventory[14] is represented by Sr-90/Y-90.

The remaining curies in Tank 19 and Tank 18 will remain in a solidified waste form. The solidified waste form will provide a long-term chemical environment to minimize leaching of most radionuclides from the residual waste.

5.3 Risk Benefit Analyses for Further Residual Waste Removal

A risk-benefit analysis[15] has been conducted to evaluate the financial costs and radiological risks associated with further removal of the residual heel in Tank 19 and Tank 18. The financial costs and radiological risks are compared to the potential benefit of reducing the dose to an individual from a projected 50-year period of an “all pathways” exposure. [88] F-Area Waste Tank 19 and Tank 18 have undergone extensive cleaning efforts to the extent that only a residual amount of radioactive material remains. Three methods for further residual waste removal have been identified for evaluation of the risk benefit for Tank 19 and Tank 18. These methods were

[13] The percentage of the total tank waste residual inventory represented by Cs-137/Ba-137m bound to the zeolite is calculated using the curie inventory after heel removal as of 2002.
[14] The percentage of the total tank waste inventory represented by Cs-37/Ba-137m bound to the zeolite is calculated using the curie inventory after heel removal as of 2003.
[15] DOE is including this risk benefit comparison based on recommendations during consultation with the NRC.

chosen for evaluation based on operational history within SRS and the DOE complex, and include (1) Mechanical – using new mixing pumps and new transfer pumps, (2) Robotic – using remote controlled cleaning equipment and a transfer system, and (3) Chemical – using OA to dissolve some of the residual waste and a transfer system. [88] Even though chemical cleaning was deemed "not practical" at the time for heel removal efforts for Tank 19 and Tank 18 (as discussed in Section 2.4), chemical cleaning was evaluated because it will be considered for further residual waste removal.

If possible, further removal of radioactive wastes inside the tanks would eliminate any potential risk from radiological exposure to the public after tank closure and after institutional control of the tank farm. However, there are no technologies available that provide complete removal of the radioactive waste. Also, a proposed residual heel removal step has financial costs to DOE as well as radiological exposure risks to DOE workers. Thus, risk/benefit analyses were performed to quantify the costs and the radiological risks and to compare them to the potential benefit, and are summarized below. [88]

5.3.1 Residual Heel Removal Methods

The residual waste removal methods described below and their associated costs, radiological risks and schedule durations are presented for evaluation purposes only and do not represent detailed plans that have been accepted by either the DOE or by the SRS facilities involved.

The scope of work associated with residual waste removal includes: (1) technical evaluation and selection of equipment specific to the residual heel removal method, 2) removal of failed equipment to provide access to the tank, (3) disposition (transport, decontamination, and disposal) of removed equipment, (4) drilling, if deemed necessary, new riser(s) in the tank top for access to facilitate residual heel removal, (5) develop design for modifications of existing tank systems and installation of new equipment, (6) perform physical modifications and install new equipment, (7) develop new procedures, (8) perform residual heel removal operations, (9) sample and analyze residual material, and (10) perform fate and transport modeling for the performance evaluation. [88]

5.3.1.1 <u>Mechanical Removal</u>

This method is similar to the process used in Phase-2, Step-3, for Tank 19 (see Section 2.4.1.4). Three mixing pumps would be used continuously to thoroughly agitate the contents of the tank solids in the subject tank with the transfer medium liquid. This well-mixed slurry would be transferred to a receipt tank via a TTP. A transfer pump in the receipt tank would return the transfer medium liquid to the subject tank – forming a recirculation loop. This recirculation process would be continuous (compared to the batch process used in Tank 19) to provide more suspension of the fast settling particles. The discharge into the receipt tank would be strategically placed with respect to the location of the transfer pump in the receipt tank to allow for maximum settling time of the suspended particles before the transfer fluid is pumped out and returned to the subject tank. This constant recirculation process would continue until additional removal of the residual heel material can no longer be realized. Even with this continuous recirculation process, it is expected some of the residual waste containing the fast settling zeolite would be left in the tank. [88]

The estimated project costs are $10.2 M for each tank. The estimated schedule duration for this method is approximately two years for each tank. The additional radiological risk to DOE personnel for this method is estimated to be ~7,500 mrem cumulative worker dose for workers involved in the implementation of this method for each tank. [88] The potential benefit of mechanical removal is a potential 50-year dose savings to an individual of 2 mrem for Tank 18, and 0.45 mrem for Tank 19.

5.3.1.2 Robotic Removal

This method is based on a system utilized at the Oak Ridge National Laboratory (ORNL) for the Gunite and Associated Tanks (GAAT) Remediation Project. The major pieces of equipment associated with this method are Confined Sluicing End Effectors (CSEEs) and a remote controlled crawler. The CSEE is a rotating water-jet cutter equipped with a vacuum head. The CSEE would be used to dislodge and mobilize sludge mounds. The crawler is a robust, tracked work platform equipped with a plow blade, spray nozzles, camera, and lights. At SRS, the crawler would be deployed from the central riser which would house a containment structure that includes maintenance systems and tether management systems for the required hydraulic hoses, water line, and electrical power and control cables. During residual waste removal operations, hydro-lancing would be performed as necessary, and the crawler plow blade would push residual solids to one of four CSEEs (one in each tank quadrant) where solids are picked up and delivered into the transfer system. [88]

The estimated project costs are $14.6 M for each tank. The estimated schedule duration for this method is approximately two years for one tank (which includes an estimated eight months for the design, development, testing and training on the use of the robotic equipment). The radiological risk to DOE personnel for this method is estimated to be ~7,500 mrem cumulative worker dose for workers involved in the implementation of this method for each tank. [88] The potential benefit of robotic removal is a potential 50-year dose savings to an individual of 2 mrem for Tank 18, and 0.45 mrem for Tank 19.

5.3.1.3 Chemical Removal

To implement this removal method several activities must be completed to update the DSA and Technical Specification Requirement (TSRs). These activities include performing accident analyses, conducting Nuclear Criticality Safety Evaluations, and performing corrosion testing and developing hydrogen generation information. If the DSA and TSRs can be revised to identify the appropriate safety controls, implementation of this method would entail the placement of two 150-hp hydraulic mixers and a transfer pump, as well as a hydro-lance to break up larger pieces of sludge to make the acid cleaning more effective. The necessary quantity of OA would be added to the tank and the tank contents would be agitated to improve the contact of the OA with the residual waste. After some dissolution of the waste has occurred, the waste would be transferred to a designated neutralization tank. This process would be repeated to reduce the residual waste volume. [88]

The estimated project costs are $11.1 M for each tank. The estimated schedule duration for this method is approximately one year per tank, in addition to an estimated eight months for the completion of safety analyses supporting the use of OA. The radiological risk to DOE personnel for this method is estimated to be ~4,500 mrem cumulative worker dose for workers involved in

the implementation of this method for one tank. The potential benefit of chemical removal is a potential 50-year dose savings to an individual of 2 mrem for Tank 18, and .045 mrem for Tank 19. [88]

5.3.2 Comparative Analysis

The financial costs and radiological risks to SRS personnel have been quantified to develop a risk benefit analyses and are presented in Table 5-4. However, there are other factors that may not be as easily quantifiable but are nevertheless important considerations when evaluating removal options. Other considerations include: (1) likelihood of success, (2) technical risk, and (3) facility impact. The "likelihood of success" is the probability a given method will remove the majority of the residual radioactivity. All three methods are judged to have a reasonable probability of success. The "technical risk" is a combination of other risks associated with a given method with respect to safety analysis and its impacts, integration with SRS systems, and operational experience. Because of its extensive use at SRS, the mechanical method has limited technical risk while the other methods have significant technical risk. These technical risks include SRS integration and operational experience risks for the robotic method and safety analysis risk (criticality and hydrogen controls) for the chemical removal method. The "facility impact" considers any adverse downstream impacts a given method may have on SRS systems. The mechanical and robotic method has significant facility manpower impact concerns, and the chemical cleaning method has significant potential facility impact concerns based on the current lack of liquid storage space in the tank farm and processing concerns with the Tank Farm Evaporators and the DWPF. [88]

The closure of the waste tanks provide a source of radioactive contamination to the environment over a long period of time. Such contamination to the environment has the potential to cause radiological exposure to individuals that would live in the area. To determine the potential radiological exposure, an "all-pathways" dose has been calculated which considers various pathways of radiological exposure. These pathways include ingestion of contaminated food, milk and water, and external exposures. [4]

The potential annual "all-pathways" dose attributed to Tank 19 is 0.009 mrem with a 50-year period exposure of 0.45 mrem (= 50 x 0.009 mrem). [4] The potential annual "all pathways" dose attributed to Tank 18 is 0.04 mrem with a 50-year period exposure of 2 mrem (= 50 x 0.04 mrem). [4]

Table 5-4 summarizes the results of the risk benefit analyses for the further removal of residual waste. [88]

Table 5-4: Considerations for Further Residual Waste Removal [88]

Residual Heel Removal Method	Financial Cost ($)[1]	Cumulative Worker Risk[2] (mrem)	Potential 50-Year Dose Savings to an Individual[3] (mrem)		Financial Cost – to – Potential Benefit Ratio[4] ($ / mrem)[2]		Worker Risk – to – Potential Benefit Ratio (present mrem / future mrem)[5]	
	Per Tank	Per Tank	Tank 18	Tank 19	Tank 18	Tank 19	Tank 18	Tank 19
Mechanical	10,200,000	7,500	2	0.45	5,100,000	22,700,000	3,750	16,700
Robotic	14,600,000	7,500	2	0.45	7,300,000	32,400,000	3,750	16,700
Chemical	11,100,000	4,500	2	0.45	5,550,000	24,700,000	2,250	10,000

1 Project cost in dollars, rounded to $100,000.
2 Risk from waste removal part of the cleaning operations only. Does not include radiological exposure associated with disposal.
3. Future hypothetical member of public.
4 Dollars spent for one future potential mrem exposure saved.
5 Present day radiological exposure received by workers for one future potential mrem exposure saved.

5.3.3 Conclusions

The potential dose savings resulting from the complete removal of radioactivity in either Tank 19 or Tank 18 are not significant when compared to the worker exposure risk and the financial cost that would result from such an effort. As illustrated in Table 5-4, the present day risk to workers from radiological exposure is 2,250 times to 16,700 times greater than the risk to an individual from a potential future 50-year exposure. The financial cost of residual heel removal ($10,200,000 to $14,600,000) to potentially save one millirem of a potential 50-year exposure in the future ranges from $5,100,000 to $32,400,000 of project cost.

6.0 RADIONUCLIDE CONCENTRATIONS OF STABILIZED WASTE

Key Points

- The solidified waste form will prevent inadvertent intrusion, minimize free standing liquid, and minimize void spaces (see Section 6.1).
- All components of the residual waste (e.g. tank heel, tank bottom and walls, tank dome, tank stiffener bands, corrosion products on the tank walls, tank equipment and airborne particulate waste inside the tank) will be incorporated into and stabilized by the grouted waste form that will fill the entire tank interior.
- Calculations assume that the FTF will be closed in or after 2020, as planned.
- The final waste forms in the tanks will contain short-lived and long-lived radionuclides.
- Concentrations of waste solidified in grout are calculated for the closed tanks and shown along with the Class C concentration limits in 10 CFR 61.55.
- The residual waste remaining in Tank 19 is approximately 96,000 Ci (in 2002).
- The residual waste remaining in Tank 18 is approximately 28,000 Ci (in 2003).

Section 3116 of the NDAA provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy …, in consultation with the Nuclear Regulatory Commission …, determines— …
>
> > (3)(A) does not exceed concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, and will be disposed of—
> >
> > > (i) in compliance with the performance objectives set out in subpart C of Part 61 of title 10, Code of Federal Regulations; and
> > >
> > > (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; or
> >
> > (3)(B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will be disposed of—
> >
> > > (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations, and
> > >
> > > (ii) pursuant to a State-approved closure plan or State-issued permit, authority for the approval or issuance of which is conferred on the State outside of this section; and

(iii) pursuant to plans developed by the Secretary in consultation with the Commission.

Section 3116(a)(3) provides that, regardless of whether the waste exceeds or does not exceed the concentration limits for Class C LLW as set out in 10 CFR 61.55, the Secretary must determine that the waste will be disposed of in compliance with the performance objectives of 10 CFR 61, Subpart C and that the waste will be disposed of in accordance with State-approved closure plans. In Section 7 of this document, information is presented that demonstrates that the waste will be disposed of in compliance with the performance objectives of 10 CFR 61, Subpart C. In Section 8 of this document, information is presented that demonstrates that waste will be disposed of in accordance with State-approved closure plans.

In situations where the waste is determined to exceed the concentration limits for Class C LLW, Section 3116(a)(3)(B)(iii) provides for consultations with NRC about the disposal plans for the waste. The DOE is not deciding in this draft 3116 Determination whether the waste does or does not exceed the concentration limits for Class C since there is no clearly applicable NRC guidance on applying the concentration limits in 10 CFR 61.55 to situations like Tank 19 and Tank 18. Nevertheless, DOE intends to take full advantage of the consultation process established by Section 3116 and is requesting that NRC identify what changes, if any, it would recommend to DOE's disposal plans (i.e., the actions DOE intends to undertake to dispose of the waste, as described in this draft 3116 Determination) to meet the criteria of Section 3116.

In order to fully comply with the provisions of Section 3116 and to facilitate the consultations with NRC, this section discusses radionuclide concentrations of wastes solidified in grout in Tanks 19 and 18. This discussion is structured in a manner that permits a comparison of various levels of solidified waste.

In the following subsection, the waste concentrations for specified radionuclides are shown along with the Class C concentration limits defined in 10 CFR 61.55. The radionuclides and their associated limits are specified in two separate tables within 10 CFR 61.55 which are reproduced below in Table 6-1 and Table 6-2.

Table 6-1: 10 CFR 61.55 Table 1 [79]

Radionuclides (Long-lived)	Concentration (Ci/m^3)
C-14	8
C-14 in activated metal	80
Ni-59 in activated metal	220
Nb-94 in activated metal	0.2
Tc-99	3
I-129	0.08
Alpha Emitting Transuranic (TRU) nuclides with half-life greater than five years	100[(1)]
Pu-241	3,500[(1)]
Cm-242	20,000[(1)]

[1] Units are in nanocuries per gram.

Table 6-2: 10 CFR 61.55 Table 2 [79]

Radionuclides (Short-lived)	Concentration, curies per cubic meter		
	Column 1 [Class A]	Column 2 [Class B]	Column 3 [Class C]
Total of all nuclides with less than 5-year half-life	700	*(1)*	*(1)*
H-3	40	*(1)*	*(1)*
Co-60	700	*(1)*	*(1)*
Ni-63	3.5	70	700
Ni-63 in activated metal	35	700	7000
Sr-90	0.04	150	7000
Cs-137	1	44	4600

[1] There are no limits established for these radionuclides in Class B or C wastes. Practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations for these wastes. These wastes shall be Class B unless the concentrations of other nuclides in the table determine the waste to be Class C independent of these nuclides.

6.1 Waste Concentrations in Grouted Tanks 19 and 18

In Tank 19 and Tank 18, residual radioactive waste will be converted into a stabilized waste form. During grouting operations, the tanks will be filled with grout to stabilize the contaminated tank heel, tank bottom and walls, tank stiffener bands, corrosion products on the tank walls, tank process equipment, airborne particulate waste inside the tank, and the tank dome and risers.

The residual radioactive material includes the tank heel, material on the stiffener bands, wall corrosion products, contaminated concrete tank dome, abandoned equipment, and airborne particulates. During stabilization of the residual material through the addition of grout, radioactive material will become a part of the final waste form during the entire filling process. The residual material on the bottom, walls, and dome of the tank and on the wall stiffener bands will be stabilized through the addition of grout. The tank includes used process equipment entering the tank through top riser openings and occupying the full height of the tank (e.g., mixer pumps, transfer pumps, etc.). This contaminated equipment and the tank risers and dome will be stabilized during grout fill. [76]

Particulate material will be incorporated into the waste form during the stabilization process. In addition, radioactive material present in the corrosion products on the tank walls may be partially moved from the walls by contact with the stabilizing grout. [61] The corrosion products moved during the grouting process will mix with the flowable grout and further result in a final waste form incorporating radioactive material at all levels. In general, the grout formulation testing, grout delivery plan, and the grout delivery field procedures assure stability of the solidified mass. [71],[76]

The reducing grout used to stabilize contamination will not be poured above the springline. In addition to the layer of reducing grout, the tanks will be filled to the top of the domes with strong grout to prevent inadvertent intrusion into the residual waste material by making the solidified tank contents recognizable as a structure with distinctly different physical properties than the surrounding strata (see Section 2.6). The strong grout will also provide for long-term stability of the tank structure.

In addition, any residual liquid is incorporated into the grouted waste form by absorption during fresh grout pouring and by application of additional dry grout as necessary during and after the pouring process. [76]

The residual waste remaining in Tank 19 proposed for in-situ disposal is approximately 96,000 Ci (in 2002) as documented in the Tank 19 waste characterization report [5] and presented in Section 2.5.1.3. The residual waste remaining in Tank 18 proposed for in-situ disposal is approximately 28,000 Ci (in 2003) as documented in the Tank 18 waste characterization report [6] and presented in Section 2.5.2.3.

In general, the grout formulation testing, grout delivery plan, and the grout delivery field procedures assure stability of the solidified mass. The volume of grout needed to fill the tank is approximately 6,300 m^3 with a density of 1.94 g/cm^3. [5]

The final waste in Tank 19 and Tank 18 will contain short-lived and long-lived radionuclides from Tables 6-1 and 6-2. Therefore, the waste concentration limits will be determined in accordance with 10 CFR 61.55(a)(5). As provided in 10 CFR 61.55(a)(5), the radionuclide limits shown in Table 6-1 above (10 CFR 61.55 Table 1) and radionuclide limits shown in Column 3 of Table 6-2 above (10 CFR 61.55 Table 2) are applicable.

Figures 6-1 and 6-2 show the grout-filled tanks, the volume of grout and the Sum-of-Fractions for the closed tanks as calculated in the Tank 19 and Tank 18 waste characterization reports. [5],[6],[91] The following three scenarios, were calculated to show what the results of the 10 CFR 61.55(a)(5) determination would be if different amounts of grout were assumed to be included in the waste form. All three scenarios included the volume or mass of the carbon steel liner, concrete tank walls, and concrete dome:

- Scenario 1, calculation to achieve a sum of fractions of 1.00 or less (Class C),
- Scenario 2, calculation to include volume (or mass) of waste and grout to the springline (top of the carbon steel liner), and
- Scenario 3, calculation to include volume (or mass) of waste and grout to the top of the tank for stabilization.

In the future, DOE will reassess the portions of the tank components that are credited in the calculations to determine concentrations. This reassessment would evaluate whether accounting for the full height of the concrete walls would significantly modify the results.

Figure 6-1: Tank 19 Radionuclide Concentrations[16] [5],[91]

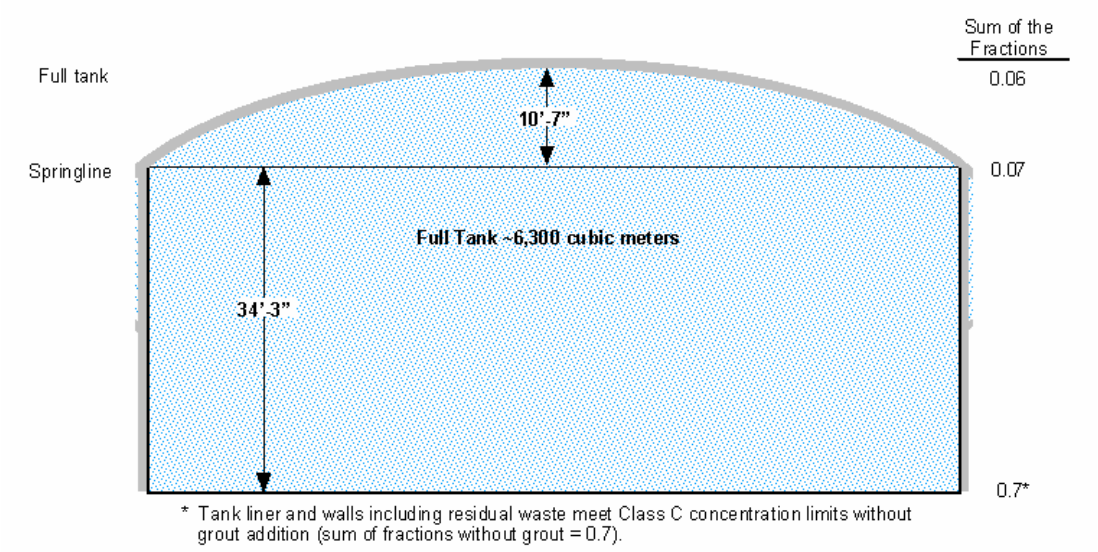
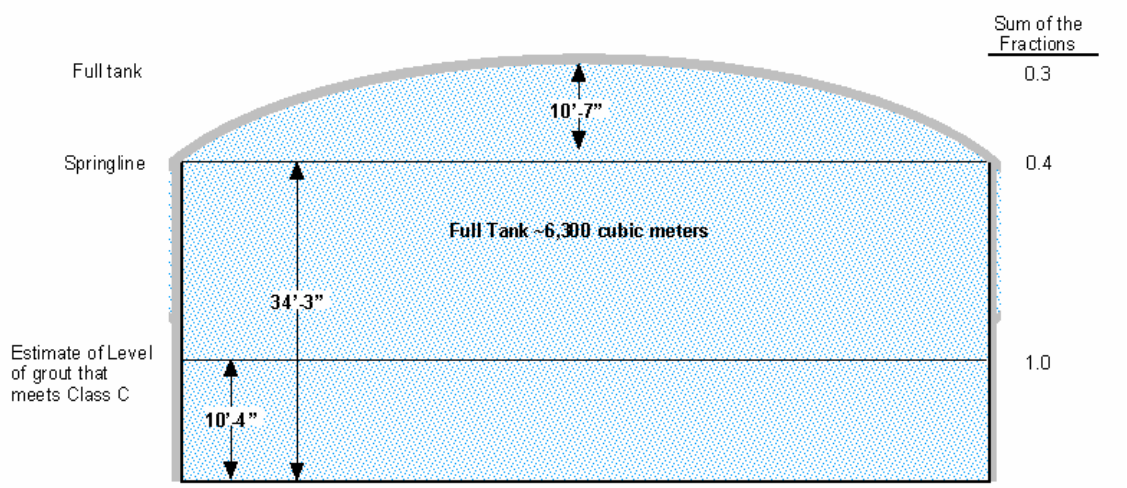


Figure 6-2: Tank 18 Radionuclide Concentrations[17] [6],[91]



[16] The radionuclide concentrations used in the sum of fraction calculations use the radionuclide inventory divided by the volume (or mass) in the assumed waste form, including the carbon steel liner, concrete walls, and concrete dome.

[17] The radionuclide concentrations used in the sum of fraction calculations use the radionuclide inventory divided by the volume (or mass) in the assumed waste form, including the carbon steel liner, concrete walls, and concrete dome.

7.0 THE WASTE WILL BE DISPOSED OF IN ACCORDANCE WITH THE PERFORMANCE OBJECTIVES SET OUT IN 10 CFR 61, SUBPART C

Section Purpose

The purpose of this section is to demonstrate that the closed tanks will achieve the performance objectives for land disposal of low-level waste found in 10 CFR 61, Subpart C.

Section Contents

Section 7.1 identifies the performance objectives. Section 7.2 describes the evaluations performed and their results.

Key Points

- Potential doses were evaluated for a period beginning 100 years after tank closure and extending to 10,000 years after tank closure. It was assumed for modeling purposes only, that the tank top, grout, and basemat would hydraulically fail 500 years after closure.
- The closest location where a member of the public could credibly be exposed is located where groundwater from the upper groundwater aquifers seeps to the surface near Fourmile Branch, located near the center of the site, about one mile from the tanks.
- For Tank 19, the estimated maximum annual all-pathways dose to a member of the public is 0.009 mrem whole-body, compared to the limit of 25 mrem/yr associated with 10 CFR 61.41.
- For Tank 18, the estimated maximum annual all-pathways dose to a member of the public is 0.04 mrem whole-body, compared to same limit of 25 mrem/yr associated with 10 CFR 61.41.
- The estimated maximum annual dose to a hypothetical inadvertent intruder is 120 mrem/yr, assuring compliance with 10 CFR 61.42.
- Demonstrate compliance with 10 CFR 61.43 and 10 CFR 61.44.

Section 3116 of the NDAA criterion provides in relevant part:

> [T]he term "high-level radioactive waste" does not include waste from reprocessing of spent nuclear fuel that the Secretary of Energy…in consultation with the Nuclear Regulatory Commission…determines -
>
> ***
>
> (3)(A) [W]ill be disposed of -
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations[.]

7.1 10 CFR Part 61 Performance Objectives [79]

Subpart C of 10 CFR Part 61 lists one general requirement and four performance objectives, which are reproduced below.

7.1.1 Section 61.40, General Requirement [79]

Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in Sections 61.41 through 61.44.

7.1.2 Section 61.41, Protection of the General Population from Releases of Radioactivity [79]

Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as reasonably achievable (ALARA).

7.1.3 Section 61.42, Protection of Individuals from Inadvertent Intrusion [79]

Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed.

7.1.4 Section 61.43, Protection of Individuals During Operations [79]

Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in Part 20 of this chapter, except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by Section 61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures ALARA.

7.1.5 Section 61.44, Stability of the Disposal Site After Closure [79]

The disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required.

7.2 10 CFR Part 61 General Requirements

7.2.1 Section 61.40, General Requirement [79]

The general provision at 10 CFR 61.40 calls for “reasonable assurance” that exposures are within the limits of the four subsequent performance objectives - 10 CFR 61.41, 10 CFR 61.42, 10 CFR 61.43, and 10 CFR 61.44 - as discussed below. The general provision of 10 CFR 61.40 is addressed within each ensuring discussion of these four objectives.

The Tank 19 and Tank 18 PODD details the analysis performed to provide “reasonable assurance” that the performance objectives are not exceeded as a result of operation and closure of the tanks. The PODD meets the performance objective of “reasonable assurance” through the use of conservative estimates for parameter values, concrete/grout failure times, containment

transport pathways and human activity models. In addition, sensitivity and uncertainty analyses are used to ensure that a conservative estimate was conducted for the closure of Tank 19 and Tank 18.

7.2.2 Section 61.41, Protection Of The General Population from Releases of Radioactivity **[79]**

10 CFR 61.41 states:

> Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment ALARA.

PRESENTATION OF INFORMATION IN THIS SECTION

For clarity, information related to protection of the general population on releases of radioactivity from Tank 19 and Tank 18 is presented in the following sections: Section 7.2.2.1 provides a general description of the analysis approach used, Section 7.2.2.2 provides a description of the conceptual model used to represent the closed tanks and the exposure pathways evaluated, Section 7.2.2.3 details the fate and transport mathematical model used to represent the conceptual model, Section 7.2.2.4 details key assumptions and modeling input parameters, Section 7.2.2.5 describes the results of the analyses, Section 7.2.2.6 describes the sensitivity analyses performed and their results, Section 7.2.2.7 describes the analysis for the FTF performed for information, and Section 7.2.2.7 discusses the conclusions. More detailed information on each of these matters can be found in the PODD, which also addresses other matters, such as the geology of the region and the tank area. [4] The PODD details the analysis performed to provide "reasonable assurance" that the performance objectives are not exceeded as a result of operation and closure of the tanks. The PODD meets the performance objective of "reasonable assurance" through the use of conservative estimates for parameter values, concrete/grout failure times, and human activity models. In addition, sensitivity and uncertainty analyses are used to ensure that a conservative estimate was developed for the closure of Tank 19 and Tank 18.

7.2.2.1 Analysis Approach

This general overview briefly describes the analysis process used by DOE to determine whether the stabilized residual waste in closed Tank 19 and Tank 18 will comply with the 10 CFR Part 61 Subpart C performance objectives relating to the protection of the general population. [79] The general approach applied by DOE in demonstrating compliance with 10 CFR 61.41 involved developing an appropriate conceptual model of the closed tanks, selecting a suitable mathematical model, calculating maximum annual radiation doses to a hypothetical member of the public from multiple exposure pathways, and combining these into an all-pathway dose for comparison to the performance objectives. [4]

The Tank 19 and Tank 18 performance evaluation focused on the residual radionuclides in the two tanks and the potential transport and exposure pathways. The radionuclide inventory

remaining in the two tanks, as described in Section 2.5, was used in modeling doses. The assumed waste form is produced by filling the tank with reducing grout to just below the springline, as described in Section 2.6. No mixing of the grout with residual waste is assumed for the performance modeling, an assumption that adds conservatism by locating all residual radioactivity at the bottom of the tank and, hence, closer to the transport mechanism (groundwater) that drives the postulated release as discussed further in this section. [4]

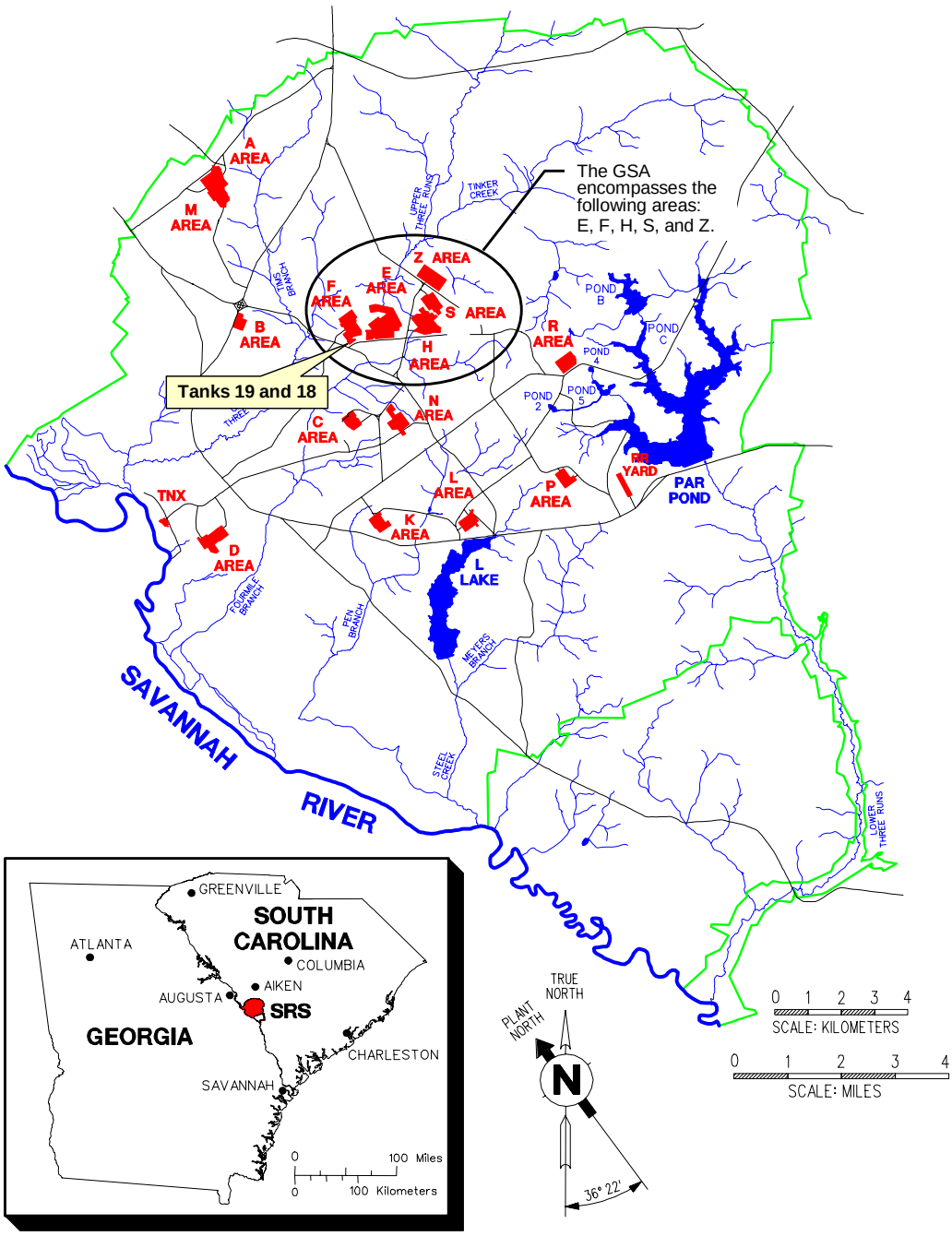
Tank 19 and Tank 18 are located in FTF. The FTF is located in F-Area, which is within the GSA near the center of the site, as depicted in Figure 7-1 below. The GSA includes numerous processing facilities, including F- and H-Area Separations Facilities (most noteworthy, the two 800-foot-long chemical separations canyons), the DWPF, the Saltstone Disposal Facility (SDF), the E-Area LLW Disposal Facility, FTF, and HTF. [4]

Unrestricted residential land use at SRS is prohibited within the GSA, which is planned to be zoned as industrial use only. The SRS Long Range Comprehensive Plan [92] describes the site's current situation, defines a vision for the evolution of the site over the next 50 years, outlines actions to achieve the vision, and guides the allocation of resources toward attainment of that vision. Among the Guidelines on which SRS land use is based include: (1) maintaining existing SRS boundaries, (2) continuing federal ownership of the land, and (3) prohibiting residential use of any SRS land.

Furthermore, unrestricted use of real property (land, facilities, and groundwater) in the GSA would not be permitted under DOE requirements. DOE Order 5400.5 [93] states in part:

> "Release of real property (land and structures) shall be in accordance with the guidelines and requirements for residual radioactive material presented in Chapter IV. These guidelines and requirements apply to both DOE-owned facilities and to private properties that are being prepared by DOE for release. Real properties owned by DOE that are being sold to the public are subject to the requirements of Section 120(h) of the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA), as amended, concerning hazardous substances, and to any other applicable Federal, State, and locate requirements." [93]

Figure 7-1: Location of Tank 19 and Tank 18



The assumptions and Institutional Control Plans regarding future land use contained in formal SRS documents, combined with DOE requirements pertaining to unrestricted release of radioactively-contaminated property such as the GSA, make residential use of the GSA not credible. Therefore, the location of maximum exposure for a hypothetical member of the public is considered to be at the boundary of the GSA. With the location of the hypothetical receptor on the GSA boundary, the limiting transport pathway for closure of Tank 19 and Tank 18 would be radionuclide inventory releases to groundwater and migration of the radionuclides to onsite surface waters. No direct radiation from the closed tanks could reach the hypothetical receptor due to the shielding afforded by the earthen cover and the one mile distance to the receptor location. These conditions also prevent release of airborne radioactivity reaching the receptor. [4]

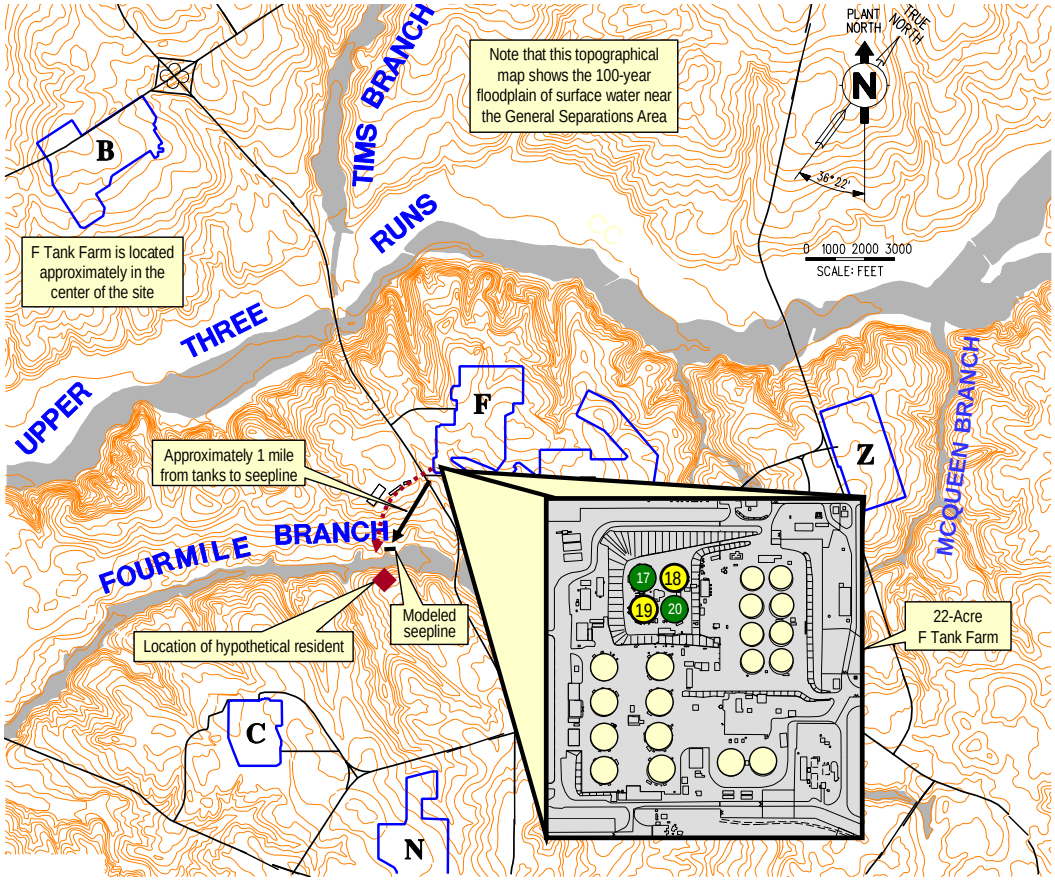
The location of Tank 19 and Tank 18 is very close to a groundwater divide that is located beneath the GSA. Due to the proximity of the tanks to the groundwater divide, both transport to Upper Three Runs and Fourmile Branch was considered for the groundwater flow and transport analysis. In reality, when Tank 19 and Tank 18 begin to leak contaminants into the soil, contaminated plumes are likely to travel toward Upper Three Runs, as well as Fourmile Branch. However, since Upper Three Runs is approximately the same distance from FTF and has a significantly greater flow than Fourmile Branch, it is conservative to model all migration from both Tank 19 and Tank 18 toward Fourmile Branch. Figure 7-2 below provides an illustration of the location of Tank 19 and 18 within the FTF on the groundwater divide between Fourmile Branch and Upper Three Runs (the divide is represented by a dashed line on the figure). [4]

Instead of estimating a percentage of groundwater flow to each of Upper Three Runs and Fourmile Branch, all flow in the upper aquifers (Water Table and Barnwell-McBean) is assumed in the conceptual model to be toward Fourmile Branch. [8] The assumed direction of flow for the third aquifer (Congaree) is toward Upper Three Runs, which is the normal flow path for this aquifer.

The point of compliance for the hypothetical member of the public is considered to be the location where groundwater from the upper aquifers seeps to the surface, i.e., the seepline at Fourmile Branch shown in Figure 7-2. The seepline is a swampy area where the groundwater from the top two aquifers (Water Table and Barnwell-McBean) first enters the surface water environment. This location was selected because:

- The seepline is the nearest appropriate place where a hypothetical member of the public could credibly be exposed to residual radioactivity from the tanks, considering planned restrictions on future use of the property [92]; and
- It is the point where this person would receive the highest potential dose, based on results from groundwater fate and transport modeling.

Figure 7-2: Location of Hypothetical Adult Resident in Relation to Tank 19 and Tank 18



As one can see in Figure 7-2, this nearest accessible point (e.g., location of hypothetical resident) is located approximately a mile southwest of the tanks near where groundwater from the two upper aquifers seeps into Fourmile Branch. Figure 7-3 and 7-4 below show Fourmile Branch and the seepline area, respectively.

Figure 7-3: Typical View of Fourmile Branch (Newman, May 2005)

Figure 7-4: Seepline Area at Fourmile Branch (Newman, May 2005)

The area where the seepline outcrops into Fourmile Branch was chosen as the potential point of maximum exposure for a hypothetical member of the public despite the location being in the central part of the site. Results of groundwater fate and transport modeling also support the seepline as the point of highest potential dose to a hypothetical member of the public. [4]

The conceptual model assumes that the radionuclide inventory released from Tank 19 and Tank 18 would contaminate the soil in the area of the seepline, the shore of the branch, and sediment in the branch, as well as water flowing in the branch. The conceptual model is discussed in further detail in the PODD. The exposure scenario selected involves hypothetical members of the public constructing a home outside the tank farm industrial area (GSA) on the south side of Fourmile Branch and living in the home. The dwelling is assumed to be in a location just downstream of the groundwater discharge (i.e., the seepline at Fourmile Branch) in order to maximize the potential contamination (and subsequent dose) received by the potential member of the public. This location allows the contaminants to thoroughly mix with the stream, and come into equilibrium with the contaminated sediments in the streams and shoreline. A hypothetical resident would receive exposure through the ten different pathways, as described below. Acute exposures were determined to either result in a lower potential dose due to lesser exposure times, or were determined to be implausible given the FTF closure concept or component configurations (Figure 7-5). [4]

7.2.2.2 Conceptual Model and Exposure Pathways

Various potential conceptual models were considered for evaluating exposures to the hypothetical member of the public with institutional controls in place. Factors considered in these conceptual models included:

- Receptor location would be outside of the GSA, which is appropriate for evaluating radiation exposure to a member of the public with institutional controls for the GSA in place, as discussed above;
- Past modeling shows that the groundwater exposure pathway dominates in all-pathway analyses;
- Scenarios involving residence would be more limiting than transient scenarios because more time would be spent in areas that could be affected by contamination; and
- Past modeling indicates that a hypothetical member of the public would receive a higher dose at the seepline than on the tank farm.

Considering such factors, the following basic conceptual models were evaluated:

- An adult resident situated on Fourmile Branch near the shore opposite to the GSA,
- A child resident situated on Fourmile Branch near the shore opposite to the GSA,
- A resident at the SRS site boundary, and
- A worker at the seepline at Fourmile Branch. [4]

The adult resident on Fourmile Branch was determined to be the most appropriate basic conceptual model because it is the most conservative of those considered; that is, it produces higher estimated doses than would any of the other models. [2] Figure 7-2 above shows the location of this point of compliance. [4]

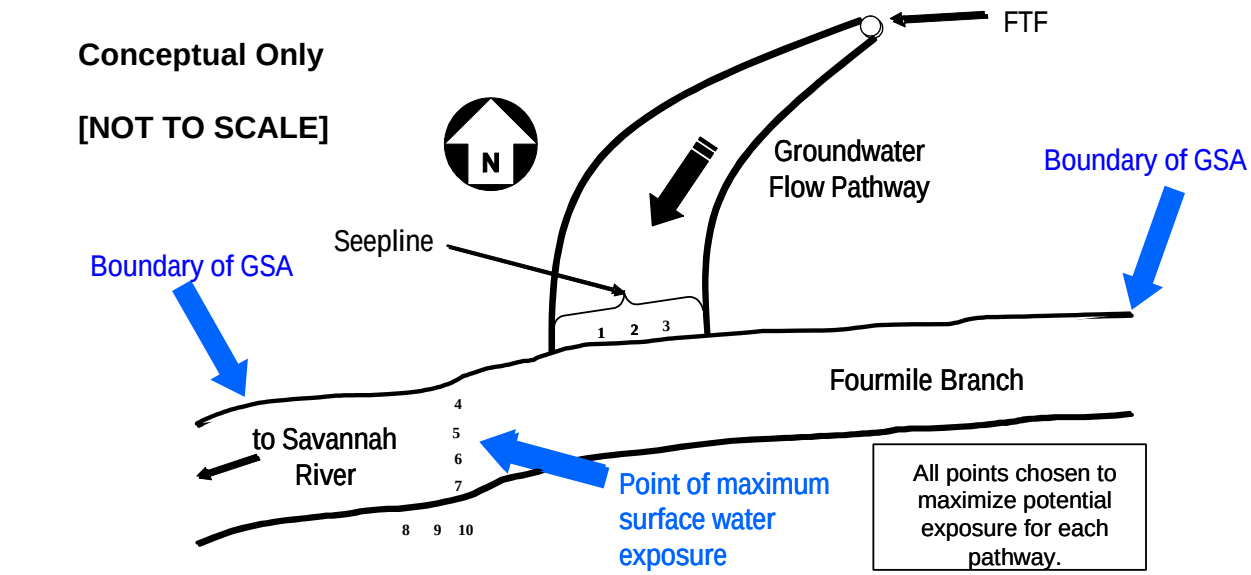
Ten different potential exposure pathways, as described below, were evaluated for the adult resident. The timeframe considered begins 100 years after tank closure and extends to 10,000 years after tank closure, consistent with NRC guidance. [94] The projected closure date is considered to be in the year 2020, which results in an evaluation start time (100 years after closure) in the year 2120. [4]

To estimate the dose from all-pathways, ten separate potential pathways via groundwater and air are assumed, based on consideration of possible exposure pathways, as shown in Figure 7-5 and listed below:

1. Incidental ingestion of soil (0.88 kg/yr) from the FTF-side shoreline deposits during recreational activities
2. Direct irradiation from the seepline (12 hrs/yr exposure)
3. Air inhalation (I-129 volatilization) at the seepline (12 hrs/yr exposure, 8,400 m^3/yr breathing rate)
4. Dermal contact with Fourmile Branch (12 hrs/yr exposure –swimming)
5. Drinking water (2 L/day) from Fourmile Branch
6. Ingestion of fish (9 kg/yr) from Fourmile Branch
7. Direct radiation from Fourmile Branch (12 hrs/yr exposure)
8. Ingestion of milk (120 L/yr) from cows fed vegetation grown on soil irrigated with Fourmile Branch water
9. Ingestion of meat (43 kg/yr) from cows fed vegetation grown on soil irrigated with Fourmile Branch Water
10. Ingestion of produce (184 kg/yr) irrigated with Fourmile Branch water [4]

The bases for the human health exposure values used in the analysis as given for the above exposure pathways are provided in the PODD.

Figure 7-5: All Exposure Pathways for Member of Public [4]



EXPOSURE PATHWAYS FOR A MEMBER OF PUBLIC

1. Incidental Ingestion of Soil From Shoreline Deposits (During Recreational Activities)
2. Direct Radiation from Seepline
3. Air Inhalation (I-129 Volatilization) At Seepline
4. Dermal Contact with Fourmile Branch
5. Drinking Water from Fourmile Branch
6. Ingestion of Fish from Fourmile Branch
7. Direct Radiation from Fourmile Branch
8. Ingestion of Milk from Cows Fed Vegetation Grown on Soil Irrigated with Fourmile Branch Water
9. Ingestion of Meat from Cows Fed Vegetation Grown on Soil Irigated with Fourmile Branch Water
10. Ingestion of Produce Irrigated with Fourmile Branch Water

7.2.2.3 Fate and Transport Mathematical Model

Computer simulations were performed to model the contaminant transport through groundwater, an approach that is typically used and accepted by regulatory agencies, to represent the complex processes associated with fate and transport of contaminants in groundwater. Details of the computer modeling along with the result of the modeling are documented in the PODD.

The mathematical model used to represent the conceptual model and calculate the human health impacts was the Multimedia Environmental Pollutant Assessment System (MEPAS), Waterborne Transport Component (RADCON), Version 4.1, multi-pathway risk model developed by Pacific Northwest Laboratory. [95],[4]

MEPAS is a tool for simulating contaminant source release, environmental transport and transformation, human exposure and dose in a multimedia environmental system for assessing the impact to the environment and public. In the environmental scenarios that MEPAS is used to address, contamination can move through the environment along several different transport pathways (i.e., vadose zone/groundwater pathway). MEPAS is used for tank closure to predict possible impacts of leaching contaminants from Tank 19 and Tank 18. MEPAS integrates

source-term, transport, and exposure models for contaminants. It assumes contaminants are transported from a contaminated area to potential human receptors through various transport pathways. MEPAS includes models to estimate human health impacts from radiation exposure.

The MEPAS code is widely used and accepted throughout the DOE complex. For example, it was used for the Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement, the Hanford Remedial Action - Environmental Impact Statement, and the No-Action Alternative for the Yucca Mountain EIS. [96]

The MEPAS modeling converts curies as inputs into concentration values at specific locations and times in the conceptual model, which are then converted into dose values in terms of total effective dose equivalent (TEDE). This conversion process, which is explained in the PODD, involves the use of multiple spreadsheets. [4]

7.2.2.4 Modeling Assumptions and Input Parameters

As previously discussed, the most important pathway for potential contamination to affect a member of the public is via the groundwater pathway. In developing the fate and transport groundwater model, it is necessary to specify various input parameters for the analysis. The goal is to use the best available data to select reasonable values that make the model as realistic and representative of actual conditions as practical. A key assumption made in the selection of the modeling parameters is that the tank top, the strong grout barrier, and the grouted waste form lose their hydraulic integrity 500 years after tank closure as a step function, which is considered a conservative assumption consistent with NRC guidance. [94]

An overview of some significant modeling parameters and assumptions associated with the groundwater pathway given the assumed time of grout failure are represented in Figure 7-6. The identified modeling parameters represent many areas that are further analyzed by sensitivity studies, for example distribution coefficients (K_d), infiltration rates, dispersion, and aquifer water balance. A general description of the assumed modeling parameters given within the figures is provided in the discussions that follow. Also provided are several other key assumptions associated with the development of the conceptual model. Detailed discussions of the groundwater fate and transport modeling assumptions with bases for the selection of the input parameter values can be found in the Tank 19 and Tank 18 PODD. [4]

Figure 7-6: Conceptual Model Showing Assumptions and Key Parameters Prior to Hydraulic Grout Failure [4]

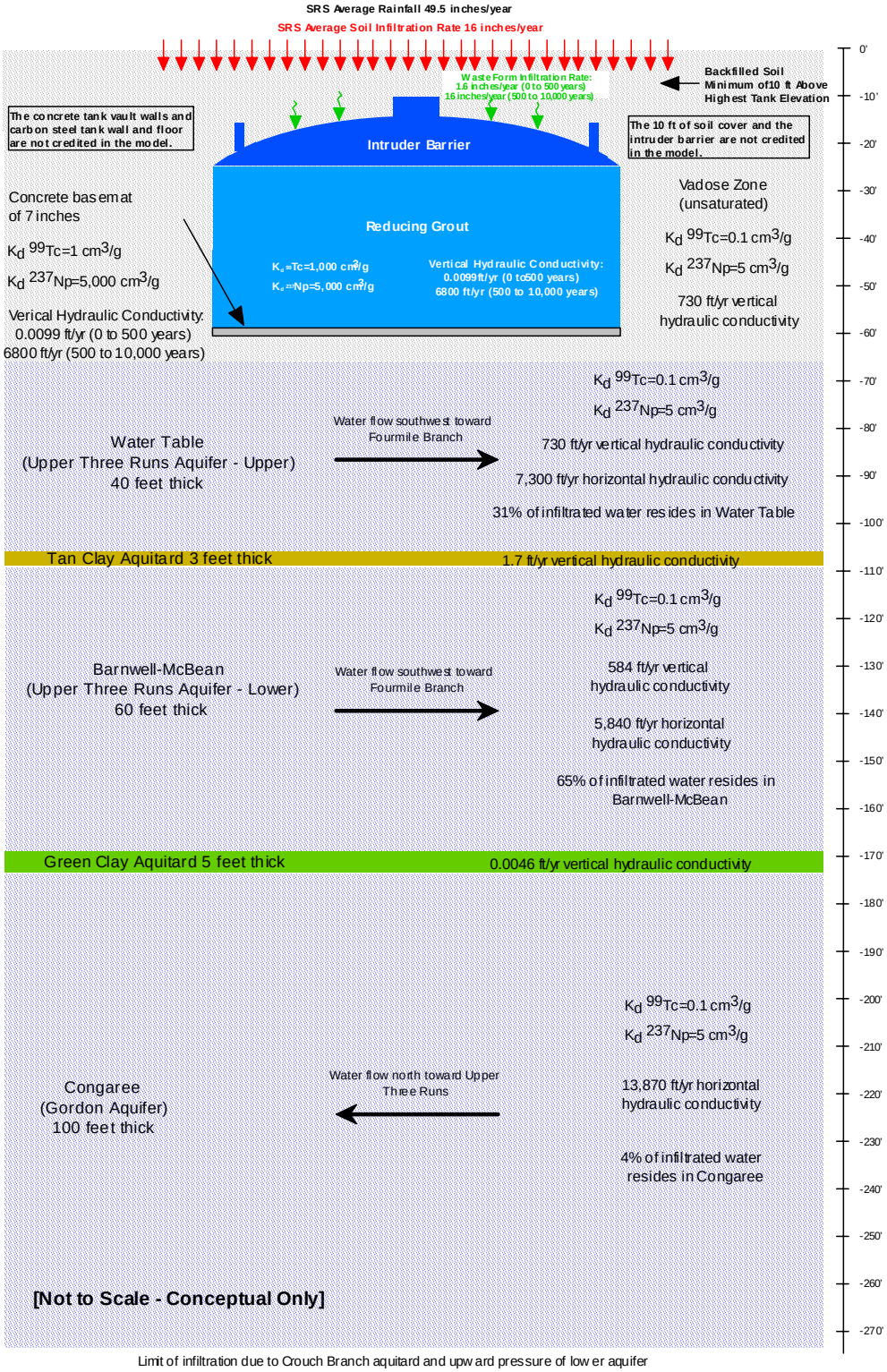


Figure 7-6 illustrates the assumed changes in the modeling parameters for the change in degradation state. The hydraulic failure of the tank top, the strong grout in the top of the tank, and the grouted waste form (500 years after tank closure) is simulated by increasing the infiltration rate in these materials to the rate observed for site soils and increasing the hydraulic conductivity to that of sand. [4]

The infiltration rate reflects movement of surface water downward, which depends on factors such as the amount of precipitation, surface runoff, evaporation, and soil type. The soil infiltration rate of 1.6 inches per year specified in Figure 7-6 equates to an infiltration rate of 4 cm/yr, which is the infiltration rate assigned to the grout monolith for the first 500 years after closure while the grout is assumed to stay intact. This value was selected based on past SRS performance assessment experience and studies, such as the performance assessment for the Z-Area Saltstone, which developed an average infiltration rate through a clay cap of 2 cm/yr. [97] Also, in the *Special Analysis: Revision of Saltstone Vault 4 Disposal Limits*, infiltration rates between 1 cm/yr and 4 cm/yr were identified as representative of upper boundary infiltration rates through the saltstone cap for the first 300 years after closure. [98]

The soil infiltration rate of 16 inches per year specified in Figure 7-6 equates to an infiltration rate of 40 cm/yr, which is based on past SRS studies of water infiltration through site soils. [99] A 40 cm/yr infiltration rate was assigned to the grout monolith and tank basemat for 500-10,000 years after tank closure. The use of the soil infiltration rate for the degraded structures is considered a very conservative approximation of the infiltration of cracked and degraded grout.

The hydraulic conductivity is a measure of the rate at which water flows through the subsurface environment. A 0.0099 ft/yr vertical hydraulic conductivity as specified in Figure 7-6 is assigned for the grout and concrete basemat for the first 500 years after closure during which the structure is assumed to remain intact. After failure of the grout, a hydraulic conductivity of 6,800 ft/yr as specified in Figure 7-6 is applied, based on water flow through sand. The use of vertical hydraulic conductivity for sand is considered a very conservative approximation of the infiltration of cracked and degraded grout. Values for vertical hydraulic conductivities in site soils were selected based on a hydrogeologic study performed for the region. [100]

Another important assumption presented in Figure 7-6 is the water balance among the three aquifers. The assumed water infiltration balance among the aquifers is 31/65/04, expressed as percentages in the Water Table Aquifer, Barnwell-McBean Aquifer, and Congaree Aquifer, respectively. That is, 31 percent of contaminants transported in groundwater would be located in the Water Table Aquifer, 65 percent in the Barnwell-McBean Aquifer, and the remaining four percent in the Congaree Aquifer. These figures are based on a site study. [101] (A sensitivity study discussed below shows that the analysis results are relatively insensitive to the water balance.)

The K_d values represented in Figure 7-6, which are chosen to determine the travel rate of radionuclides in soil and other matrices, such as grout, are particularly important. Where such data were available, K_d values used were derived from laboratory or field testing of SRS site material. The remaining K_d values were determined from literature sources. Analyses of the sensitivity of the modeling results to Tc-99 K_d values show that the predicted peak annual doses are sensitive to this parameter and increase proportionally with a decrease in the Tc-99 K_d value. These sensitivity studies are discussed below.

In some cases, a solubility model is better suited for describing the interaction of radionuclides then the K_d model, e.g., reducing grout with plutonium and uranium. Additional solubility models are presently being developed for future applications.

Other important modeling assumptions with inherent conservatisms are noted in Figure 7-6. These key assumptions are:

- The planned strong grout intruder barrier described in Section 2.6 was considered in calculating surface water infiltration through the closure system. However, no credit was taken in analyses for the earthen cover that will eventually be installed when the FTF itself is closed, even though this cover is presumed to be topped with a low-permeability geomembrane barrier. Not considering the cover adds conservatism with respect to contaminant migration time, since the cover will reduce surface water infiltration into the closure system. [4]
- Neither the concrete tank nor the carbon steel liner are considered in the conceptual model. Ignoring the tank structure in this manner adds conservatism because the tank structure will prevent release of contaminants to groundwater so long as the tank maintains its integrity. [4]

7.2.2.5 Results of the Analysis

The results of the MEPAS modeling were used to determine an all-pathways dose to a member of the public for comparison with the 10 CFR 61.41 performance objectives. The ICRP-30 dose methodology was used in computing annual doses for comparison to the performance objectives, as reported in the USEPA Federal Guidance Report 11. [102] The all-pathways analysis as described above and detailed in the PODD [4] produced the following results:

Table 7-1: Maximum Annual All-Pathway Doses to a Member of the Public with Institutional Controls in Place, 100 – 10,000 Years [4]

Dose Type	Limit (mrem/yr)	Factor	Tank 18	Tank 19
Whole Body	25	Annual Dose (mrem/yr)	0.04	0.009
		Peak Year After Closure	6405	735
		Controlling Nuclide	Np-237	Tc-99
Thyroid	75	Annual Dose (mrem/yr)	0.00005	0.04
		Peak Year After Closure	6405	735
		Controlling Nuclide	Np-237	Tc-99
Any Other Organ	25	Annual Dose (mrem/yr)	0.97	0.02
		Peak Year After Closure	6405	735
		Controlling Nuclide	Np-237	Tc-99

The results of the analyses show that estimated maximum annual radiation doses from all pathways are well below the applicable limits. [4]

Figures 7-7 through 7-10 below graphically depict the all-pathway dose peaks. Each figure shows estimated whole-body total effective dose equivalent in mrem/yr during the period from 100 to 10,000 years following tank closure and the radionuclides that produce this dose. [4]

Table 7-2, which follows these figures, shows the specific radionuclides that produce the most whole-body dose on a percentage basis. [4]

Figure 7-7: Tank 19 All-Pathways Beta Peak Dose [4]

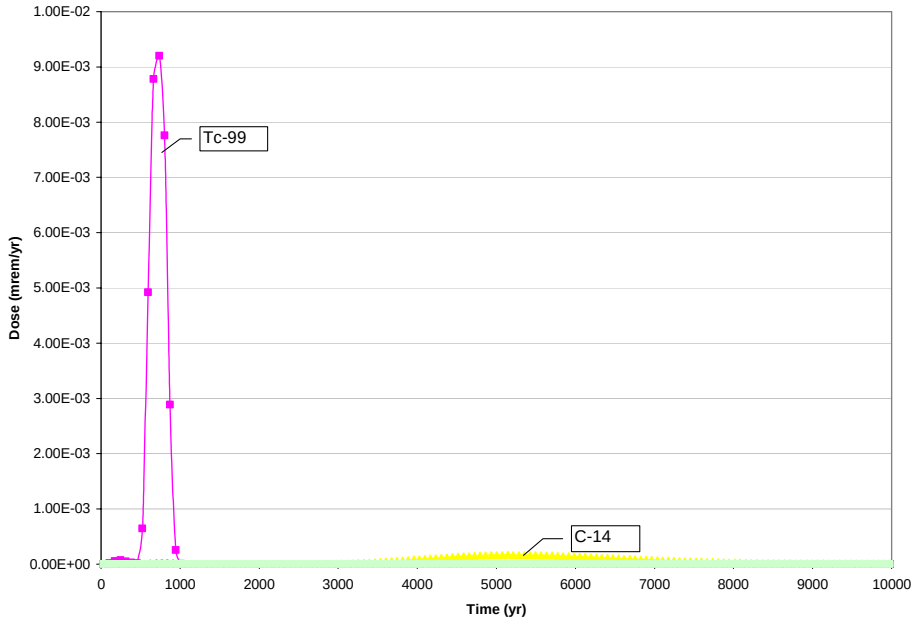


Figure 7-8: Tank 19 All-Pathways Alpha Peak Dose [4]

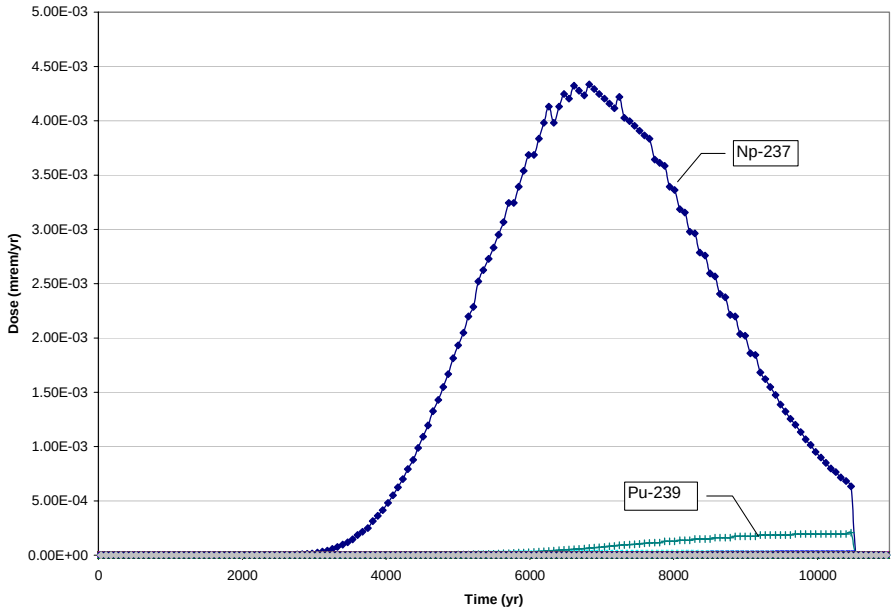


Figure 7-9: Tank 18 All-Pathways Beta Peak Dose [4]

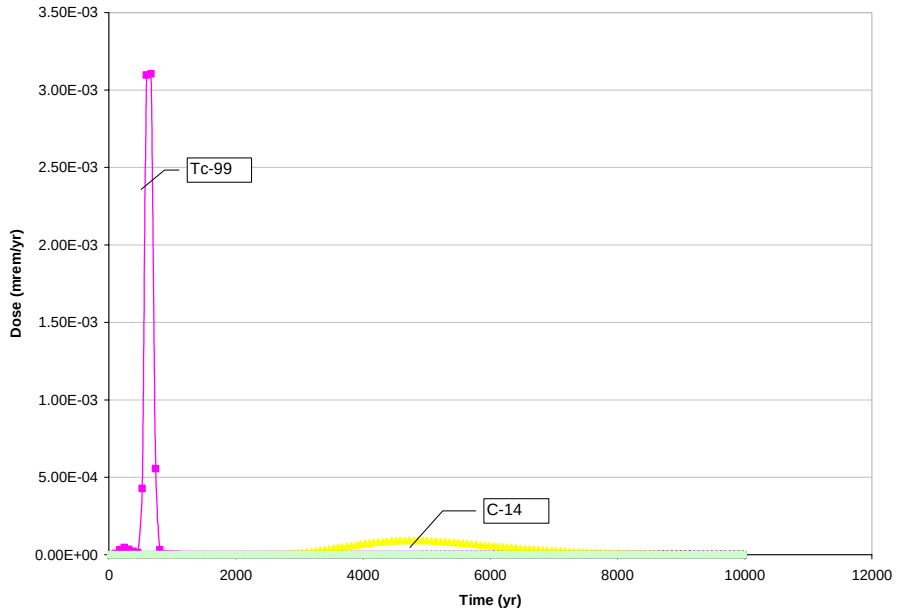


Figure 7-10: Tank 18 All-Pathways Alpha Peak Dose [4]

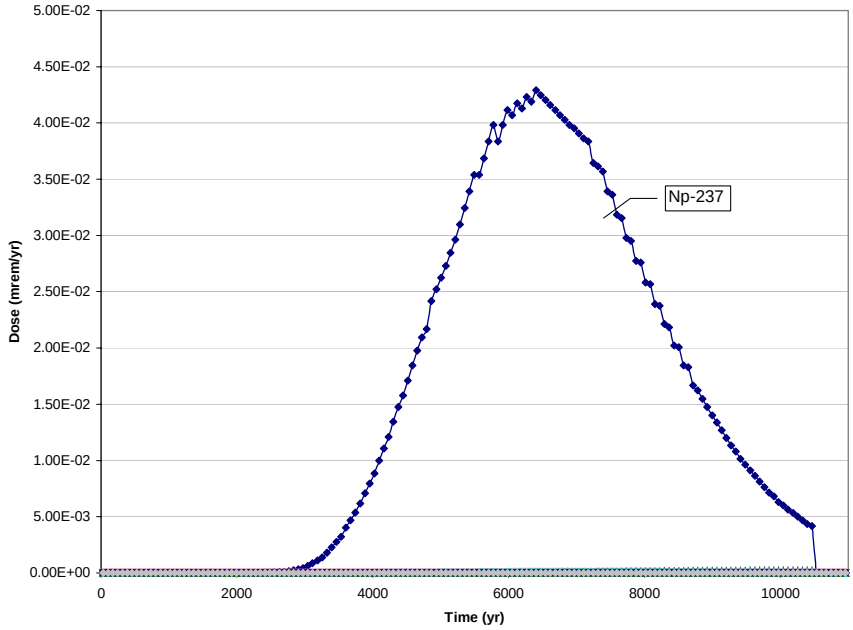


Table 7-2: Peak Annual Whole-Body Dose Contributors[a][4]

Source	10 CFR 61.41 Subpart C Performance Objectives (mrem)	Peak Year Whole-Body Dose (mrem)	Tc-99 Contribution (%)	I-129 Contribution (%)	Np-237[b] Contribution (%)
Tank 19	25	0.009	99.8	0.2	0
Tank 18	25	0.04	0	0	99.7

a. Tc-99, I-129, and Np-237 were called out in this table as these three nuclides contributed nearly 100% of the dose.
b. Tank 18 received much higher amounts of Np-237 than Tank 19 through transfers of laboratory wastes.

From the results reflected in Tables 7-1 and 7-2 and the associated graphs, one can see that:

- The sum of the peak all-pathways whole-body dose for both beta-gamma and alpha-emitting radionuclides over the 10,000-year period after closure for Tank 19 is 0.009 mrem/yr, which is 0.036% of the NRC performance objective in 10 CFR 61.41.
- The sum of the peak all-pathways whole-body dose for both beta-gamma and alpha-emitting radionuclides over the 10,000-year period after closure for Tank 18 is 0.04 mrem/yr, which is 0.16% of the NRC performance objective.
- The peak annual dose to the thyroid from Tank 19 is 0.04 mrem/yr compared to the 10 CFR 61.41 limit of 75 mrem/yr. Conversion factors specified by the U. S. EPA were used to calculate annual doses to the thyroid and other critical organs for each relevant radionuclide on an organ-by-organ basis. [102]
- The peak annual dose to any other organ from Tank 19 is 0.02 mrem/yr compared to the 10 CFR 61.41 limit of 25 mrem/yr.
- The peak annual dose to the thyroid for Tank 18 is 0.00005 mrem/yr compared to the 10 CFR 61.41 limit of 75 mrem/yr.
- The peak annual dose to any other organ from Tank 18 is 0.97 mrem/yr compared to the 10 CFR 61.41 limit of 25 mrem/yr.

Therefore, doses to the whole body, the thyroid, and any other organ of a member of the public from disposal of the residual waste in Tank 19 and Tank 18 will not exceed the performance objectives set forth in 10 CFR 61.41. This situation, coupled with the results of sensitivity analyses described below – which showed the estimated peak annual dose approaching none of the performance objectives with significant variations in key input parameters – provide reasonable assurance that the performance objectives of 10 CFR 61.41 will be achieved.

7.2.2.6 Uncertainty and Sensitivity Studies and Analysis

Various sensitivity analyses were performed to help put the modeling results into perspective, and to identify input parameters sensitive to the results of the analytical modeling. An additional analysis was performed for informational purposes that modeled the release of the radionuclide inventory from the entire FTF – all 22 tanks along with the ancillary equipment in this tank farm. The results of this modeling, explained below, provide additional perspective on the results of the analyses for Tank 19 and Tank 18.

The objective of the sensitivity studies performed was to investigate uncertainties that are inherent in conceptual models and mathematical models such as MEPAS to help confirm that the base case modeling provides reasonable assurance that the performance objectives will be achieved. The results of the uncertainty and sensitivity analyses reinforce the conclusions that the base case modeling clearly demonstrates the planned closure of Tank 19 and Tank 18 will be protective of human health and the environment and in full compliance with the performance objectives of 10 CFR 61.41.

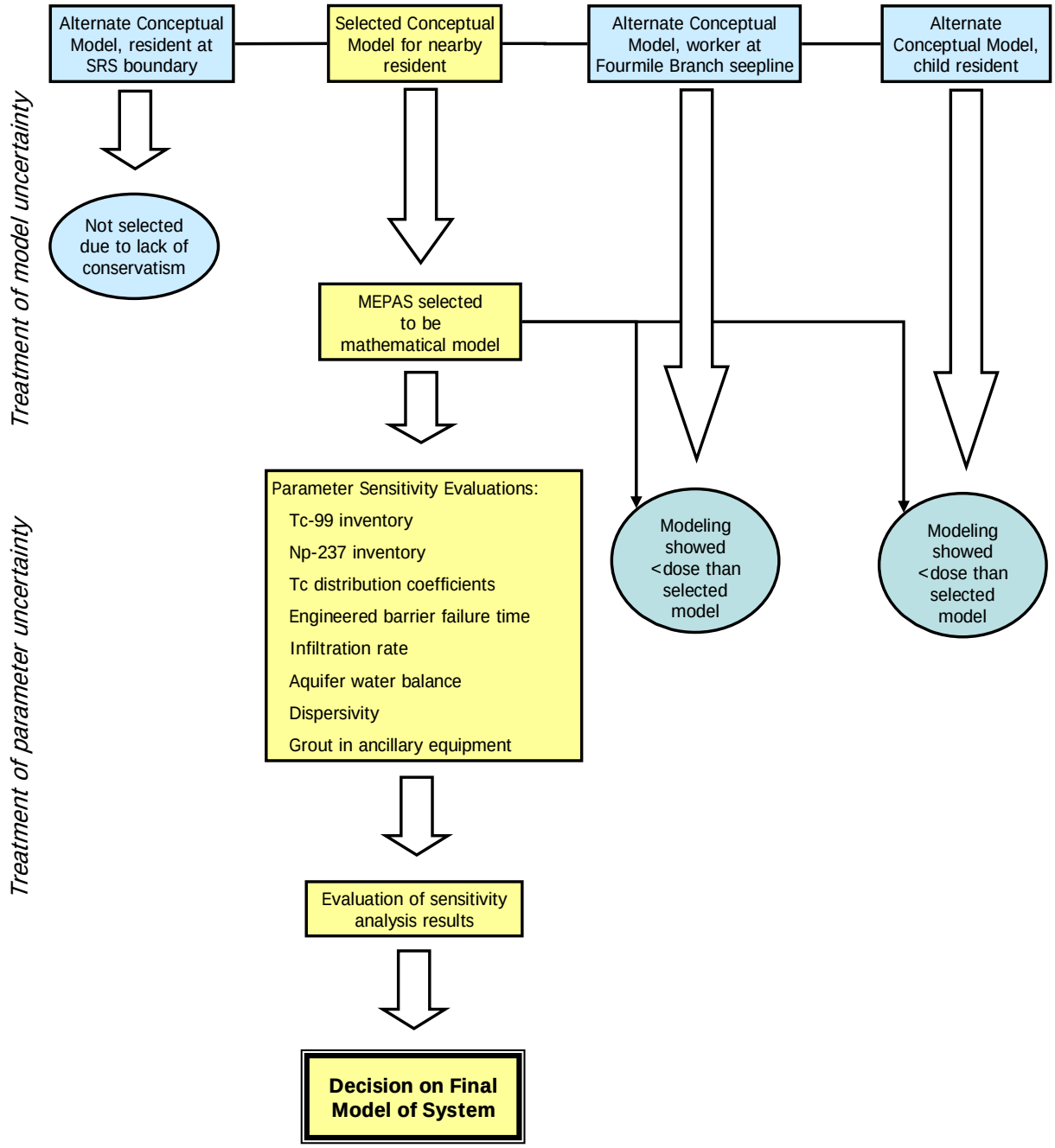
Figure 7-12 below illustrates the process used in the sensitivity analyses. The results, which are detailed in the PODD, are summarized as follows. Potential key uncertainties and sensitivities were considered and ten different modeling parameters were identified to evaluate quantitatively. Sensitivity analyses were then accomplished by performing MEPAS modeling runs with different parameter values, input either individually or in combination, and evaluating the resulting exposure estimates, with variations in calculated peak annual dose as the measure of sensitivity. The primary objective was to identify which parameters have the most impact on the modeling results. [4] Among the results:

- The most sensitive input parameter involves dispersion of contaminants during transport through groundwater: reducing the longitudinal, transverse, and vertical dispersivity each by 50 percent results in increasing the peak estimated annual dose by 56 percent.
- The second most sensitive input parameter is the Tc-99 inventory, with peak estimated dose directly proportional to estimated Tc-99 inventory.
- Doubling the infiltration rate after engineering barrier (grout and basemat) failure increases peak annual dose by 32 percent.
- Decreasing the Tc-99 K_d value for grout from 1000 to 1 cm^3/g results in a 32 percent increase in estimated peak annual dose.
- Doubling the Np-237 inventory produces an increase in the peak annual dose of 24 percent (increases in Np-237 inventory below a 50 percent increase have no effect on peak dose because Tc-99 peak doses are higher than those from Np-237).
- Decreasing the time of grout and basemat failure from 500 years to 100 years produces an eight percent increase in estimated peak annual dose.
- Decreasing the Tc-99 K_d value for the basemat or for soil does not result in a significant increase in estimated peak annual dose.

- Changing the water infiltration balance from 31/65/04 to 80/20/0 (percentages in the Water Table Aquifer, Barnwell-McBean Aquifer, and Congaree Aquifer, respectively) does not significantly increase peak estimated annual dose.
- Whether ancillary equipment (transfer lines, evaporators, pump pits, diversion boxes, etc.) is filled with grout has no effect on estimated peak annual dose.

It was concluded from the sensitivity studies that the input parameters could vary widely from base-case values and the estimated dose would still be far below the 25 mrem/yr limit. [4]

Figure 7-11: Uncertainty Analysis Process for Member of the Public Model [4]



7.2.2.7 F-Tank Farm Modeling

The release of the radionuclide inventory from the entire FTF was analyzed for perspective. The results of this all-pathways analysis reinforced the conclusion that the performance objectives of 10 CFR 61.41 will be achieved with reasonable assurance for all 22 FTF tanks and ancillary equipment.

Except for the estimated residual radionuclide inventory, the FTF analysis followed the same process described above and detailed in the PODD for Tank 19 and Tank 18. The conceptual model used was similar except for the increased radionuclide inventory, and MEPAS was used as the mathematical model.

As explained in the PODD, the source-term employed in the modeling of FTF was based upon Tank 19 and Tank 18 residual materials sample results, the estimated radionuclide inventory remaining in the previously closed Tank 17 and Tank 20, and an *a priori* estimate of the source term from the remaining eighteen FTF tanks, and all of FTF ancillary equipment. The radionuclide inventory estimated for the other eighteen FTF tanks and ancillary equipment was based on process records and cleaning experience. The ancillary equipment includes pump tanks, pump pits, transfer lines, evaporator cells, etc. Details of the inventory estimates appear in Appendix E.

Table 7-3 presents the results of this analysis, with the Tank 19 and Tank 18 results from Table 7-1 shown for comparison.

Table 7-3: Peak Annual All-Pathways FTF Dose [4]

Source	10 CFR 61.41 Subpart C Performance Objectives (mrem)	Peak Year Whole-Body Dose mrem (peak year)	Tc-99 Contribution (%)	I-129 Contribution (%)	Np-237 Contribution (%)
FTF	**25**	**0.08 (595)**	**87.1**	**12.9**	**0**
Tank 19	25	0.009 (735)	99.8	0.2	0
Tank 18	25	0.04 (6405)	0	0	99.7

The all-pathways analysis produced the following estimates of peak annual dose from residual radionuclides estimated to be present in all 22 tanks at the time of tank farm closure for the period from 100 years through 10,000 years after closure:

- 0.08 mrem whole-body, compared to the 10 CFR 61.41 limit of 25 mrem/yr;
- 4.4 mrem to the thyroid, compared to the 10 CFR 61.41 limit of 75 mrem/yr; and
- 2.6 mrem to any other organ, compared to the 10 CFR 61.41 limit of 25 mrem/yr

As can be seen from the data, estimated doses for the entire FTF are well below the applicable limits, reinforcing the conclusion that Tank 19 and Tank 18 will meet performance objectives of 10 CFR 61.41.

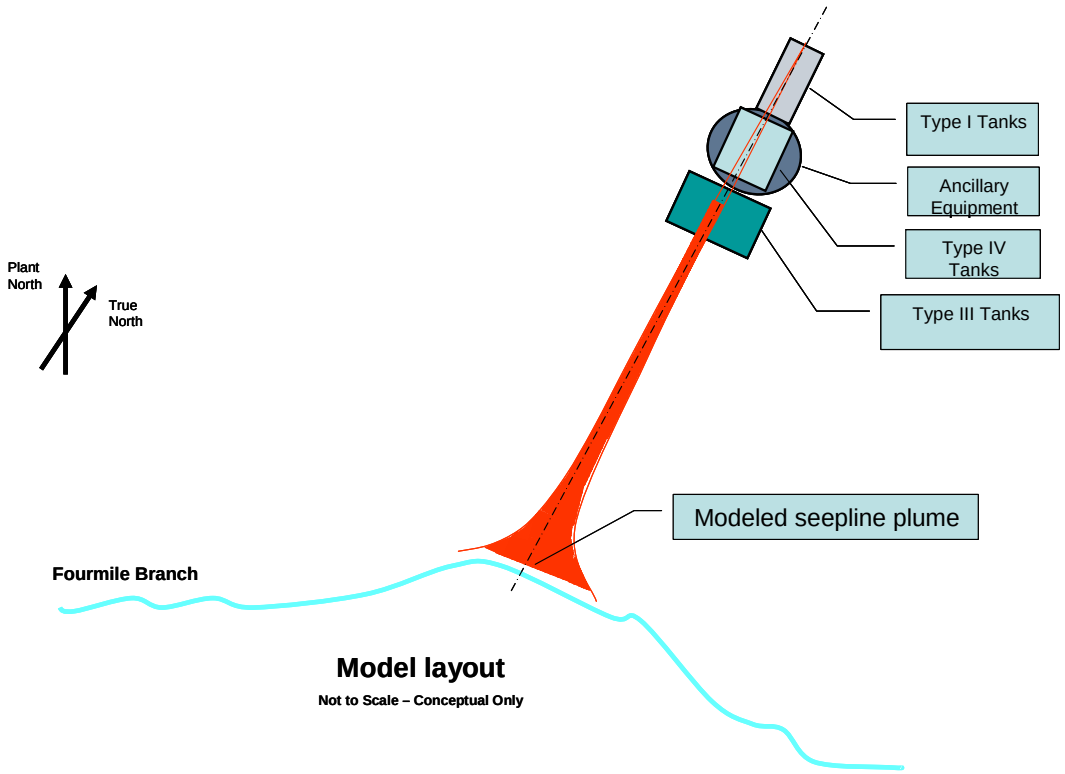
As noted previously, this analysis was performed to provide additional perspective for the Tank 19 and Tank 18 results.

Comparison of key modeling assumptions used in the analysis of the entire FTF with actual known conditions helps place the FTF results into context. Table 7-4 provides such a comparison. Each modeling assumption is considered conservative.

Table 7-4: Comparison of Modeling Assumptions for F-Tank Farm to Actual Conditions [4]

Actual Conditions	Modeling Assumption
FTF has 22 individual tanks that will, upon closure and grouting, fail hydraulically and chemically slowly and at independent rates, over time.	The modeling assumption is that all 22 tanks will fail hydraulically and chemically at 500 years from closure. Hydraulic failure is defined as allowing water to move through the grout monolith at the same rate as sand. Chemical failure is defined as the grout having total oxidizing properties, instead of reducing properties.
FTF has 22 high-level radioactive waste tanks ranging across 22 acres. Each tank has a unique plume that may, or may not overlap other plumes in the vicinity; however, plume centerlines do not overlap.	FTF tanks are represented as contamination sources in three super tanks along one centerline (thus with forced overlap) located at the same initial physical location and driven *en mass* along the same centerline to Fourmile Branch. This causes the centerline of the plumes to be "forced" to overlap (Figure 7-12).
Ancillary equipment (transfer lines, pump pits, diversion boxes, evaporators, etc.) is spaced irregularly across the 22 acres of the FTF site.	The FTF ancillary equipment source term is represented as a contamination source in one super tank along the same centerline with the three other super tanks.
Each of the 22 FTF tanks has a 3/8-inch carbon steel liner inside a concrete tank. It has been estimated that this carbon steel will remain essentially intact for approximately 700 years. An intact tank liner will restrict contamination to the environment.	The tanks are modeled without any credit assigned for contamination hold-up due to the carbon steel liner.
Each of the 22 FTF tanks has a grout intruder cap and >10-feet of soil backfilled above the waste form. The intruder barrier and soil backfill will impede the flow of water to the top of the reducing grout.	The tanks are modeled without credit assigned for the intruder barrier and soil backfill.
Closure of soils in the FTF is regulated by CERCLA and administered by SCDHEC and EPA IV. A presumed end state of the FTF closure is a low-infiltration cap.	The tanks are modeled without any credit assigned for the reduced water infiltration afforded by a low-infiltration cap.
The FTF tanks are located above a groundwater divide that would result in contaminated plumes in the upper aquifers migrating toward both Upper Three Runs and Fourmile Branch.	The model forces all flow in the upper aquifers toward Fourmile Branch. It is considered conservative to model migration from the FTF tanks toward Fourmile Branch since Upper Three Runs has a significantly greater flow.

Figure 7-12: Plume Centerline Overlap [4]



7.2.2.8 Conclusions

The SRS Land Use Plan requires Federal ownership and control of the site well beyond 100 years after closure of FTF. DOE 5400.5 precludes release of the GSA unless the radiological hazard meets the requirements of DOE 5400.5, Chapter 4, which will result in virtual perpetual DOE control.[18]

No unrestricted use of the land or groundwater will be permitted for the GSA, which includes FTF. Thus, a member of the public could not contact the groundwater in the vicinity of Tank 19 or Tank 18. Due to the restrictions in the SRS Land Use Plan, a member of the public could only

[18] Although for the purposes of this analysis it is assumed that there is possible loss of institutional control for periods beyond 100 years which results in unrestricted access of the site, DOE requirements do not permit termination of necessary controls or unrestricted access to areas requiring radiological control. The Atomic Energy Act, and the Department of Energy Organization Act, assigns to the Department the responsibility to protect the public, the environment, and property from the hazards associated with its research, development, production, or other activities. This responsibility, which lacks any limit in time, includes protecting the public and environment from radiation or radioactive material. DOE policy (DOE P 454.1, *Use of Institutional Controls*) and requirements (DOE 5400.5, *Radiation Protection of the Public and Environment*) mandate continued control of property until the radiological hazard associated with this property is reduced to levels at which regulation under the AEA is no longer needed to ensure protection of the public and environment. By these requirements, DOE establishes "durable institutional controls" that are effectively similar to those required in 10 CFR 20.1403(e)(2)(ii) for sites requiring long-term control. Given these requirements, long-term DOE control over the GSA to prevent residential use is a reasonable assumption.

contact potentially contaminated surface water off-site, approximately six miles from the facility where the mouth of Fourmile Branch enters the Savannah River [92] Further, the approximate one-mile distance between Tank 19 and Tank 18 and the seepline provides additional protection for a member of the public.

The solidified waste form in Tank 19 and Tank 18 reduce releases to the environment to very low levels as demonstrated by the dose calculations discussed previously. Also, the engineered grout in Tank 19 and Tank 18 was designed to retard migration of the radionuclides from the grouted waste form. [71] Moreover, sensitivity studies were conducted on key modeling assumptions to demonstrate reasonable assurance that the performance objective for protection of the general population will be met.

One final matter concerns whether reasonable efforts have been made to maintain releases of radioactivity in effluents to the general environment ALARA in accordance with 10 CFR 61.41. This element of the performance objectives will be achieved for reasons such as the following:

- The location of the tanks in an industrial area in the central part of the site is consistent with ALARA principles in that it provides a large Federally-owned buffer area around the tank site.
- The controls over GSA property use discussed previously serve to prevent residential use of the tank area, which serves ALARA purposes.
- The ALARA process was followed in tank closure system design as evidenced by the fact that estimated peak annual doses to a member of the public with institutional controls in place are small fractions of the applicable limits, serving as a *de facto* demonstration that ALARA will be achieved with the planned closure system.

In summary, the results of the various evaluations – the analyses for Tank 19 and Tank 18, the sensitivity analyses, the FTF analysis, and ALARA considerations – provide reasonable assurance that Tank 19 and Tank 18 will meet the performance objectives of 10 CFR 61.41 after they are closed in place as planned.

7.2.3 Section 61.42, Protection of Individuals From Inadvertent Intrusion

10 CFR 61.42 states [79]:

> Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed.

PRESENTATION OF INFORMATION IN THIS SECTION

Information related to protection of the general population from inadvertent intrusion into the area of Tank 19 and Tank 18 is presented in the following sections: Section 7.2.3.1 contains a general description of the analysis approach used in the conceptual model, Section 7.2.3.2 describes the conceptual model exposure pathways evaluated, Section 7.2.3.3 describes the mathematical model used to represent the conceptual model, 7.2.3.4 provides the results of the analyses, Section 7.2.3.5 details the sensitivity analyses performed and their results, and Section 7.2.3.6 discusses the conclusions. More detailed information on each of these matters can be found in the PODD.

7.2.3.1 General Approach

An inadvertent intruder is considered to be a person or persons that occupy a site in the vicinity of either Tank 19 or Tank 18 and engage in normal personal pursuits. Although the feasibility of human intrusion into a near surface disposal facility like Tank 19 and Tank 18 is only hypothetical, consideration of that possibility was evaluated for compliance with the requirement of 10 CFR 61.42 for a 10,000-year period after closure. The analysis assumes that this intruder could establish a permanent homestead on the disposal site and that he or she has no *a priori* knowledge of waste disposal activities at the site. [4]

The intruder analysis considers that active Federal institutional control over the disposal facility will be maintained at least 100 years after closure. [103] For direct intrusion into disposal units after loss of active institutional control, DOE examined three potential intruder scenarios which include: (1) Intruder-Construction, (2) Intruder-Recreation and (3) Intruder-Agricultural scenarios. A general overview of these scenarios is provided as follows with details given.

(1) The Intruder-Construction scenario involves a hypothetical intruder who constructs a residence directly above either Tank 19 or Tank 18, or directly above an abandoned FTF transfer line. As part of the home construction the intruder digs a three meter basement, which is considered the typical excavation depth in developing a basement for a home. The assumed duration of the construction activity is three months.

(2) The Intruder-Recreation scenario involves a hypothetical intruder who engages in recreational activities in the area of the Tank 19 and Tank 18. The recreational intruder is assumed to spend four hours per day at the seepline, swim in Fourmile Branch, and eat fish from Fourmile Branch.

(3) The Intruder-Agricultural scenario involves a hypothetical intruder who resides permanently in an area near either Tank 19 or Tank 18. The

> agricultural intruder is assumed to drill through an abandoned FTF transfer line in constructing a well for domestic water supply. The contaminated well cuttings are brought to the surface and spread over a specified area that is subsequently tilled for a garden. [104] The agricultural intruder obtains drinking water from a well which supplies water from the Congaree Aquifer. The Congaree Aquifer is the groundwater source below Tank 19 and Tank 18 that has a sufficient yield to support the water needs of the Intruder-Agricultural, as explained in the PODD. [4]

The scenario of drilling through Tank 19 or Tank 18 was not analyzed as part of the Intruder-Agricultural scenario because drilling practices in this area do not include drilling through a barrier similar to rock. The closed tank is filled with grout and capped with a >2,000 psi strong grout. This presents a dramatically more difficult media through which to drill than is seen in this area of South Carolina and provides reasonable assurance that drilling activities would be precluded and moved to a more desirable location. [4] Therefore, the Intruder-Agricultural scenario considers drilling through a transfer line, although this scenario is not directly germane to the evaluation because FTF waste transfer lines are beyond the scope of the evaluation related to closing Tank 19 and Tank 18. DOE will address closure of the waste transfer lines and other ancillary equipment later as part of a separate Section 3116 Waste Determination. [4]

The developed Intruder-Agricultural scenario was determined to be the most limiting inadvertent intruder scenario (i.e., the scenario resulting in the highest dose potential) and as such was analyzed in depth. The Intruder-Agricultural scenario bounds the Intruder-Construction scenario because the depth of the planned soil cover over both Tank 19 and Tank 18 (>3 meters) prevents contact with the waste form. Once grouted, the dose hazard at the tank top from the waste in the grouted tank becomes effectively zero due to the shielding present. The Intruder-Agricultural scenario also bounds the Intruder-Recreation scenario due to the higher exposure potential from contact with the exhumed transfer line when compared to the exposure potential at the seepline and Fourmile Branch. The longer transport times involved in the Intruder-Recreation scenario case allow significant decay of shorter-lived radionuclides such as Sr-90 before they reach the receptor. [4]

7.2.3.2 Conceptual Model and Exposure Pathways

The Intruder-Agricultural scenario assumes that an inadvertent intruder moves onto the disposal site and drills through an abandoned transfer line in constructing a residential well. The evaluation of the post-drilling scenario assumes an 8-inch drilling diameter based on the typical size of pipe used for a residential well. The drill cuttings are brought to the surface, and are assumed to be spread over an area that is subsequently used for a garden. The garden size (contaminated area) is assumed to be 1,000 m^2 and the waste is assumed to be mixed to a depth of 15 centimeters based on the typical plow depth when an area is tilled for a garden. Further details and bases for the analysis assumptions and applied input parameters are provided in the PODD. [4]

The Intruder-Agricultural scenario assumes that the intruder is subjected to the following exposure pathways shown in Figure 7-13:

1. Direct radiation from the well drilling cuttings in garden soil (~88 hrs/yr exposure);

2. Internal radiation from ingesting produce (92 kg/yr) from the garden;
3. Internal radiation from ingesting soil (0.037 kg/yr) from the garden;
4. Internal radiation from drinking well water (730 L/yr) from the Congaree Aquifer;
5. Inhalation of resuspended soil from the garden (8,000 m^3/yr breathing rate). [4]

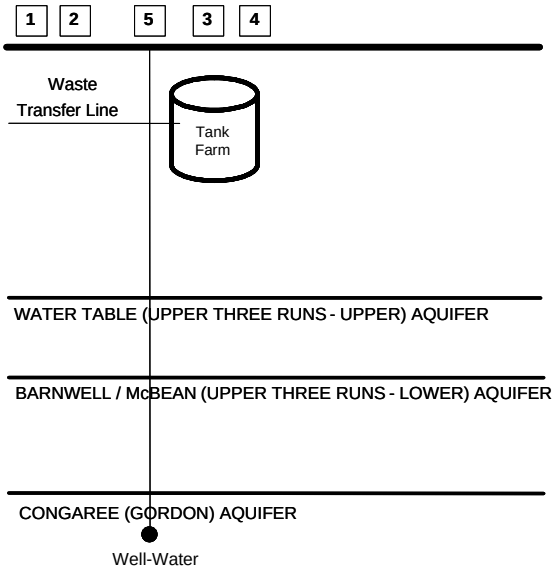
As explained in the PODD, the 92 kg/yr produce ingestion assumption is based on the 184 kg/yr site-specific values adjusted for consistency with DOE guidance. [105],[106] The soil ingestion assumption is based on EPA guidance. [107] The water consumption is based on NRC guidance. [108]

Figure 7-13: Conceptual Representation of Inadvertent Intruder-Agricultural Scenario (10 CFR 61.42)

(Based on information in the PODD)

INTRUDER - AGRICULTURAL SCENARIO
(Assuming well drilled through abandoned transfer line)

1. Direct Radiation from Drill Cuttings in Garden
2. Internal Radiation from Ingesting Produce from Garden
3. Internal Radiation from Ingesting Soil in Garden
4. Inhalation of Resuspended Soil
5. Well-Water from Congaree Aquifer Drilled through Abandoned Tank Farm Transfer Line



[Not to Scale - Conceptual Only]

7.2.3.3 Mathematical Model

The Agricultural-Intruder analysis was performed using an automated inadvertent intruder analysis code known as the Automated Intruder Analysis Application. This computer code was developed by SRNL for the calculation of intruder doses based upon dose conversion factors and specific input parameters that are applied by the user. The code can account for such parameters as radionuclide decay, decay product in-growth, and shield thickness degradation while determining intruder doses at specified intervals over time. [109]

The Automated Intruder Analysis Application was set to perform the intruder analysis for ten-year intervals from an initial start time 100 years after closure of Tank 19 and 18 to 10,000 years post-closure. The intruder analysis start time is based on an assumed 100-year institutional control period, and a 10,000-year evaluation period for post-closure performance is selected to find the maximum contribution from each radionuclide of the assessed inventory (i.e., ensure that long-lived radionuclides are adequately addressed in determining estimated peak doses). The inventory, applied input parameters, and modeling assumptions (e.g., dose conversion factors, human health exposure parameters, transport factors, etc.) can be found in the PODD.

7.2.3.4 Results

The results of the intruder modeling were analyzed and the maximum estimated dose for the evaluated Intruder-Agricultural scenario between 100 years and 10,000 years after tank closure was determined to be 120 mrem/yr TEDE. This maximum dose occurred at the peak dose year 2120, which is immediately following loss of institutional control at 100 years after the projected closure date (2020). The major contributors to the maximum estimated intruder dose were Sr-90, due largely to internal radiation exposure. The total annual dose was shown to rapidly decline over time for the evaluated intrusion period being 47 mrem/yr at 155 years post-closure, 10 mrem/yr at 225 years post-closure, and below 1 mrem/yr by 515 years post-closure. This total dose is well within the performance objective of 500 mrem for all times after closure. [104]

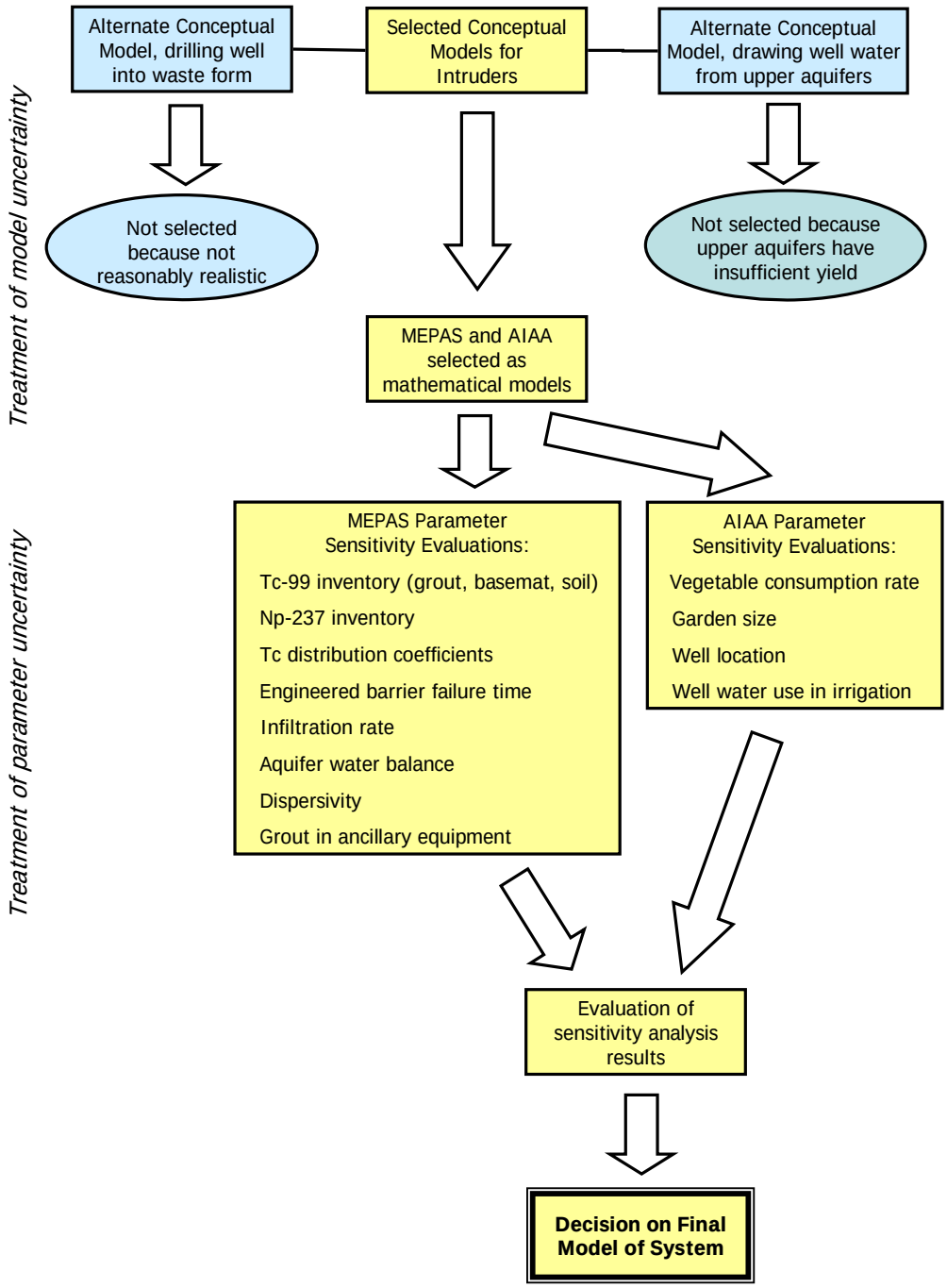
Complete results of the Intruder-Agricultural analysis using the Automated Intruder Analysis Application are documented in the PODD.

7.2.3.5 Sensitivity Studies Related to Intruder Analysis

As explained in the PODD, uncertainty analyses were performed related to modeling the inadvertent intrusion, similar to those performed in evaluating potential doses to a member of the public with institutional controls in place. It is concluded through an evaluation of the results of these analyses that the base case parameters used were appropriate and the results provide reasonable assurance of protection for inadvertent intruders.

Figure 7-14 below illustrates the applied uncertainty analysis process.

Figure 7-14: Uncertainty Analysis Process for Inadvertent Model



Four input parameters associated with the Intruder-Agricultural scenario were performed: (1) vegetable consumption rate, (2) garden size, (3) well location, and (4) use of well water for irrigation purposes. These analyses showed that:

- Estimated doses vary linearly in inverse proportion to garden area,
- Doubling vegetable consumption would increase dose by approximately 94%,
- Maximum potential doses from a well located 100 meters hydraulically downgradient from FTF would be approximately 25% of potential maximum doses from the well located one-meter downgradient, and
- Use of water from the one-meter well to irrigate the garden would add approximately four mrem to peak estimated annual dose.

7.2.3.6 Conclusions

The area around Tank 19 and Tank 18 will be controlled according to the SRS Land Use Plan. As such there is reasonable assurance individuals will be protected from the radiation exposure resulting from an inadvertent intrusion into the area in that the SRS Land Use Plan requires Federal ownership and control of the site well beyond 100 years after closure of each tank. [92] Moreover, DOE 5400.5 precludes release of the area unless the radiological hazard meets the requirements of DOE 5400.5 Chapter 4, which requires virtual perpetual DOE control. Also, no unrestricted use of the land or groundwater will be permitted for the central portion of the site, which includes Tank 19 and Tank 18. [92]

In the unlikely event there were inadvertent human intrusions, the resulting doses would be low. Grouted Tank 19 and grouted Tank 18 will protect any intruder from inadvertently gaining access to the tanks and contacting the waste form following the period of active institutional controls. As explained in the PODD, the 2,000-psi grout in the tank dome can be considered to be a man-made sedimentary rock-like material. It is not possible to drill through such material with the standard well drilling equipment used to penetrate soils in the SRS area. Based on grout studies performed after the closure of Tanks 17 and 20, the reducing grout in Tank 19 and Tank 18 will also act as an intruder barrier with all the same physical and chemical properties as the 2,000-psi grout in the tank dome. [71]

The structural integrity of Tank 19 and Tank 18 will persist well into the future as a solid monolith of grout. [75] The grout therefore provides reasonable assurance that any inadvertent intruder would be protected from significant radiation exposure.

Based on the results of the analyses described in this section, there is reasonable assurance that the disposal of the solidified Tank 19 residual waste and solidified Tank 18 residual waste will meet the limits of the NRC performance objective specified in 10 CFR 61.42.

7.2.4 Section 61.43, Protection Of Individuals During Operations

10 CFR 61.43 states:

> Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in part 20 of this chapter, except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by Section 61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures ALARA.

The performance objective in 10 CFR 61.43 cross-references "the standards for radiation protection in Part 20". Consistent with Section 3116 (a) of the NDAA, the cross-referenced "standards for radiation protection" in 10 CFR Part 20 that are considered in detail in this Draft 3116 Determination are the dose limits for the public and the workers during disposal operations set forth in 10 CFR 20.1101(d), 10 CFR 20.1201(a)(1)(i), 10 CFR 20.1201(a)(1)(ii), 10 CFR 20.1201(a)(2)(i), 10 CFR 20.1201(a)(2)(ii), 10 CFR 20.1201(e), 10 CFR 20.1208(a), 10 CFR 20.1301(a)(1), 10 CFR 20.1301(a)(2), and 10 CFR 20.1301(b)[19]. As will be discussed in the following sections, these dose limits correspond to the dose limits in 10 CFR Part 835 and relevant DOE Orders which establish DOE regulatory and contractual requirements for DOE facilities and activities. The following subsections show that disposal of Tank 19 and Tank 18 residual waste meets these dose limits and that doses will be maintained ALARA[20].

[19] The introductory "notwithstanding" phrase to Section 3116 makes it clear that the provisions of Section 3116 (a) are to apply in lieu of other laws that "define classes of radioactive waste". As is evident from the plain language of this introductory "notwithstanding" phrase, Section 3116 (a) pertains to classification and disposal, and radiation protection standards for disposal, of certain waste at certain DOE sites. Thus, the factors for consideration set forth in subsection (a) 1 through (3) of Section 3116 are those which pertain to classification and disposal of waste, and the radiation protection standards for disposal. The Joint Explanatory Statement of the Committee of Conference in Conference Report 108-767, accompanying H.R. 4200 (the NDAA), also confirms that 3116 (a) concerns classification, disposal, and radiation protection standards associated with disposal, and does not concern general environmental laws or laws regulating radioactive waste for purposes other than disposal. Moreover, in the plain language of Section 3116, Congress directed that the Secretary of Energy consult with the NRC but did not mandate that DOE obtain a license or any other authorization from NRC, and did not grant NRC any general regulatory, administrative, or enforcement authority for disposal of the DOE wastes covered by Section 3116. As such, the "standards for radiation protection" in 10 CFR Part 20 (as cross-referenced in the performance objective at 10 CFR 61.43) which are relevant in the context of Section 3116 of the NDAA are the dose limits for radiation protection of the public and the workers during disposal operations, and not those which address general licensing, administrative, programmatic, or enforcement matters administered by NRC for NRC licensees. Accordingly, this Draft 3116 Determination addresses in detail the radiation dose limits for the public and the workers during disposal operations that are contained in the provisions of 10 CFR Part 20 referenced above. Although 10 CFR 20.1206(e) contains limits for planned special exposures for adult workers, there will not be any such planned special exposures for disposal operations at FTF for the Tank 19 or Tank 18 waste. Therefore, this limit is not discussed further in this Draft 3116 Determination. Likewise, 10 CFR 20.1207 specifies occupational dose limits for minors. However, there will not be minors working at FTF who will receive an occupational dose. Therefore, this limit is not discussed further in this Draft 3116 Determination.

[20] In addition, 10 CFR Part 835, like Part 20 for NRC licensees, includes requirements that do not set dose limits, such as requirements for radiation protection programs, monitoring, entrance controls for radiation areas, posting, records, reporting, and training.

7.2.4.1 Air Emissions Limit for Individual Member of the Public (10 CFR 20.1101(d))

The NRC regulation at 10 CFR 20.1101(d) provides in relevant part:

> [A] constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established … such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions.

DOE similarly limits doses from air emissions to the public to 10 mrem/yr in DOE Order 5400.5. DOE is also subject to and complies with the EPA requirement in 40 CFR 61.92, which has the same limit of 10 mrem/yr.[21] The estimated dose per year from airborne emissions to the maximally exposed individual member of the public located at or beyond the SRS site boundary from all operations at SRS ranged from 0.04 mrem to 0.07 mrem from 1999 through 2003. [110],[111],[112],[113],[114].[22] These values (0.04 mrem to 0.07 mrem from 1999 to 2003) for all of the SRS operations (not just disposal operations at FTF) are well below the dose limit specified in 10 CFR 20.1101(d) of 10 mrem (0.1mSV/yr). The dose from air emissions from closure of Tank 19 and Tank 18 will be a negligible fraction of the dose from all operations at SRS.

7.2.4.2 Total Effective Dose Equivalent Limit for Adult Workers (10 CFR 20.1201(a)(1)(i))

The NRC regulation at 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.
>
> (1) An annual limit, which is the more limiting of –
>
> (i) The total effective dose equivalent being equal to 5 rems (0.05 Sv).
>
> ***

DOE's regulation at 10 CFR 835.202 (a)(1) has the same annual dose limit for the annual occupational dose to general employees[23]. For the occupational dose to adults during closure of

[21] 40 CFR 61.92 provides as follows: Emissions of radionuclides to the ambient air from DOE facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem/yr.

[22] It is assumed that the individual is an adult living at the site perimeter who is exposed to the maximum yearly radioactive atmospheric release and maximum radiation concentration in food for 365 days per year. For the airborne pathway the dose is developed by the input of atmospheric release data, vegetation consumption data, milk consumption data, and beef consumption data.

[23] DOE's regulation requires that the occupational dose per year for general employees shall not exceed both a total effective dose equivalent of 5 rems and the sum of the deep dose equivalent for external exposures and the committed dose equivalent to any other organ or tissue other than the lens of the eye of 50 rem. NRC's regulation specifies that either of these two limits shall be met by NRC licensees, whichever is more limiting. This Draft 3116 Determination will show that DOE will meet the more stringent of the dose limits in 10 CFR Part 835 and the relevant dose limits in 10 CFR Part 20 (see subsection 7.2.3 of this Draft 3116 Determination for listing of relevant dose limits in 10 CFR Part 20). Because DOE imposes stricter, separate requirements, the provisions of 10 CFR 20.1201(a)(1) and (a)(2), which correlate to 10 CFR 835.202(a)(1) and 10 CFR 835.202(a)(2)), are discussed in separate subsections in this Draft 3116 Determination.

Tank 19 and Tank 18, the TEDE per year will be controlled using the principles of ALARA, and will be below 5 rem. [115] In this regard, occupational doses to workers have been well within the annual limits specified in 10 CFR 1201 (2) for all work activities at SRS. The highest dose received by an SRS worker in 2003 was 1,808 mrem TEDE. [116] Since 1998, the highest dose received by an SRS worker has been at or below 1,808 mrem/yr. The highest total dose received by an FTF worker from 1995 – 2004 was 545 mrem. [117] Furthermore, operations will consist predominantly of running a clean grout batch plant, and mechanical transfer of grout to Tank 19 and Tank 18; therefore, limited hands-on work is anticipated. All work performed at SRS and FTF will be performed in accordance with the Administrative Control Program 5.8.2.1, *Radiological Protection Program*, of the CSTF Technical Safety Requirements (TSR) as further discussed in Section 7.2.4.14 of this document. [118] Thus, the total effective dose equivalent to workers from closure of Tank 19 and Tank 18 is expected to remain well below the NRC limit, based on the highest dose received by an FTF worker in 1997 was 215 mrem. Since Tank 17 and Tank 20 were closed in 1997, this dose is of particular interest in addressing the dose potential for a worker during closure operations[24]. Given that the highest dose for the entire FTF was 215 mrem in 1997 provides assurance that doses received by a worker during closure activities will be extremely low. This expectation is further substantiated considering that maximum worker dose in 1997 was not the highest total dose recorded from 1997 to 2004 (545 mrem). [117]

7.2.4.3 Any Individual Organ or Tissue Dose Limit for Adult Workers (10 CFR 20.1201(a)(1)(ii))

The NRC regulation at 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a)[C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.
>
> (1) An annual limit, which is the more limiting of –
>
> ***
>
> (ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

The dose limit specified in 10 CFR 20.1201(a)(1)(ii) is the same as that specified in 10 CFR 835.202 (a)(2). For the occupational dose to adults during closure of Tank 19 and Tank 18, the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye will be controlled to ALARA, below a maximum of 50 rem/yr. SRS Engineering Standard 01064, *Radiological Design Requirements*, provides the design basis annual occupational exposure limits for any organ or tissue other than the eye cannot exceed 10 rem/yr, which is well below the NRC limit of 50 rem/yr. [115],[119]

Furthermore, operations will consist predominantly of running a clean grout batch plant outside the boundary of FTF and mechanical transfer of clean grout to Tank 19 and Tank 18; therefore, limited hands-on work is anticipated, with limited associated exposure anticipated for Tank 19

[24] Specific worker doses for Tank 17 and Tank 20 closure activities were not tabulated in 1997. However, worker doses for Tank 19 and Tank 18 will be tracked.

and Tank 18 disposal operations controlled under the Radiological Protection Program of the CSTF TSR Administrative Control Program, 5.8.2.1 (see 7.2.4.2). [118]

7.2.4.4 Annual Dose Limit to the Lens of the Eye for Adult Workers (10 CFR 20.1201(a)(2)(i))

The NRC regulation at 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.
>
> ***
>
> (2) The annual limits to the lens of the eye, to the skin of the whole body or to the skin of the extremities, which are:
>
> (i) A lens dose equivalent of 15 rems (0.15 Sv)

The dose limit specified in 10 CFR 20.1201(a)(2)(i) is the same as that specified in DOE's regulation at 10 CFR 835.202 (a)(3). For the occupational dose to adults during closure of Tank 19 and Tank 18, the annual dose limit to the lens of the eye will be controlled using the principles of ALARA, and will be below 15 rem/yr. SRS Engineering Standard 01064, *Radiological Design Requirements*, provides the design basis annual occupational exposure limits for the lens of the eye cannot exceed 3 rem/yr, which is well below the NRC limit of 15 rem/yr. [115],[119]

Furthermore, operations will consist predominantly of running a clean grout batch plant outside the boundary of FTF, and mechanical transfer of clean grout to Tank 19 and Tank 18; therefore, limited hands-on work is anticipated for Tank 19 and Tank 18 disposal operations controlled under the Radiological Protection Program of the CSTF TSR Administrative Control Program, 5.8.2.1 (see 7.2.4.2). [118]

7.2.4.5 Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers (10 CFR 20.1201(a)(2)(ii))

The NRC regulation at 10 CFR 20.1201(a), concerning occupational dose limits for adults, provides in relevant part:

> (a) [C]ontrol the occupational dose to individual adults, except for planned special exposures…to the following dose limits.
>
> ***
>
> (2) The annual limits to the lens of the eye, the skin of the whole body, or to the skin of the extremities, which are:
>
> ***
>
> (ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

This NRC dose limit specified in 10 CFR 20.1201(a)(2)(ii) is the same as the DOE dose limit specified in 10 CFR 835.202 (a)(4). For the occupational dose to adults during closure of Tank

19 and Tank 18, which involve limited hands-on activity, the annual dose limit to the skin of the whole body or to the skin of any extremity will be controlled using the principles of ALARA, and will be below a shallow-dose equivalent of 50 rem/yr as prescribed by the WSRC 5Q Radiological Control Manual (see 7.2.4.2). [115]

7.2.4.6 Limit on Soluble Uranium Intake (10 CFR 20.1201(e))

The NRC regulation at 10 CFR 20.1201(e), concerning occupational dose limits for adults, provides in relevant part:

> (e) In addition to the annual dose limits, … limit the soluble uranium intake by an individual to 10 milligrams in a week in consideration of chemical toxicity [.]

In addition to the annual dose limits to adults during closure of Tank 19 and Tank 18, the soluble uranium intake by an individual is controlled to less than 10 milligrams (mg) per week. DOE Order 440.1A requirements for soluble uranium intake are the more restrictive of the concentrations in the American Conference of Governmental Industrial Hygienists Threshold Limit Values (0.2 mg per cubic meter which is the same as noted in 10 CFR 20 Appendix B footnote 3) or the Occupational Safety and Health Administration (OSHA) Permissible Exposure Limit (PEL) (0.05 mg per cubic meter). [120] The OSHA PEL limit for soluble uranium, which equates to a soluble uranium intake of 2.4 mg per week, is the more restrictive of the two and therefore is the limit imposed for closure of Tank 19 and Tank 18. Accordingly, the soluble uranium intake, if any, during closure of Tank 19 and Tank 18 will be controlled to 2.4 mg per week, which is below the NRC limit in 10 CFR 20.1201(e). WSRC implements DOE Order 440.1A through the WSRC 4Q Industrial Hygiene Manual. [121]

Furthermore, operations will consist predominantly of running a clean grout batch plant, and mechanical transfer of grout to Tank 19 and Tank 18; therefore, limited hands-on work is anticipated, with limited associated exposure anticipated for Tank 19 and Tank 18 disposal operations controlled under the Radiological Protection Program of the CSTF TSR Administrative Control Program, 5.8.2.1. [118]

7.2.4.7 Dose Equivalent to an Embryo/Fetus (10 CFR 20.1208(a))

The NRC regulation at 10 CFR 20.1208(a), concerning the dose equivalent to an embryo/fetus, provides in relevant part:

> (a) [E]nsure that the dose equivalent to the embryo/fetus during the entire pregnancy, due to the occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv).

DOE's regulation at 10 CFR 835.206 (a) has the same dose limit. For the occupational dose to an embryo/fetus during closure of Tank 19 and Tank 18, doses will be controlled so the dose equivalent to the embryo/fetus during the entire pregnancy for a declared pregnant worker will not exceed 0.5 rem as controlled through the WSRC 5Q Radiological Control Manual. [115] Furthermore, after declaration of pregnancy, DOE provides the option of a mutually agreeable assignment of work tasks, without loss of pay or promotional opportunity, such that further occupational radiation exposure during the remainder of the gestation period is unlikely. [115] In addition, personnel dosimetry is provided and used to carefully track exposure as controlled by WSRC 5Q Radiological Control Manual. [115]

7.2.4.8 Dose Limits for Individual Members of the Public (10 CFR 20.1301(a)(1))

The NRC regulation at 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

(a) [C]onduct operations so that -

(1) The total effective dose equivalent to individual members of the public …does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released…., from voluntary participation in medical research programs, and from the …disposal of radioactive material into sanitary sewerage [.]

DOE Order 5400.5 II.1.a similarly limits public doses to less than 100 mrem/yr. However, DOE's application of the limit is more restrictive in that it requires DOE to make a reasonable effort to ensure multiple sources (e.g., DOE sources and NRC regulated sources) do not combine to cause the limit to be exceeded. For individual members of the public during closure of Tank 19 and Tank 18, the TEDE limit to an individual member of the public will be controlled to less than 0.1 rem/yr as controlled by the WSRC 5Q Radiological Control Manual. [115] The air pathway is the predominant pathway for doses to the public from SRS operations, including closure of Tank 19 and Tank 18, and, as discussed in subsection 7.2.3.1 of this Draft 3116 Determination, doses from the air pathway to members of the public have been, and are expected to continue to be, well below the 0.1 rem annual limit specified in 10 CFR 20.1301(a). [110],[111],[112],[113],[114],[115]

7.2.4.9 Dose Limits for Individual Members of the Public (10 CFR 20.1301(a)(2))

The NRC regulation at 10 CFR 20.1301(a), concerning dose limits for individual members of the public, provides in relevant part:

(a) [C]onduct operations so that –

* * *

(2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released …, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

DOE's regulation at 10 CFR 835.602 establishes the expectation the TEDE in Controlled Areas will be less than 0.1 rem in a year. For individual members of the public during closure of Tank 19 and Tank 18, operations will be conducted such that the dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material, will be less than 0.00005 rem per hour above background. WSRC 5Q Radiological Control Manual, Chapter 2, Article 232, also restricts the TEDE in Controlled Areas to less than 0.1 rem in a year. To ensure these dose limits are met, the following measures have been instituted within Controlled Areas. Per 10 CFR 835.603(g), Radioactive Materials Areas have been established for accumulations of radioactive material that could result in a radiation dose of 100 mrem in a year or greater. In addition, SRS has established Radiological Buffer Areas (RBAs) around posted Radiological Areas. Standard SRS practice is to assume a 2,000 hour per year continuous occupancy at the outer boundary of these areas; therefore, the dose rate at a

RBA boundary is 0.05 mrem/hr (0.1 rem/2,000 hr = 0.00005 rem/hr). Since the Controlled Area encompasses a RBA, it is ensured the dose in the Controlled Area (but outside of Radioactive Material Areas and RBA) will be less than 0.1 rem in a year. [115] Therefore, SRS implementation of the provisions at 10 CFR 835.602 and 10 CFR 835.603 provides a more restrictive limit than the dose limit specified in 10 CFR 20.1301(a)(2). Furthermore, training is required for individual members of the public for entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits, use of dosimetry is required if a member of the public is expected to enter a controlled area and receive a dose that may exceed 0.05 rem/yr[25].

7.2.4.10 Dose Limits for Individual Members of the Public (10 CFR 20.1301(b))

The NRC regulation at 10 CFR 20.1301(b), concerning dose limits for individual members of the public, provides in relevant part:

> (b) If … members of the public [are permitted] to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

DOE's regulation at 10 CFR 835.208 has the same dose limit. The TEDE limit to an individual member of the public granted access to controlled areas during closure of Tank 19 and Tank 18 will be controlled to 0.1 rem/yr. [115] Furthermore, training is required for individual members of the public for entry into controlled areas. In addition, to ensure no member of the public exceeds radiation exposure limits, use of dosimetry is required if a member of the public is expected to enter a controlled area and receive a dose that may exceed 0.05 rem/yr[26].

7.2.4.11 As Low As Reasonably Achievable (ALARA) (10 CFR 20.1003)

The NRC regulation at 10 CFR 20.1003 defines ALARA in relevant part:

> ALARA … means making every reasonable effort to maintain exposures to radiation as far below the dose limits … as is practical consistent with the purpose for which the … activity is undertaken…[.]

DOE has a similar requirement, and DOE's regulation at 10 CFR 835.2 defines ALARA as "… the approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as reasonable…". For radiological work activities during closure of Tank 19 and Tank 18, every reasonable effort will be made to maintain exposures to radiation as far below the dose limits as is practical consistent with the purpose for which the activity is undertaken. Furthermore, DOE's regulation at 10 CFR 835.101(c) requires the contents of each Radiation Protection Program (RPP) to include formal plans and measure for applying the ALARA process to occupational exposure as further discussed in Section 7.2.4.13 of this document. [122]

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[25] 10 CFR 20.1003 defines restricted areas as an area, access to which is limited … for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. This is the same as the definition in 10 CFR 835.2 for a controlled area.

[26] 10 CFR 20.1301(d) allows licensees to request NRC authorization to allow an individual member of the public to operate up to an annual dose limit of 0.5 rem (5 mSv). 10 CFR 835 is more restrictive for the dose to an individual member of the public with a limit of 0.1 rem maximum annual dose as discussed in Subsection 7.2.4.8. WSRC protects this limit of 0.1 rem maximum annual dose by limiting the maximum dose to a member of the public to 0.05 rem/yr. Therefore, this limit is not discussed further in this waste determination.

7.2.4.12 Reasonable Assurance

Measures that provide reasonable assurance that closure of Tank 19 and Tank 18 will comply with the applicable dose limits and with the ALARA provisions include: (1) the documented RPP, [122] which is further discussed below in Section 7.2.4.13; (2) the DSA (further discussed in Section 7.2.4.14); (3) design (further discussed in Section 7.2.4.15); (4) regulatory and contractual enforcement mechanisms (further discussed in Section 7.2.4.16); and (5) access controls, training, and dosimetry (further discussed in Section 7.2.4.17). In addition, the following discusses the exposure history at SRS and at FTF.

7.2.4.13 SRS Radiation Protection Program

The DOE regulates occupational radiation exposure at its facilities through 10 CFR Part 835, *Occupational Radiation Protection*. [126] Part 835 establishes exposure limits and other requirements to ensure DOE facilities are operated in a manner such that occupational exposure to workers is maintained within acceptable limits and as far below these limits as is reasonably achievable. The requirements in Part 835 are nuclear safety requirements which, if violated, provide a basis for the assessment of civil penalties under the section 234A of the Atomic Energy Act of 1954, as amended (Public Law 83-703).

Pursuant to Part 835, activities at SRS, including disposal operations at FTF, must be conducted in compliance with the documented RPP for SRS as approved by DOE. The key elements of the RPP include monitoring of individuals and work areas, control of access to areas containing radiation and radioactive materials, use of warning signs and labels, methods to control the spread of radioactive contamination, radiation safety training, objectives for the design of facilities, criteria for levels of radiation and radioactive material in the workplace, and continually updated records to document compliance with the provisions of Part 835. The RPP also includes formal plans and measures for applying the ALARA process. [122]

The requirements of Part 835 as contained in the approved RPP are incorporated in WSRC's Standards/Requirements Identification Document system (SRIDs). The SRID system links the requirements of Part 835 to the site-level and lower-level implementing policies and procedures that control radiological work activities conducted across the site. These requirements are primarily contained in the WSRC 5Q Radiological Control Manual, and its lower tier manuals, e.g., WSRC 5Q1.1, *Radiation and Contamination Control Procedures Manual*, and WSRC 5Q1.2, *Radiation Monitoring Procedures Manual*. These procedures control the planning of radiological work, the use of radiation monitoring devices by employees, the bioassay program, the air monitoring program, the contamination control program, the ALARA program, the training of general employees, radiological workers, Radiological Control Inspectors, and health physics professionals and technicians, and the other aspects of an occupational RPP as required by Part 835.

7.2.4.14 Documented Safety Analysis (DSA)

A DSA has been approved by DOE for operation of FTF (including Tank 19 and Tank 18), HTF, and 299-H maintenance facilities in accordance with 10 CFR Part 830. FTF, HTF, and 299-H facilities are collectively referred to hereafter as CSTF. [39] As the first step in the development of the DSA, a formal Hazard Analysis (HA) was performed as the initial analytical effort and

systematically presents an analysis of potential process-related, Natural Phenomena Hazards (NPH), and external hazards associated with CSTF that can affect the public, the workers, and the environment due to a single or multiple failures. [124] The HA was performed by a group of subject matter experts, with expertise in the fields of operations, engineering, industrial hygiene, radiological protection, environmental compliance, and maintenance. [125]

The HA consisted of three basic phases: hazard identification, hazard classification, and hazard evaluation. [124] During the hazard identification phase, all possible radiological and chemical hazardous materials associated with the normal and abnormal operations of the facility were identified, along with all potential energy sources available to disperse the hazardous materials to the environment. [125]

During the hazard classification phase, the maximum quantities of hazardous materials possible in CSTF are evaluated against the criterion listed in DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, to determine the overall hazard classification of the facility.

During the third and final phase of the HA, all possible normal and abnormal operational events that could result in exposing facility workers or the public to hazardous material were evaluated to determine the magnitude of the risk. During the hazard evaluation phase, the consequence and frequency of each operational event was qualitatively determined, and the resulting level of risk identified. The purpose of identifying the level of risk was to determine which operational events posed some level of risk (and thus required additional evaluation) and those events which presented negligible risk to the facility workers and public. Additionally, for those waste tanks and operational facilities that have reduced their curie inventories to less than a total effective inhalation dose potential of less than or equal to 1.0E+11 rem, a new mode of operation was determined, Removed From Service, as defined in the CSTF DSA, Chapter 5. [39]

Tank 19 and Tank 18 are classified as “removed from service.” In addition to their limited total effective inhalation dose potential, administrative controls and engineering controls are placed on these waste tanks. Administrative controls include prohibiting transfers into and out of these waste tanks, and limit the amount of liquid additions to those necessary to: flushing equipment, maintaining tank chemistry, maintaining tank ballast, prevent exposed solids on the floor of the waste tank from drying to the point that they pose special airborne release concerns, and achieving waste immobilization for closure (e.g., grouting). Administrative controls also include prohibiting the operation of waste tank mixing devices, and placing transfer lines and their secondary containments going into and out of these waste tanks Out-of-Service. Engineering controls include physically isolating air, steam, and BW lines going to these waste tanks and physical isolation of pressurized sources to any mixer/pump column. All cooling water supply and return to the waste tank shall be physically isolated. [39]

The DSA analyzed the hazards identified in the HA that could impact the offsite public and facility workers during normal operations and accident conditions. The DSA identified the basis for derivation of the CSTF TSR and also discussed summary descriptions of the key features of safety management programs at SRS as they pertain to the CSTF. [39],[118]

The CSTF TSR document identified controls necessary to achieve safe operation at the CST facilities. In part, these TSR administrative controls require: (1) a facility manager be assigned

who is accountable for safe operation and in command of activities necessary to maintain safe operation, (2) personnel who carry out radiological controls functions for CSTF have sufficient organizational freedom to ensure independence from operating pressure, (3) CSTF personnel receive initial and continuing training including radiological control training, and (4) an RPP shall be prepared consistent with 10 CFR Part 835. [118] In addition, the design requirements for CSTF implemented 10 CFR Part 835 and, in particular, implemented ALARA principles.

7.2.4.15 Radiological Design for Protection of Occupational Workers and the Public

New SRS radiological facilities and facility modifications are designed to meet the requirements of 10 CFR 835 Subpart K, Design and Control. SRS Engineering Standard 01064, Radiological Design Requirements provides the requirements necessary to ensure compliance with 10 CFR Part 835. [126],[119] The standard refers to 10 CFR Part 835, DOE Orders, DOE Standards, DOE handbooks, national consensus standards, SRS manuals, SRS engineering standards, SRS engineering guides, and site operating experience in order to meet the 10 CFR Part 835 specific requirements and additional requirements to ensure the design provides for protection of the worker and the environment.

The standard covers the full spectrum of radiological design requirements and not just radiation exposure limits. The following are the specific areas addressed in the standard: radiation exposure limits; facility and equipment layout; area radiation levels; radiation shielding; internal radiation exposure; radiological monitoring; confinement; and ventilation. [119]

The design requirements for several of the important sections of the standard are highlighted in order to understand the design limits and philosophy for SRS designs. The first area of interest is the radiation exposure limits. The following is an excerpt of the standard which presents the exposure limits and philosophy for both external and internal radiation exposure.

> "During the design of new facilities or modification of existing facilities, the design objective for controlling personnel exposure from external sources of radiation in areas of continuous occupancy (2000 hours per year) shall be to maintain exposure levels below an average of 0.5 mrem per hour and as far below this average as is reasonably achievable. The design objectives for exposure rates for potential exposure to a radiological worker where occupancy differs from the above shall be ALARA and shall not exceed the external limits in Table 5-1. Regarding the control of airborne radioactive material, the design objective shall be, under normal conditions, to avoid releases to the workplace atmosphere and in any situation, to control the inhalation of such material by workers to levels that are ALARA; confinement and ventilation shall normally be used [6.3, 6.12]. Table 5-1 summarizes the design basis external radiation exposure limits.

Design Basis Annual Occupational Radiation Exposure Limits

Type of Exposure	Limit (rem)
Whole Body TEDE	1.0
Internal CEDE	0.5
Lens of Eye	3
Extremity	10
Any Organ (other than eye) or Tissue	10

To meet the Site's no deliberate intake policy, engineered controls will be evaluated and implemented to ensure, under normal operating conditions, no worker will receive a deliberate intake of radionuclides (i.e., committed effective dose equivalent (CEDE) = 0 rem). As a result, the TEDE limit will be independent of the CEDE limit. The 0.5 rem CEDE limit in Table 5-1 is to be applied to potential intakes from anticipated potential releases or anticipated off-normal maintenance. Under these anticipated potential conditions, engineered controls will be evaluated and implemented to minimize the potential for workers to receive intakes that will exceed the 0.5 rem CEDE. This evaluation will not take credit for the use of respiratory protection.

The dose to any member of the public or a minor exposed to radiation at a DOE facility shall not exceed 0.1 rem TEDE in a year."

The facility design also incorporates radiation zoning criteria in order to ensure the exposure limits presented above are met by providing adequate radiation shielding. Areas in which non-radiological workers are present are assumed to have continuous occupancy (2,000 hours per year) and are designed to a dose rate less than 0.05 mrem per hour to ensure the annual dose is less than 100 mrem. [119] Other zoning criteria are established to ensure radiological worker doses are ALARA and less than 1,000 mrem/yr to meet the 10 CFR 835.1002 design requirements.

The design is also required to provide necessary radiological monitoring or sampling for airborne and surface contamination to ensure the engineered controls are performing their function and, in the event of a failure or upset condition, workers are warned and exposures avoided. [119]

Radiological protection personnel ensure the requirements of the standard are addressed and presented in design summary documentation. The incorporation of all the radiological design criteria in the engineering standard ensures the requirements of 10 CFR Part 835 are met and the design provides for the radiological safety of the workers and environment. [119]

7.2.4.16 Regulatory and Contractual Enforcement

Any violation of the requirements in 10 CFR Part 835 is subject to civil penalties pursuant to section 234A of the Atomic Energy Act of 1954, as amended, 42 USC 2011 et seq., as implemented by DOE regulations in 10 CFR Part 820. In addition, the requirements in 10 CFR Part 835 and all applicable DOE Orders are incorporated into all contracts with DOE contractors, including WSRC, the DOE contractor for disposal operations at Tank 19 and Tank 18 as well as other operations at SRS. DOE enforces these contractual requirements through contract enforcement measures, including the reduction of contract fees.

7.2.4.17 Access Controls, Training, Dosimetry, and Monitoring

Training or an escort is required for individual members of the public for entry into controlled areas. In addition, use of dosimetry is required if a member of the public is expected to enter a controlled area and exceed 0.05 rem/yr to ensure no member of the public exceeds radiation exposure limits. [115]

In addition, worker radiation exposure monitoring is performed for all workers expected to receive 100 mrem/yr from internal and external sources of radiation to provide assurance no worker exceeds radiation exposure limits and all radiation dose are maintained as far below the limits as is reasonably achievable. [115]

7.2.4.18 Occupational Radiation Exposure History for Savannah River Site

The effectiveness of the RPPs, including the effectiveness of oversight programs to ensure they are implemented properly is demonstrated by the occupational radiation exposure results as documented for 2003. [116] The highest dose received by an SRS worker in 2003 was 1,808 mrem TEDE compared to the DOE Administrative Control Limit of 2,000 mrem/yr and the 10 CFR Part 835 limit of 5,000 mrem/yr.

In addition, for all work activities, the average TEDE exposure for workers receiving a TEDE dose at SRS has been 75 mrem/yr or less since 2001. [127] The highest total dose received by an FTF worker from 1995 – 2004 was 545 mrem. [117] It is expected that exposures for workers at FTF will be at or below this average based on design requirements based on discussions from Sections 7.2.4.1 through 7.2.4.17 above , and the CSTF DSA, TSR, WSRC 5Q Radiological Control Manual, WSRC 4Q Industrial Hygiene Manual, and past experience with FTF operations. [39],[118],[115],[121]

7.2.5 Section 61.44, Stability Of The Disposal Site After Closure

10 CFR 61.44 states:

> The disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required.

As discussed in this section, operational closure of the solidified Tank 19 and Tank 18 waste and tanks in situ will meet the performance objectives of 10 CFR 61.44 for long-term stability of the disposal site.

7.2.5.1 Siting

A comprehensive review of site geology, seismology, hydrology, meteorology, and environmental setting are presented in Section 2 of the PODD, which is briefly summarized in this subsection. [4] The siting of both Tank 19 and Tank 18 is such that it provides long-term stability consistent with this performance objective. Favorable siting features of Tank 19 and Tank 18 are that both are located in FTF, located in the north-central portion of the SRS, bounded by Upper Three Runs Creek to the north, Fourmile Branch to the south, and McQueens Branch to the east. Land use within the F- and H-Areas is classified as heavy industrial. [92]

This area is entirely within the SRS boundaries and is used for either industrial purposes or as forested land. SRS will continue under federal control with restricted recreational and industrial/maintenance worker use, with no residential use. [4]

Seismic activity associated within the entire SRS is closely related to the South Carolina Piedmont type activity. This activity is characterized by occasional small, shallow events associated with strain release near small scale faults, intrusive bodies, and the edges of metamorphic belts, however, the Charleston area is the most significant seismic zone affecting the SRS. Seismic history for the southeast covers a period of approximately 300 years, with the most significant earthquake event being that of Charleston 1886. Section 2 of the PODD contains additional information on historical seismic events and earthquakes as well as identification of faults in the SRS vicinity. [4]

The base of Tank 19 is 228-feet above MSL, and the Tank 18 base is 229-feet above MSL. [4] The Probable Maximum Flood value is 175-feet above MSL and the 500-year and 100-year flood plains are 153.1-feet and 151.7-feet above MSL, respectively thereby placing the tanks well above the expected flood levels. Section 2 of the PODD discusses the surface water flows, characteristics and controls for the major water ways that impact the SRS. [77]

F-Area is on a groundwater divide between Upper Three Runs and Fourmile Branch. The average depth to groundwater in the FTF is approximately 32-feet. Because the tanks are partially buried, the depths to the water table below the tanks are approximately five-feet. Section 2 of the PODD contains additional information on the stratigraphy and hydrogeology of the area underneath the tanks that impacts the performance of the closure. [4]

A more extensive discussion in the PODD provides details on the soil types both regionally and locally surrounding Tank 19 and Tank 18. The SRS lies within the Aiken Plateau, and this Plateau slopes to the southeast approximately five-feet per mile. The Plateau is bounded by the

Savannah and Congaree Rivers and extends from the fall line to the Orangeburg Escarpment. The highly dissected surface of the Aiken Plateau is characterized by broad interfluvial areas with narrow, steep-sided valleys. Local relief can be as much as 300-feet. The soils in the vicinity of Tank 19 and Tank 18 are well drained soils that formed in heterogeneous materials, which are the spoil or refuse from excavations and major construction operations. The soils range from sandy to clayey, depending on source of the material or geological parent. There are also areas that consist of cross-bedded, poorly sorted sand with lenses and layers of silt and clay. Clayey soil has demonstrated good retention for most radionuclides. [4]

7.2.5.1 Design

The design of Tank 19 and Tank 18 closure is such that it provides long-term stability consistent with this performance objective.

Tank 19 and Tank 18 are single-shelled carbon steel waste tanks with no secondary containment. These tanks are 85-feet in diameter and 34-feet high with a domed roof. Tank 19 and Tank 18 were constructed in 1956 and were placed in service in 1961 and 1960, respectfully. Section 2.1 of this determination outlines in detail the construction materials, standards, methods of construction as well as specific features of Tank 19 and Tank 18. [10]

7.2.5.3 Use/Operation

Both Tank 19 and Tank 18 use/operation will support long-term stability consistent with this performance objective. During operations the tank structure is maintained by the SRS Corrosion Control Program. [128] Tank 19 and Tank 18 are monitored for structural integrity. [129] This program includes a Tank Inspection Program, a Tank Leak Detection System, and a Groundwater Monitoring Program. [130],[131],[132],[128] The tank monitoring program continues up to and through the grouting phase and for the full term of institutional control. Tank monitoring continues after grouting with a Groundwater Monitoring Program. [130]

7.2.5.4 Closure

Final closure of Tank 19 and Tank 18 will support long-term stability consistent with this performance objective.

Grouting of Tank 19 and Tank 18 is discussed in Section 2.6 where the details of the grout mixes, methods of placement for the grout and post stabilization are discussed. Long-term stability of the disposal site after closure means that the waste maintains structural integrity under the expected disposal conditions. As such, the long-term stability of the closed Tank 19 and Tank 18 is an important element of meeting the performance objectives. Stability prevents subsidence, water infiltration, and radionuclide release because of disintegration of the waste form, and minimizes the likelihood of intrusion into the waste. The waste form itself can provide structural stability by processing the waste to a stable form or by placing the waste in a disposal structure that provides stability after disposal.

The FFA also covers spills that are located in the proximity of Tank 19 and Tank 18 that will require assessment and possible remediation in the future. [1] Any additional closure activity, such as a low permeability cover, that may occur as a result of closure of these spills has not been factored in the analysis for this determination. [4]

The federal government (Department of Energy) will maintain ownership of the GSA, which includes Tank 19 and Tank 18. The SRS Land Use Plan requires Federal ownership and control of the site well beyond 100 years after closure of the tanks. [92] DOE 5400.5 precludes release of an area unless the radiological hazard meets the requirements of DOE 5400.5 Chapter 4. This requirement ensures virtual perpetual DOE control.

The grouted tanks provide a long-term, stable waste form. The long-term stability of the facility is evaluated in the PODD, which provides a degradation analysis of the grouted tanks and shows the grout will likely remain intact for at least 500 years, but much more likely to remain intact for much longer. [4] As shown above, the site conditions do not present hazards that impact the stability of the closed Tank 19 and Tank 18. In addition, the methods used to close Tank 19 and Tank 18 will result in a closed facility that does not require ongoing active maintenance of the facility following closure. As such, the performance of the closed Tank 19 and Tank 18 complies with the objectives of 10 CFR 61.44.

8.0 STATE-APPROVED CLOSURE PLAN

Section Purpose

The purpose of this section is to explain that the two tanks will be closed in accordance with State-approved closure plans.

Section Contents

This section explains how the tanks are regulated by the State of South Carolina and how State approval for tank closure will be obtained.

Key Points

- Tank waste storage and removal are governed by an industrial wastewater operating permit issued by the SCDHEC.
- The overall plan for closing FTF and HTF, typically referred to as the GCP, will be updated and submitted to the State of South Carolina for approval.
- A specific closure plan for each tank, commonly referred to as a tank closure module, will be developed and submitted to the State of South Carolina for approval and this approval received before activities to close each tank proceed.

NDAA CRITERION

Section 3116 of the NDAA provides in relevant part:

> [T]he term "high level waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy..., in consultation with the Nuclear Regulatory Commission…, determines –
>
> (3)(A) … [W]ill be disposed of -
>
> ***
>
> a. Pursuant to a State-approved closure plan or State-issued permit, authority for the approval of or issuance of which is conferred on the State outside of this section …[.]

STATE-APPROVED CLOSURE PLAN

Tank 19 and Tank 18 waste storage and removal operations are governed by an industrial wastewater operating permit (#17,424-IW) issued by the SCDHEC on March 3, 1993. [16] The permit was issued under the authority of the South Carolina Pollution Control Act, S. C. Code Ann., Section 48-1-10, et seq. (1985) (PCA) and all regulations implementing that Act. The State of South Carolina has authority for approval of wastewater treatment facility closure under Chapter 61, Article 82 ("Proper Closeout of Wastewater Treatment Facilities") of the SCDHEC Regulations. For more information on the regulatory structure under which Tank 19 and Tank 18 are operated and closed, see Appendix B of this document.

The overall plan for closure of the FTF and HTF will be developed and submitted to the State of South Carolina for approval under the Wastewater Operating Permit. This plan is typically

referred to as the GCP. This overall strategy includes performance standards for the closure activities, a determination of applicable performance objectives, and strategy for ensuring the tank system is stabilized. The GCP will also include a description of the tank farm performance assessment process and Groundwater Transport Segment (GTS) methodology that includes results from the fate and transport modeling. The GCP will demonstrate conformance of Tank 19 and Tank 18 as well as the entire FTF and HTF with the appropriate performance objectives.

Prior to commencement of final closure activities, individual tank closure modules describing the details of Tank 19 and Tank 18 closure will be developed and submitted to the State of South Carolina for approval. The tank-specific closure modules will describe the Tank 19 and Tank 18 systems, characteristics of any remaining heel in the tanks, and the Tank 19 and Tank 18 stabilization process. A tank performance assessment demonstrating conformance with the performance objectives from the GCP will be presented in each module.

Each module will also include a demonstration that closure of the relative tank (i.e., Tank 19 or Tank 18) systems combined with previous tank closures, and the estimated impacts for closure of the remaining tanks in FTF will comply with SCDHEC and EPA, Region IV requirements.

CONCLUSION

Final closure of Tank 19 and Tank 18 will be in accordance with State-approved closure plans.

9.0 CONCLUSION

As shown in the preceding sections of this draft 3116 Determination, the solidified residual Tank 19 waste and solidified residual Tank 18 waste are not high-level waste based on the considerations set forth in §3116 (a) of the NDAA and may be disposed of as low-level waste at SRS in accordance with Section 3116 of the NDAA. This draft 3116 Determination will be finalized after DOE has completed consultation with the NRC, and, although not required by Section 3116, after public review and comment.

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APPENDIX A: SYSTEM DESCRIPTION

Appendix Purpose

The purpose of this appendix is to provide additional information on the site system for managing high-level waste

Appendix Contents

This appendix briefly describes the history of the underground waste storage tanks and their contents, and explains how waste is removed from the tanks and the methods used to treat and dispose of this waste.

Key Points

- The site has 51 underground waste storage tanks that entered service between 1954 and 1986.
- There are four types of tanks, designated Types I, II, III, and IV
- The 27 Type III tanks meet current EPA requirements for full secondary containment and leak detection; the other 24, including Tank 19 and Tank 18, do not meet these requirements.
- Approximately 37,000,000 gallons of radioactive waste are stored in the active tanks, most of this from separation of special nuclear materials and enriched uranium in the two site nuclear materials processing facilities known as F Canyon and H Canyon.
- Tanks 17 and Tank 20 have been closed in place with State approval.
- Waste removed from the tanks is being converted into borosilicate glass by the vitrification process that takes place in the DWPF, with the solidified glass contained in stainless steel canisters, which are being stored on site until the Federal geologic repository becomes available for permanent disposal.
- Salt solution from the tanks is treated and disposed of in a grout matrix in the Saltstone Disposal Facility.

HISTORY

Most of the tank waste inventory currently stored at SRS is a complex mixture of chemical and radioactive waste generated during the acid-side separation of special nuclear materials and enriched uranium from irradiated targets and spent fuel using the PUREX process in F-Canyon and the modified PUREX process in H-Canyon (HM process). Waste generated from the recovery of Pu-238 in H-Canyon for the production of heat sources for space missions is also included. The variability in both nuclide and chemical content is due to the fact that waste streams from the 1st cycle (high heat) and 2nd cycle (low heat) extractions from each canyon were stored in separate tanks to better manage waste heat generation. When these streams were neutralized with caustic, the resulting precipitate settled into four characteristic sludges presently found in the tanks where they were originally deposited. The soluble portions of the 1st and 2nd cycle waste were similarly partitioned but have and continue to undergo blending in the course of

waste transfer and staging of salt waste for evaporative concentration to supernate and saltcake. Historically, new waste receipts have been segregated into four general categories in the SRS Tank Farms: PUREX high activity waste (HAW), PUREX low activity waste (LAW), HM HAW, and HM LAW. Because of this segregation, settled sludge solids contained in tanks that received new waste are readily identified as one of these four categories. Fission product concentrations are about three orders of magnitude higher in both PUREX and HM HAW sludges than the corresponding LAW sludges. Because of differences in the PUREX and HM processes, the chemical compositions of principal sludge components (iron, aluminum, uranium, manganese, nickel, and mercury) also vary over a broad range between these sludges. Combining and blending salt solutions has tended to reduce soluble waste into blended PUREX salt and concentrate and HM salt and concentrate, rather than maintaining four distinct salt compositions. Continued blending and evaporation of the salt solution deposits crystallized salts with overlying and interstitial concentrated salt solution in salt tanks located in both Tank Farms. More recently, with transfers of sludge slurries to sludge washing tanks, removal of saltcakes for tank closure, receipts of DWPF recycle, and space limitations restricting full evaporator operations, salt solutions have been transferred between the two Tank Farms. Intermingling of PUREX and HM salt waste will continue until processing in the SWPF can begin.

BACKGROUND

Approximately 37 Mgal [2] of radioactive waste are currently stored in underground waste storage tanks as waste by-products generated from activities conducted at SRS since the site became operational in 1951. Continued long-term storage of these radioactive wastes poses an environmental risk. Therefore, since 1996, DOE and its contractor, WSRC, have been removing waste from tanks, pre-treating it, vitrifying it, and pouring the vitrified waste into canisters for long-term disposal in a Federal repository. As of July 1, 2005, 1,927 canisters of waste have been vitrified for shipment to a Federal repository when the repository is licensed and operational. All canisters filled to date contain vitrified sludge waste. [1]

The Liquid Waste and Waste Solidification System is the integrated series of facilities at SRS that safely manage the existing waste inventory and safely disposition waste stored in the tanks into a final glass or grout form. This system includes facilities for storage, evaporation, waste removal, pre-treatment, vitrification, and disposal. These facilities are briefly described in the text that follows.

TANK STORAGE

Since it became operational in 1951, SRS, a 300-square-mile DOE complex located in the State of South Carolina, has produced nuclear material for national defense, research, medical, and space programs. The separation of fissionable nuclear material from irradiated targets and fuels resulted in the generation of large quantities of radioactive waste which are currently stored onsite in large underground waste storage tanks. The waste was converted to an alkaline solution, metal oxides settled as sludge, and supernate evaporated to form saltcake. [1]

Tanks under construction. Note tank size relative to construction workers. Later, dirt is backfilled around the tanks to provide shielding.

SRS has a total of 51 underground waste storage tanks, all of which were placed into operation between 1954 and 1986. There are four types of waste tanks – Types I through IV. Type III tanks are the newest tanks and were placed into operation between 1969 and 1986. There are a total of 27 Type III tanks. These tanks meet current EPA requirements for full secondary containment and leak detection. The remaining 24 tanks do not have full secondary containment and do not meet EPA requirements for secondary containment. Type I tanks are the oldest tanks and were constructed between 1952 and 1953. Type II waste tanks were constructed between 1955 and 1956. There are eight Type IV tanks that were constructed between 1958 and 1962. Two of these Type IV tanks, in FTF, have been isolated, operationally closed, and grouted. Twelve tanks without secondary containment have a history of leakage. [3] Sufficient waste has been removed from these tanks such that there are currently no active leak sites. The first tank without secondary containment began receiving waste in 1954. This tank is still in service. [1]

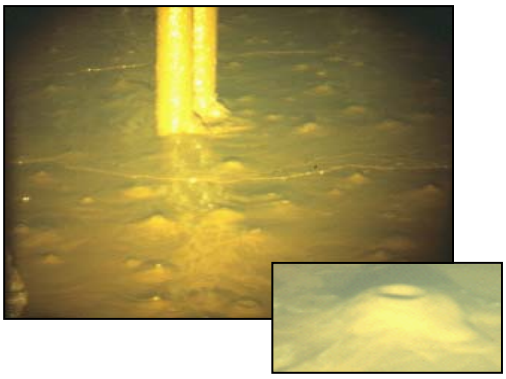
Salt waste is dissolved in the liquid portion of the waste. It can be in normal solution as Supernate (top picture) or, after evaporation, as salt cake (bottom picture) or concentrated supernate. The pipes in all the pictures are

Approximately 37 Mgal of radioactive waste, containing 426 MCi of radioactivity, are currently stored in 49 active waste storage tanks

tanks located in two separate locations, H Tank Farm (29 tanks) and FTF (20 tanks). [2] This waste is a complex mixture of insoluble metal hydroxide solids, commonly referred to as sludge, and soluble salt supernate. The supernate volume is reduced by evaporation, which also concentrates the soluble salts to their solubility limit. The resultant solution crystallizes as salts. The resulting crystalline solids are commonly referred to as saltcake. The saltcake and supernate combined are referred to as salt waste (combined approximately 34 Mgal). [2]

The sludge component of the radioactive waste represents approximately 3 Mgal (7% of total) of waste but contains approximately 203 MCi (48% of total). The salt waste makes up the remaining 33.8 Mgal (93% of total) of waste and contains approximately 223 MCi (52% of total). Of that salt waste, the supernate accounts for approximately 17 Mgal and 211 MCi of the 223 MCi total salt related curies. The saltcake accounts for approximately 16 Mgal and 12 MCi of the remaining salt waste. [2] The sludge contains the majority of the long-lived (half-life > 30 years) radionuclides (i.e., actinides) and strontium. The sludge is currently being stabilized in DWPF through a vitrification process that immobilizes the waste in a borosilicate glass matrix. [2]

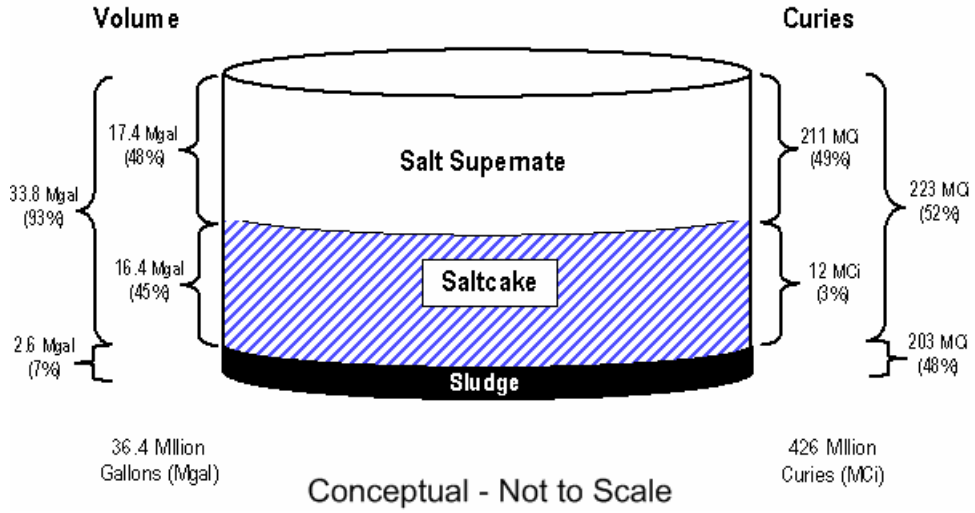
Radioactive waste volumes and radioactivity inventories reported herein are based on the WCS database (see Appendix D), which includes the chemical and radionuclide inventories on a tank-by-tank basis. [4] WCS is a dynamic database frequently updated with new data from ongoing operations such as decanting and concentrating of free supernate via evaporators, preparation of sludge batches for DWPF feed, waste transfers between tanks, waste sample analyses, and influent receipts such as F and H-Canyon waste and DWPF recycle. Volumes and curies referenced in this evaluation are current as of December 1, 2004. [2]



Sludge consists of insoluble solids that settle to the bottom of a tank. Note the offgas bubbles, including hydrogen generated from radiolysis.

Approximately 95% of the salt waste radioactivity is short-lived (half-life 30-years or less) Cs-137 and its daughter product, Ba-137m, along with lower levels of actinide contamination. [2] Depending on the particular waste stream (e.g., canyon waste, DWPF recycle waste), the cesium concentration may vary. The precipitation of salts following evaporation can also change the cesium concentration. The concentration of cesium is significantly lower than non-radioactive salts in the waste, such as sodium nitrate and nitrite; therefore, the cesium does not reach its solubility limit and only a small fraction precipitates. [5] As a result, the cesium concentration in the saltcake is much lower than that in the liquid supernate and interstitial liquid fraction of the salt waste.

Waste Tank Composite Inventory (As of 12/01/04) [2]



WASTE TANK SPACE MANAGEMENT

To make better use of available tank storage capacity, incoming liquid waste is evaporated to reduce its volume. This is critical because most of the SRS Type III waste storage tanks are already at or near full capacity. Since 1951, the Tank Farms have received over 140 Mgal of liquid waste, of which over 100 Mgal have been evaporated, leaving approximately 37 Mgal in the storage tanks. Projected available tank space is carefully tracked to ensure that the Tank Farms do not become "water logged", a term meaning that so much of the usable Type III compliant tank space has been filled that normal operations and waste removal and processing operations cannot continue. A portion of tank space must be reserved as contingency space should a new tank leak be realized. Waste receipts and transfers are normal Tank Farm activities as the Tank Farms receive new waste from the H Separations Canyon, stabilization programs from both the F- and H-Canyon facilities, liquid waste from DWPF processing (typically referred to as "DWPF Recycle"), and wash water from sludge washing. The Tank Farms also make routine transfers to and from waste tanks and evaporators. Currently, there is very little waste that has not had the water evaporated from it to its maximum extent. The working capacity of the Tank Farms has steadily decreased and this trend will continue until salt processing becomes operational or the system becomes water logged. Three evaporator systems are currently operating at SRS - the 2H, 3H, and 2F systems. [1]

WASTE REMOVAL FROM TANKS

During waste removal, water that has been chemically treated to prevent corrosion of the carbon steel waste tanks, inhibited water (IW), is added to the waste tanks and agitated by slurry pumps. If the tank contains salt, this IW, and agitation if required, dilutes the concentrated salt or re-dissolves the saltcake. If the tank contains sludge, this water, and agitation, suspends the insoluble sludge particles. In either case, the resulting liquid slurry, which now contains the dissolved salt or suspended sludge, can be pumped out of the tanks and transferred to waste treatment tanks. [1]

Typical Waste Removal equipment includes three to four 45-foot long slurry pumps and one transfer pump or jet. Note the substantial structural steel required to support the loads in the picture above. At right is the typical installation of a transfer pump (Tank 8) requiring difficult, high-risk entries into High Level Waste Tanks.

Waste removal is a multi-year process. First, each waste tank must be retrofitted with slurry and transfer pumps, infrastructure to support the pumps, and various service upgrades (power, water, air, and/or steam). These retrofits can take between two and four years to complete. Then, the pumps are operated to slurry the waste. Initially, the pumps operate near the top of the liquid and are lowered sequentially to the proper depths as waste is slurried and transferred out of the tanks. Waste removal activities remove the bulk of the waste to prepare the tank for closure. [1]

SAFE DISPOSAL OF THE WASTE

The goal is to convert all of the waste into one of two final waste forms: glass, which will contain 99% of the radioactivity; and saltstone grout, which will contain most of the volume. Each of the waste types at SRS needs to be treated to accomplish disposal in these two waste forms. The sludge must be washed to remove non-radioactive salts that would interfere with glass production. The washed sludge can then be sent to DWPF for vitrification. The salt must be treated to separate the bulk of the radionuclides from the non-radioactive salts in the waste. However, until the startup of SWPF, Interim Salt Processing will be used to accomplish this activity. After startup of SWPF, this separation will be accomplished in SWPF.

SALT PROCESSING

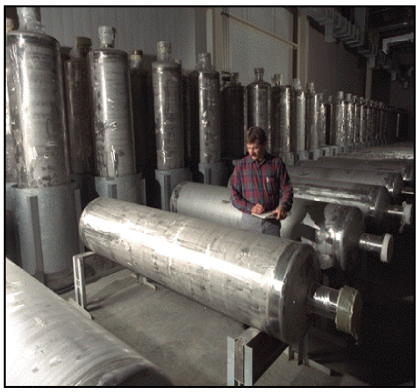
Salt waste accounts for over 90% of the total waste in the tank farms by volume. In order to facilitate waste removal activities including continued disposition of the sludge waste, salt waste must be treated and disposed of. A final DOE technology selection for salt processing was completed and a Salt Processing EIS ROD was issued in October 2001. The ROD designated Caustic Side Solvent Extraction (CSSX) as the preferred alternative to be used to separate cesium from the salt waste. The removal of the cesium via a CSSX process as well as the removal of strontium and actinides will occur in the Salt Waste Processing Facility (SWPF). The SWPF is a high-capacity salt treatment facility that is currently in design and scheduled to initiate hot operations in 2009.

Prior to startup of the SWPF, interim salt processing will occur to the extent necessary to permit the continued removal, treatment and disposition of sludge from the old-style, noncompliant tanks (e.g., Tank 19 and Tank 18) and to create enough tank space to maximize the feed to SWPF at the start of its operation. For salt in selected tanks that are relatively low in radioactive content (e.g., Tank 41 and Tank 25), the treatment process of Deliquification, Dissolution, and Adjustment (DDA) will be utilized. For other salt waste that is still relatively low in radioactive content but that is nominally higher in radioactive content than the DDA salt waste, the waste will be processed through Actinide Removal Processing (ARP) facilities to remove strontium and actinides and through an Modular CSSX Unit (MCU) to remove cesium. The latter two processes are expected to be available in 2007. After startup of SWPF, all remaining salt waste will be processed through this facility.

SLUDGE PROCESSING

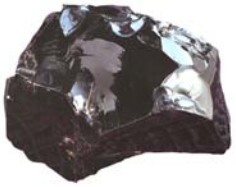
Sludge is “washed” to reduce the amount of non-radioactive soluble salts remaining in the sludge slurry. The processed sludge is called “washed sludge.” During sludge processing, large volumes of wash water are generated and must be volume-reduced by evaporation. Over the life of the waste removal program, the sludge currently stored in tanks at SRS will be blended into separate sludge “batches” to be processed and fed to DWPF for vitrification.

DWPF VITRIFICATION



Canisters prior to being filled with radioactive glass (canisters are ~10 feet in length)

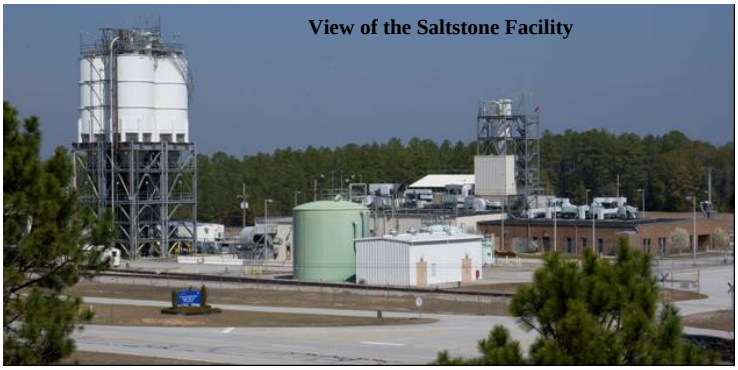
Final processing for the washed sludge and salt waste occurs at DWPF. This waste includes MST/sludge from ARP or SWPF, the cesium strip effluent from MCU or SWPF, and the washed sludge slurry from Extended Sludge Processing. In a complex sequence of carefully controlled chemical reactions, this waste is blended with glass frit and melted to vitrify it into a borosilicate glass form. The resulting molten glass is poured into stainless steel canisters. As the filled canisters cool, the molten glass solidifies, immobilizing the radioactive waste within the glass structure. After the canisters have cooled, they are permanently sealed, and the external surfaces are decontaminated to meet United States Department of Transportation requirements. The canisters are then ready to be stored on an interim basis on-site in the Glass Waste Storage Building (GWSB), pending shipment to a Federal Repository for permanent disposal. A low-level recycle waste stream from DWPF is returned to the Tank Farms. DWPF has been fully operational since 1996. [1]



Sample of Vitrified Radioactive Glass

SALTSTONE: ON-SITE DISPOSAL OF LOW-LEVEL WASTE

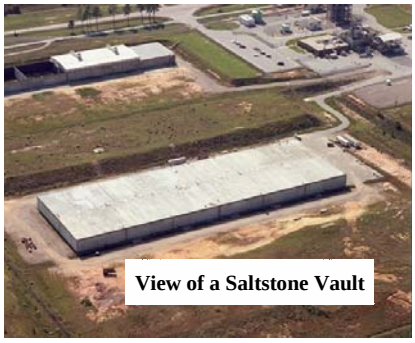
The Saltstone Facility, located in Z-Area, consists of two facility segments: SPF and SDF. SPF is permitted as a wastewater treatment facility per SCDHEC Regulations R.61-67. SPF receives and treats the salt solution to produce grout by mixing the LLW liquid stream with cementitious materials (cement, flyash, and slag). A slurry of the components is pumped into the disposal vaults, located in SDF, where the saltstone grout solidifies into a monolithic, non-hazardous, solid LLW form. SDF is permitted as an ISWLF site, as defined by SCDHEC Regulations R61-66 and R.61-107.16. [6]



View of the Saltstone Facility

The facility will contain many large concrete vaults divided into cells. Each of the cells will be filled with solid saltstone grout. The grout itself provides primary containment of the waste, and the walls, floor, and roof of the vaults provide secondary containment.

Approximately 15 feet of overburden were removed to prepare and level the site for vault construction. All vaults will be built at or slightly below the grade level that exists after the overburden and leveling operations are complete. The bottom of the saltstone grout monoliths will be at least 5 feet above the historic high water table beneath the Z-Area site, thus, avoiding disposal of waste in a zone of water table fluctuation. Run-on and runoff controls are installed to minimize site erosion during the operational period. [6]


View of a Saltstone Vault

The current vault (Vault #4) has the dimensions of approximately 200 feet wide, by 600 feet in length, by 26 feet in height. The vault is divided into 12 cells, with each cell measuring approximately 100 feet by 100 feet. The vault is covered with a sloped, permanent roof that has a minimum thickness of 4 inches, and a minimum slope of 0.24 inches/foot. The vault walls are approximately 1.5 feet thick, with the base mat having a thickness of 2 feet. Operationally, the cells of the vault will be filled to a height of approximately 25 feet with saltstone, and a layer of uncontaminated grout, with an average thickness of 2 feet, will be poured to fill in the space between the saltstone grout and the sloped roof. The other current vault (Vault #1) has the dimensions of approximately 100 feet wide, by 600 feet in length, by 25 feet in height. The vault is divided into 6 cells, with each cell measuring approximately 100 feet by 100 feet. [6]

Closure operations will begin near the end of the active disposal period in the SDF, i.e., after most or all of the vaults have been constructed and filled. Backfill of native soil will be placed around the vaults. The present closure concept includes two moisture barriers consisting of clay/gravel drainage systems along with backfill layers and a shallow-rooted bamboo vegetative cover. [6]

Construction of the Saltstone Facility and the first two vaults was completed between February 1986 and July 1998. The Saltstone Facility started radioactive operations June 12, 1990. Future vaults will be constructed on an as-needed basis in coordination with salt processing production rates.[6]

REFERENCES

1 HLW-2002-00025, *Savannah River Site High Level Waste System Plan*, Westinghouse Savannah River Company, Aiken, South Carolina, Revision 13, March 2002.

2 Ledbetter, L. S., CBU-PIT-2004-00024, *12/01/04 – December Monthly WCS Curie and Volume Inventory Report*, Westinghouse Savannah River Company, Aiken, South Carolina, Revision 0, December 2004.

3 Waltz, R. S., CBU-LTS-2004-00104, *"SRS High Level Waste Tank Leaksite Information (8/04),"* August 2004.

4 Hester, J. R., WSRC-TR-96-0264, *High Level Waste Characterization System (WCS)*, Westinghouse Savannah River Company, Aiken, South Carolina, Revision 0, December 1996.

5 Brooke, J. N., et al., WSRC-TR-99-00358, *Hydrological Methods Can Separate Cesium from Nuclear Waste Saltcake*, Westinghouse Savannah River Company, Aiken, South Carolina, Revision 0, October 1999.

6 WSRC-SA-2003-00001, *Saltstone Facility Documented Safety Analysis*, Westinghouse Savannah River Company, Aiken, South Carolina, November 2004.

APPENDIX B: ENVIRONMENTAL REGULATORY INTEGRATION FOR THE HIGH LEVEL WASTE TANK SYSTEM

Section 3116 of the Ronald W. Reagan NDAA for FY 2005 provides for determinations that relate to specifically identified waste. For example, in the context of a waste tank system, separate NDAA waste determinations may be made for specific waste streams, residual waste in tanks, and identified tank system equipment. This appropriate focus on specific waste determinations does not necessarily provide information concerning the fully integrated regulatory approach for an entire tank system or site. The purpose of this appendix is to briefly describe the fully integrated regulatory context, other than the regulatory authority of the Secretary of Energy, in which such individual waste determinations are made pertaining to the SRS waste Tank System.

In accordance with the Comprehensive Environmental Response, CERCLA 42 U.S.C. 9620(e)(1), as amended by the Superfund Amendments and Reauthorization Act of 1986 (SARA), Pub. L. 99-499 (hereinafter jointly referred to as CERCLA), [1] the DOE, EPA, and the SCDHEC entered into a FFA, effective August 16, 1993. [2] This agreement provides for a comprehensive remediation of SRS, governs the corrective/remedial action process from site investigation through site remediation, and describes procedures for that process. In pertinent part, the FFA establishes the regulatory framework for the operation, new construction, and eventual closure of the liquid waste tank systems. The FFA results in enforceable timetables for the closure of tanks as well as provisions for new construction and prevention and mitigation of releases or potential releases from the tank systems.

Pursuant to the FFA, Section IX, SRS submitted applications to SCDHEC for a Clean Water Act/South Carolina Pollution Control Act wastewater construction permit and operation permits for the tank systems on July 8, 1991 and received construction and operating approval from SCDHEC on March 3, 1993 (Permit # 17,424-IW). [3]

The FFA, Section IX. E., addresses the eventual removal of tanks and ancillary equipment from service and the final closure of the tanks. For tanks and systems that are governed by a wastewater permit, the closure must be performed in accordance with the South Carolina Pollution Control Act, S. C. Code Ann., Section 48-1-10, et seq. (1985) (PCA) and all regulations implementing that Act. As described in Section 8, DOE is meeting that requirement by submittal of the General Closure Plan and the individual closure modules for each tank that is closed. [4]

The FFA, Section IX E.2., addresses the interface between the closure of the tanks and ancillary equipment and the closure and remediation of those soils, structures and equipment that DOE cannot practically remove or decontaminate. For the structures, equipment, and soils that require additional removal or decontamination, CERCLA response actions will be necessary to ensure protection of human health and the environment.

DOE has determined that there are previous spill sites in the tank systems that are not covered by the wastewater permits which will require response actions under the FFA. These spill sites were previously placed on the FFA Appendix G (Site Evaluation List) by DOE at the time of the FFA approval and have subsequently been placed on Appendix C (RCRA/CERLA Units List) (H-Area Tank Farm on May 20, 2002, and FTF on March 26, 2003) [5], [6] for evaluation and possible remediation. DOE, with the approval of EPA and SCDHEC, has determined that a formal Site Evaluation is unnecessary. Instead, two "Operable Units" (OUs) were created to include the areas known as F-Tank Farm and H-Tank Farm. [2]Figure B-1 shows the approximate locations of the operable unit boundaries.

Spills and releases are known to have occurred in operating areas, and in particular the operating areas in the central part of the site known as the GSA. These releases are known to have produced groundwater contamination. DOE addresses the potential groundwater contamination from the GSA in five distinct separate divisions or OUs. [2] They are:

- GSA Eastern Groundwater OU (H-Area including the H-Tank Farm)
- GSA Western Groundwater OU (F-Area including the F-Tank Farm)
- H-Area Seepage Basins (RCRA Action)
- F-Area Seepage Basins (RCRA Action)
- Mixed Waste Management Facility (MWMF) Groundwater (RCRA Action)

For the three RCRA units the corrective actions are underway or in the evaluation stage since the units are out of service and have undergone RCRA closure. The GSA Eastern and GSA Western Groundwater OUs address groundwater emanating from underneath the operating areas that include the F and H Tank Farms. [2] DOE is monitoring the groundwater from those two operable units under formal FFA approved plans until closure of the two areas. Figures B-2 and B-3 show the locations where groundwater monitoring samples are collected. DOE, EPA, and SCDHEC have concluded that the most appropriate action is to continue to monitor the groundwater to ensure that surface water resources are adequately protected. DOE, EPA, and SCDHEC have agreed on this plan to ensure that possible sources of contamination are brought to appropriate closure prior to approving a Record of Decision on these groundwater units. Dates for the investigation and decision on these groundwater units are being reserved pending closure of the industrial areas.

Figure B-1: General Separations Area Groundwater Units

Figure B-2: Sample Locations at the GSA Eastern Groundwater Operable Unit

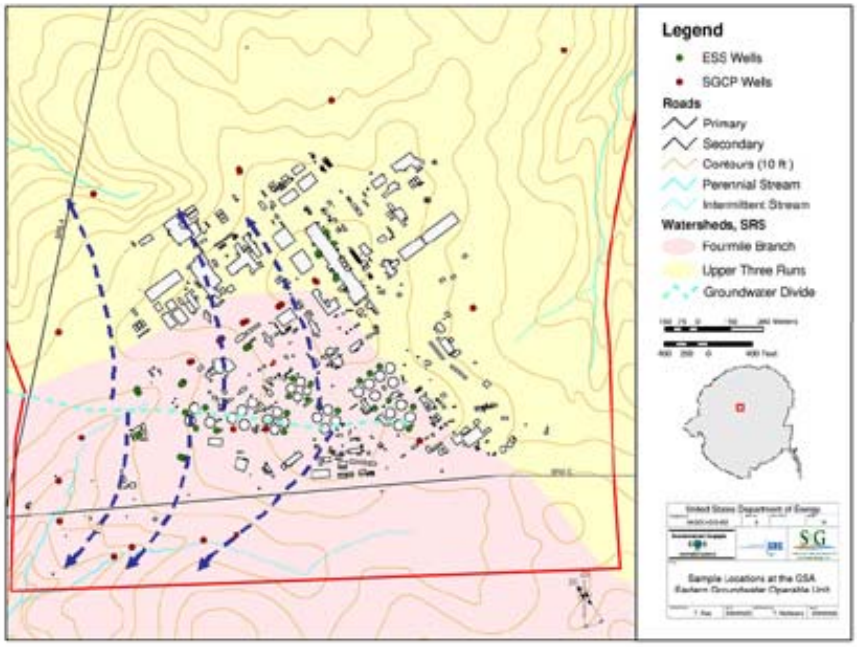
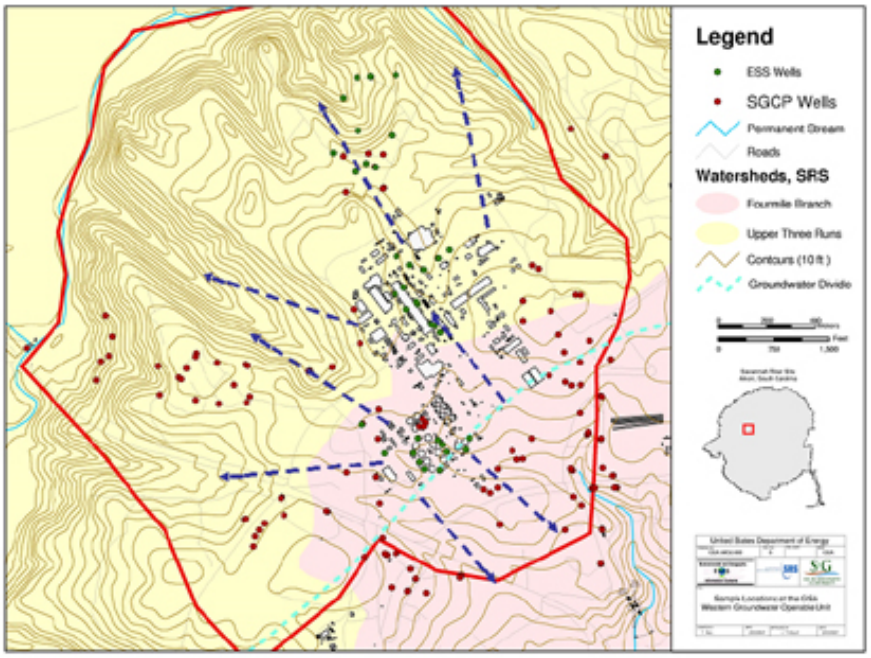


Figure B-3: Sample Locations at the GSA Western Groundwater Operable Unit

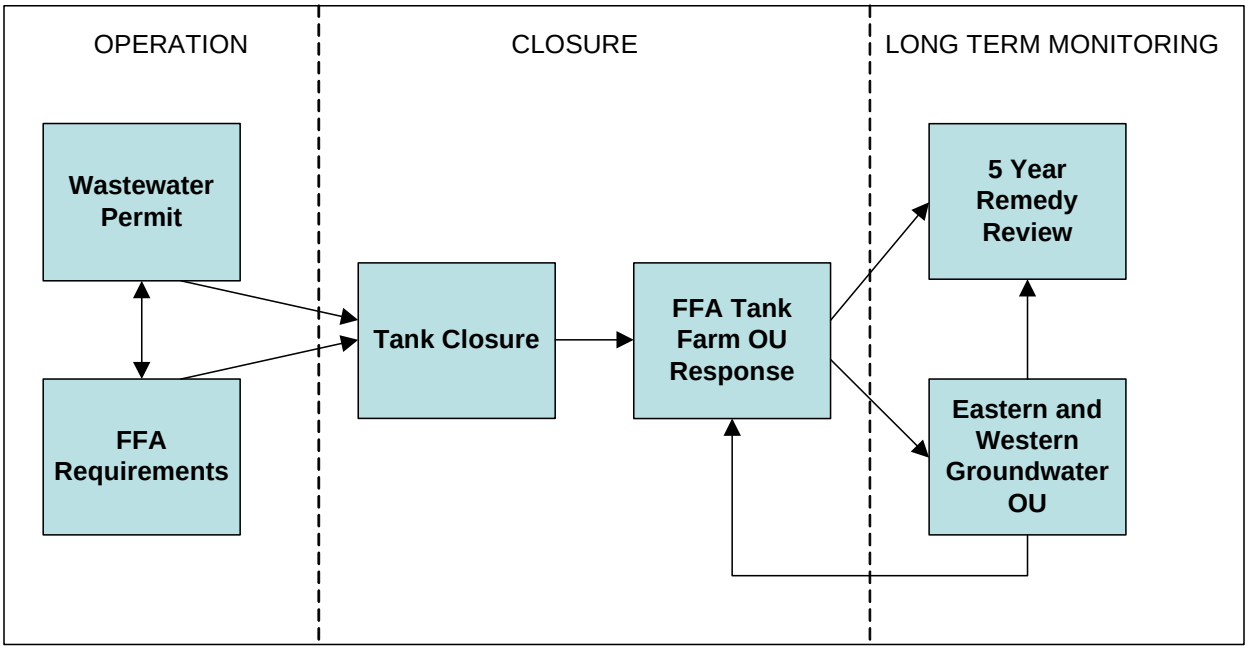


The FFA is currently based on a strategy that relies on an industrial area-by-area closure to fulfill its requirements. Both FTF and HTF are identified as Operable Units with planning dates established by DOE and approved by both EPA and SCDHEC. Once closure of the tanks and ancillary equipment is complete, enforceable dates for the final closure under the FFA can be established by DOE. All response actions under CERCLA are subject to review and comment by EPA, SCDHEC, stakeholders and the public and approval by both EPA and SCDHEC. A formal dispute resolution process is set forth in the FFA in case of disagreement on any actions to be taken.

The FFA requires a review of the selected remedy every five years, consistent with 42 U.S.C 9621(c), to ensure that human health and the environment are being protected by the selected remedy. Should DOE, EPA, and SCDHEC conclude that additional action or modification of the remedy is appropriate, DOE will implement any additional response.

Figure B-4 shows a graphical representation of the relationship between the various regulations and status of the tank systems.

Figure B-4: Environmental Requirements for the Tank Systems



The FFA is designed and is being implemented as written in a manner that integrates multiple regulatory programs and facility activities into a cohesive and protective strategy. There are no identified regulatory gaps in the operation, closure and long-term monitoring of the Tank Farms.

REFERENCES

1 PIT-MISC-0127, *Title 42 - The Public Health and Welfare, Chapter 103 - Comprehensive Environmental Response, Compensation, and Liability, Subchapter I - Hazardous Substances Releases, Liability, Compensation*, U. S. Department of Energy, Washington, D.C. January 2003

2 WSRC-OS-94-42, *Federal Facility Agreement for the Savannah River Site*, Savannah River Operations Office, Aiken, South Carolina, August 1993

3 Chapman, J. W., PIT-MISC-0109, *Partial Permit to Operate F and H Area High Level Radioactive Waste Tank Farms Construction Permit No.: 17,424-IW*, South Carolina Department of Health and Environmental Control, Columbia South Carolina, March 1993

4 PIT-MISC-0108, *South Carolina Pollution Control Act*, South Carolina Code Ann., Section 48-1-10, et seq., 1985

5 Hennessey, B. T., OD-02-208, Recommendation for Ten Site Evaluation Areas Located in H-Area Tank Farm, Savannah River Operations Office, Aiken, South Carolina, May 2002

6 Johnson, T., OD-03-179, Recommendation for Ten Site Evaluation Areas Located in F-Area Tank Farm, Savannah River Operations Office, Aiken, South Carolina, March 2003

APPENDIX C: TANK 19 AND TANK 18 WASTE REMOVAL PROGRAM TECHNOLOGY REVIEW

Appendix Purpose

The purpose of this appendix is to describe the technologies used and considered by the site to remove waste from underground waste tanks.

Appendix Contents

This appendix describes the site waste removal program. Table C-1 summarizes site waste removal technologies history from 1966 through 2003. Table C-2 details various waste removal technologies considered in recent years at SRS. Table C-3 details various waste removal technologies considered, but not used at SRS.

Key Points

- The site has considerable experience with waste removal dating back to 1966.
- The site has considered a large number of different technologies in recent years in its efforts to identify the best available technologies to remove waste from the tanks.

INTRODUCTION

The SRS Waste Removal (WR) program includes the design, construction, testing, and operation of waste suspension and transfer equipment used to remove sludge and salt from the SRS waste tanks. Table C-1 describes the history of waste removal technologies at SRS. Table C-2 describes the waste removal technology used at SRS. Table C-3 describes waste removal technology that has been evaluated for potential use.

BACKGROUND

WR consists of the following functions:

- Preparation of bulk waste
- Transfer of bulk waste
- Preparation of heel
- Transfer of heel
- Spray washing
- Annulus cleaning (as appropriate)
- Tank isolation
- Tank closure

Additional WR functions include sampling, improved ventilation systems, alternative level indication, internal camera inspection equipment, controlled release of trapped hydrogen, and mapping of residual waste deposits.

These efforts are made technically challenging by a number of significant physical and process-related obstacles to waste removal. Physical obstacles (see Figure C-1) include:

- Support columns (produce shadowing effects)
- Horizontal cooling coils
- Vertical cooling coils
- Tank integrity
- Tank bottoms located 45 to 50 feet (for Type IV tanks) below ground surface
- Carbon steel tanks
- Limited tank-top loading capacity
- Contamination containment for potentially leaking equipment
- Non-symmetrical riser positions
- Confined spaces
- Contaminated large equipment disposition
- Potential leakage from primary tank
- Transfer line over pressurization
- Limited openings into the primary tank and annulus space (no larger than 24 inches in diameter)
- Ventilation duct at the bottom of the annulus space
- High volumes of material in very low tank levels 2,710 gallons/in. to 3,540 gallons/in.
- High radiation rates in tank and at riser openings
- Surrounding tanks are processing waste
- Dark, sometimes hot, humid environment in tanks
- Transfers out of the tanks are from one riser location

Process-related obstacles include nuclear criticality safety and in-depth knowledge of waste characteristics.

Figure C-1: Obstacles to Waste Removal

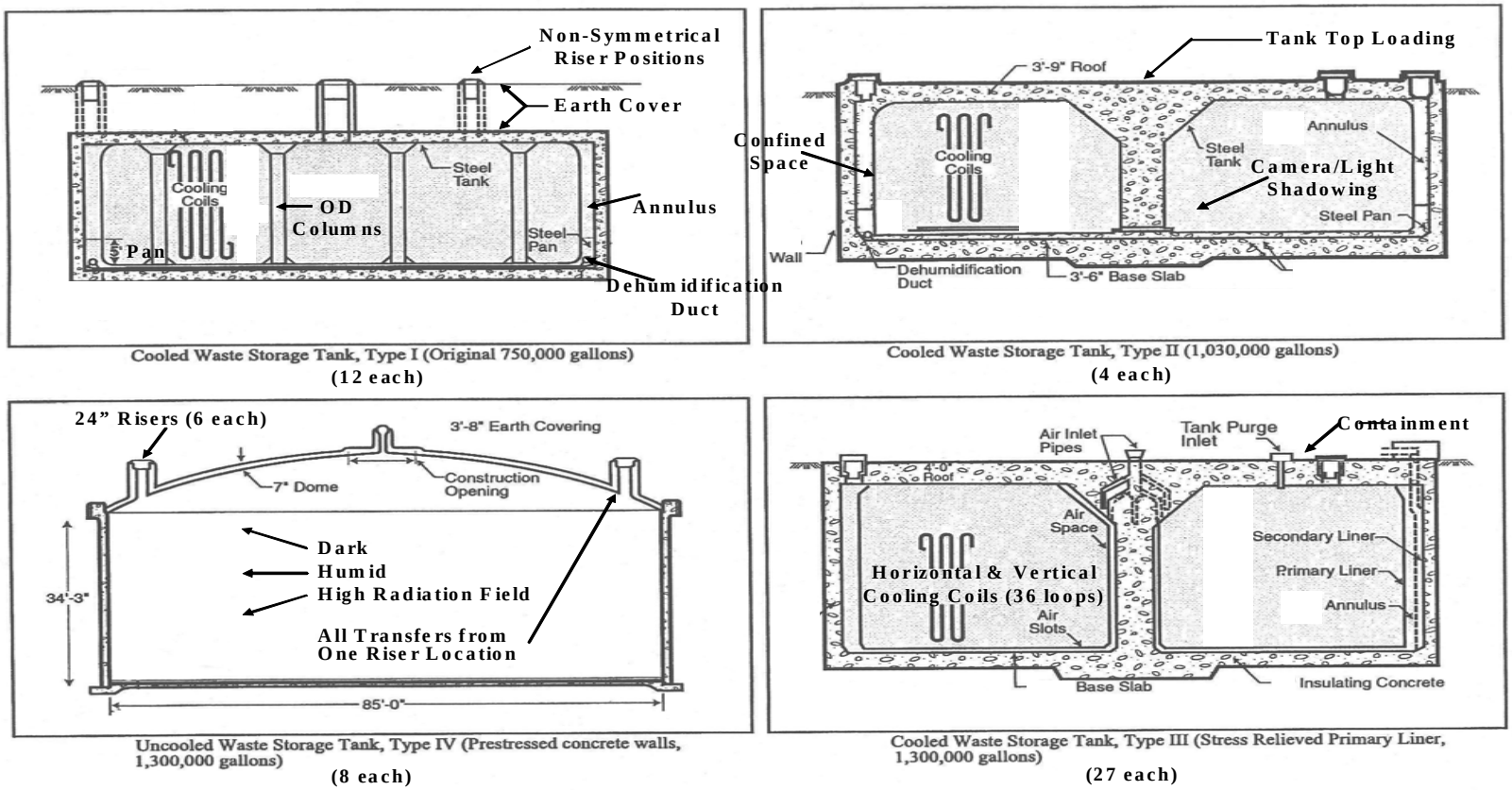


Table C-1. HLW Waste Removal Technologies History (1966-2003)

Tank No.	Waste Type	WR Date	TECHNOLOGY USED SLUDGE SLURRY	Technology Used Salt Dissolution	Transfer Method	Comments
1F	Sludge	1969	Water Sluicers (4)	NA	*	Note 1
2F	Sludge	1966	Water Sluicers (4)	NA	*	Note 1
3F	Sludge	1968	Water Sluicers (3)	NA	*	Note 1
8F	Sludge	2001	LPIs (4)	NA	TTP	Note 2
17F	Sludge/Salt	1985	BWs (3)	NA	TTP	Note 3
17F	Sludge Heel	1997	Flygt Mixers (3)/Water Sluicer (1)	NA	TTP	Note 3
18F	Sludge	1987	BWs (3)	NA	TTP	Note 4
18F	Sludge	2003	ADMP (1)	NA	TP	Note 16
19F	Salt	1982	NA	BWs (2)	TTJ	Note 5
19F	Sludge/Zeolite Heel	2001	Flygt Mixers (3)	NA	GP/PB	Note 15
20F	Salt	1983	Pumpdown of ballast water only	BWs (3)	TJ	Note 6
20F	Sludge Heel	1983	BW (3)	NA	TJ	Note 6
33F	Salt	1983	NA	Density Gradient		Note 14
9H	Sludge	1966	Water Sluicers (4)	NA	*	Note 1
10H	Sludge	1967	Water Sluicers (3)	NA	*	Note 1
10H	Salt	1980	NA	Density Gradient	TJ	Note 7
11H	Sludge	1969	Water Sluicers (4)	NA	*	Note 1
14H	Sludge	1968	Water Sluicers (2)	NA	*	Note 1
15H	Sludge	1982	BWs (2)	NA	TP	Note 8
16H	Sludge/Heel	1979	BWs (3)	NA	TP	Note 9
16H	Sludge/Heel	1980	Chemical Cleaning	NA	TP	Note 9
21H	Sludge	1986	BWs (3)	NA	TP	Note 10
22H	Salt	1986	NA	BW (3)	TP	Note 10
22H	Sludge	1986	BWs (3)	NA	TP	Note 10
24H	Salt/Heel	1981	NA	BW (2)	TTJ	Note 11
24H	Zeolite Heel	1985	Chemical Cleaning		TTJ	Note 11
40H	Sludge Processing	1987	Quad Volute Pumps	N/A		Note 12

Tank No.	Waste Type	WR Date	TECHNOLOGY USED SLUDGE SLURRY	Technology Used Salt Dissolution	Transfer Method	Comments
42H	Sludge Processing	1983	Quad Volute Pumps	N/A		Note 13
51H	Sludge Processing	1983	Quad Volute Pumps	N/A		Note 12
TTP-Telescoping Transfer Pump, TTJ-Telescoping Transfer Jet, LPI- Lawrence slurry pump,						
BW - Bingham-Willamette slurry pump, TJ – Transfer jet, TP – Transfer pump, GP - Goulds pump, PB - Pit Bull pump						

*	Tanks 1F, 2F, 3F, 9H, 10H, 11H, and 14H had the sludges removed by sluicers, which also were the transfer pumps for moving the sludges to Tank 7F and 13H. Except for Tank 11H, these tanks were converted to salt service.
Note 1	Tanks were filled with salt after sludge removal. Density gradient methods for salt removal were tested in Tank 10H. Salt from all these tanks will be removed with three LPIs in each tank.
Note 2	Sludge has been slurried with four LPIs and has been transferred to ESP. A second transfer of 15 kgal of sludge heel is also planned to reduce the existing heel.
Note 3	Salt and sludge were removed from Tank 17F and transferred to Tank 18F, where sludge settled and salt solutions were processed in the evaporator system. Heel was removed by a combination of Flygt Mixers and a water monitor. Following heel removal, the tank was filled with grout and closed.
Note 4	Tank 18F received and stored the sludge from Tanks 17F, 19F, and 20F. The salt from those tanks passed through Tank 18F to the evaporator systems. Tank 18F sludge was slurried and transferred to ESP.
Note 5	Tank 19F had the soluble salts removed using two BWs. The remaining heel was removed using three ITT Flygt Mixers. The solution was transferred using a 200-gpm submersible BIBO pump. Solutions were transferred to Tank 18F, where solids were allowed to settle and the supernate was recycled to tank 19F using a 200-gpm pump.
Note 6	Salt in Tank 20F was initially removed by density gradient method, when inhibitor control required excessive fresh inhibitors, the process was ended and the remainder of the salt was removed using three BWs. Tank 20F has been filled with grout and closed.
Note 7	Tank 10H was used to demonstrate Density Gradient techniques. See Note 1 for remaining salt.
Note 8	Sludge was removed from Tank 15H to provide for demonstration of sludge processing in Tank 42H.
Note 9	Tank 16H was a demonstration tank for removal of sludge using long-shaft slurry pumps (BWs), essentially all sludge was removed by a combination of bulk removal, water washing and oxalic acid cleaning. Tank 16H annulus remains to be cleaned before the tank can be closed.
Note 10	Salt and bulk sludge were removed from Tanks 21 and 22H. Heel will be removed after the tanks are removed from service as DWPF recycle tanks.
Note 11	Salt from Tank 24H was removed as part of demonstration for salt removal using slurry pumps. After salt removal, oxalic acid and water washing was unsuccessful in removing the zeolite heel. Tank 24H is currently used to store DWPF recycle.
Note 12	Tanks 40H and 51H are the two ESP processing/DWPF feed tanks. Four Quad Volute pumps are used to wash and feed to DWPF ~400 kgal of sludge per batch.
Note 13	Tank 42H was used to demonstrate sludge washing and aluminum dissolution process, but has had all its sludge removed to Tank 51H and is now in supernate storage service.

Note 14	Salt dissolution was done on Tank 33F so that it could be used as the fresh high heat waste receipt tank. The tank is currently a low-heat waste receipt tank and an intermediate feed tank for the 2F evaporator.
Note 15	Sludge removal was performed to prepare Tank 19 for FFA Tank Closure. A BIBO transfer pump was used to transfer waste from the tank.
Note 16	Sludge removal was performed to prepare Tank 18 for FFA Tank Closure. A BIBO transfer pump was used to transfer waste from the tank.

Table C-2: SRS Recent Waste Removal Technology

Technology Type	System Attributes	Waste Form	System Operability	Comments
Slurry Pump (SP): ~Forty SPs in HLW. The SPs range from the following specs: 1) 150hp/26 ft effective cleaning radius (ECR), 1750 rpm/1200 gpm/2 nozzles, 2) 300 hp/40 ft ECR/2200 rpm/4000 gpm/4 nozzles, 3) 75hp/900 rpm/900 gpm/2 nozzles [1], [2]	45 ft long-shaft centrifugal pump, journal bearings with a lower product lube bearing, mechanical seals used to reject contamination in 30 psi bearing water column, 1200 gpm, two radial 1-1/2 in. nozzles, 12K lb weight, 2-1/2 in. nitronic 50 shaft with tungsten carbide product lube bearing shaft coating, 14 in. 304L SS column, 1780 rpm, 150 hp, 480 v, 165 amp, variable frequency drive control, 360 degrees rotation utilizing a 1/3-rpm Rotek bearing and electrical motor slip ring, 10 in. spacer cans used to raise and lower pumps.	Non-Newtonian, Bingham plastic. In F-Tank Farm, the waste is composed of Al, Fe, Mn, Ni, U^{238}, Mg, and Zn. HTF is made up of Al and Fe. The average density of the waste in g/ml is 1.38 in FTF and 1.26 in HTF and the wt% insoluble solids is 19.3 in FTF and 11.7 in HTF. The average yield stress in dynes/cm^2 is 200 in FTF and 15 in HTF and the average particle size is 0.5 microns. The average pH of the waste is between 13-14 for both Tank Farms. The SpG = 1.3 to 1.5	SPs can be installed in 24 in. risers. SPs must be submerged at a min. of 16 in. of fluid above the bottom of the pump screen (10 in. above center line of pump discharge nozzles) in order to prevent vortexing and roostertailing. Pump operated at 1600 rpm to protect against resonant frequency excessive vibration. Tank 8 performance: 132.8 kgal of sludge, 230 kgal liquid, 2:1 water to sludge ratio, 15 kgal sludge remained.	Given the access limitations and structural impediments in HLW tanks, the slurry pumps will meet the fundamental installation requirements that other technologies cannot. The long-shafted centrifugal pump will be considered for future waste removal in SRS HLW tanks.
Hydrolaser/Hydrolance – Both were used in Tk 19, Vendor (Augusta Industrial) operated Hydrolaser. [3]	High pressure delivery (10K to 30K psi) system, hose and nozzle (32 gpm/1/4 in. dia). three nozzles, 120° apart and 75° downward. One nozzle straight down.	This technology has been demonstrated in Tank 19 on zeolite and sludge (fast settling solids, 1-6 fps).	The hydrolaser broke up a 42 in. high by 30 in. dia. mound of zeolite using a 13:1 fluidic dislodging ratio. Minimal impact on sludge at 15 feet.	Very effective at contact on extremely hard materials, yet very ineffective at distances over 6 feet with most materials.
Submersible Mixer (Flygt Mixer): SRS has used six mixers since 1997 (one 15 hp, two 4 hp, and three 50 hp). Tank 17 and 19 operating procedures are utilized to operate the mixers. [3]	The 50 hp motor in the Tank 19 design turns a shrouded propeller at speeds up to 860 rpm. This mixer delivers 9,000 gpm (9K gpm is less than vendor published values of 20K gpm due to introduction of a shroud on mixer discharge) while providing a velocity of 1.0 fps at a 50 ft distance and an Effective Cleaning Radius (ECR) of 21 ft.	Same waste form as for SP above. Mixers are challenged by a stationary single-point transfer location due to their limited suspension and ECR.	These mixers require a 36 in. liquid level for operation to prevent excessive vortexing. The 10,000 lb. mixer mast assembly rests on the tank floor, while the Rotek bearing is supported by structural steel that prevents tank top loading.	The effectiveness of the mixers was challenged during the R&D of the product due to multiple failures. The failures have been addressed through CFD modeling, structural fatigue analysis, and a large number of small and large scale testing with sludge/ zeolite simulants.
Sluicers (Waterbrush): This is a water monitor technology that was utilized in Tank 17F (one sluicer). [4], [5]	100 gpm flowrate. Assembly was mounted from the Tank 17 ceiling. High powered lights included for aiming assistance; pan and tilt remote/automatic operation, capable of directing a concentrated 3 in. to 4 in. diameter stream at 80 ft.	Same waste form as for SP above. Waterbrush extremely effective at moving fast settling solids through the use of eroding properties.	The sluicer's 360-degree rotation gave capability to reach all areas of the tank. A drawback of this technology is its high water addition to sludge removal ratio (from 4:1 to 15:1).	Sluicing is a viable technology for both salt and sludge removal Future HLW system planning will involve limited water additions.
Bladed Agitator: Numerous bladed agitators are in use in Pump Tanks in the SRS Tank Farms. [6]	Electric motor driven shaft with three levels of blades along the shaft. The agitator extends to the bottom of the Pump Tank, which is 20 ft long.	Same waste form as for SP above. The pump tank is approximately 8 kgal in volume.	The bladed agitators were effective in the ~8 kgal Pump Tanks, but as designed would not be satisfactory in the much larger volume SRS waste tank (1,300 kgal).	The agitators have had a limited lifespan of approximately 5 years due to motor failures. The agitator would not be effective in a HLW tank due to the small cleaning radius generated by the

Technology Type	System Attributes	Waste Form	System Operability	Comments
				blades (10 to 15 feet). In addition, the agitator would have to be fabricated at a much greater length than has been fabricated to date to be able to mix the contents of a 1,300 kgal tank.
Flusher Nozzle (Water Mouse): Modified sewer cleaning tool was utilized in Tank 17 heel removal. The technology was procured off the shelf except for the steering mechanism, which was designed and built by SRS. [5]	Tank 17 had a distribution of sludge around the perimeter of the tank. The water mouse propelled itself to the tank wall with 80 gpm well water pressurized by a vendor supplied 2 Kpsi pump. The unit moved the heel inward from the perimeter to a distance 5 to 10 feet from the tank wall.	Same waste form as for SP above. The beginning 10 kgal sludge heel in Tank 17 had been reduced to 2.2 kgal of heavily washed, fast settling solids (including 10% concrete fragments from tank ceiling).	The initial use of the water mouse (using vendor-provided steering) proved to be hazardous due to the 45 ft high x 85 ft diameter working space the system needed to be used in. The new steering mechanism designed by SRS eliminated the control issue.	This technology can only be utilized if water additions are allowed in the HLW system. Its effectiveness was proven to be valuable in the heel redistribution process.
SRS Crawler: One SRS designed, vendor supplied water monitor (Akron) on vendor-supplied submersible tracks (INTUK). Not installed. [7]	100 gpm steerable water monitor mounted to a tracked vehicle, operates in < 4 in. deep sludge/liquid. Collapsible platform installable through a 24 in. dia riser, then expands to a 5 ft x 5 ft working system.	Same waste form as for SP above.	The crawler was staged to be used to move residual solids that remain at the completion of Flygt mixer operations in Tank 19.	The crawler is able to get the sluicing tool to the sludge/solids locations. Other crawlers are available commercially (Houdini, ARD).
Air Driven Transfer Pumps (Wilden): Numerous pumps are used throughout SRS and in industry to pump heavy slurries/ materials from tanks/sumps. [4] **Air Driven Transfer Pumps (Pitbull)** [3]	Double diaphragm 120 gpm @ 40 ft of head capacity used in Tank 17. Required a 120 psi dry/lubricated air supply. A flushing system and a set of nibbler dams provided to assist with pushing material to the suction of the pump, and an anti-cavitation plate to support low level pumpdowns.	Same waste form as for SP above.	The pumps transferred a total of 7.6 kgal of sludge from Tank 17. The ratio of water to sludge ranged from 9:1 to 19:1. System performance was monitored by measuring the rad rates on the above-grade transfer line. Note: Three pumps were used in Tank 17 due to system failures caused by inadequate flushing, air line freezing, and material compatibility.	Materials of construction, flushing, dry/ lubricated air are all very significant design attributes that need to be considered for successful installations.
Submersible Centrifugal Pump (BIBO, GPM, Goulds): One pump is currently installed in Tank 19 and has successfully operated for over 400 hours. [3]	Centrifugal pump, 13 hp to 20 hp, submersible, 180 gpm @ 125 ft head, capable of pumping down to 1-1/2 in., used to transfer waste from Tank 19 to Tank 18.	Same waste form as for SP above.	The pumps are stationary in Tanks 19 and 18. They can be elevated or lowered only with great difficulty due to the fact that they are designed for use in fixed positions.	Proven system that would be considered disposable as long as the space that the pump system occupied in the tank riser was not needed for isolation and closure activities.
Telescoping Transfer Pump (TTP): Approx. twenty TTPs in HLW System. [8]	Long shafted centrifugal pump, telescoping, 80 to 100 gpm, 75 hp, 4,800 lb w/o bearing water in column, 3600 rpm, 460 VAC motor, 45 ft long, 2-1/2 in. dia. discharge nozzle, nitronic 50 shaft, 2.5 in. dia. shaft, 304L SS column.	Same waste form as for SP above.	The TTP requires a 24 in. dia. riser to accommodate the 23 in. dia. pump casing. Bearing water for contamination control in pump column. The pump can be telescoped to different elevations.	The Tank 8 TTP worked flawlessly once the impeller clearance caused by a cold set was resolved. The system is effective for emptying an HLW tank down to a tank level of 3 inches.
Advanced Design Mixer	Variable-speed 55-ft long centrifugal pump with air-pressurized column	Same waste form as for SP above.	Used in Tank 18, 1000 hours over an 8-	ADMP performance (flow) deteriorated

Technology Type	System Attributes	Waste Form	System Operability	Comments
Pump (ADMP): Submersible 300 hp, 55 ft long centrifugal pump with two discharge nozzles at 5,200 gpm per nozzle. One ADMP installed in Tank 18 operated for 1000 hours. [11], [12]	assembly, two tangentially opposed 6-in. diameter discharge nozzles (nozzle centerlines located 27 in. above tank bottom). Nozzles oscillate though 180º of rotation. Driven by 300-hp VFD induction motor. 10,400 gpm total flow rate at 52 ft head at 1185 rpm. Produces a nozzle discharge velocity of 58.9 ft/s, yielding a 52 ft ECR in HLW tank. Impeller vanes can pass solids up to 7/8 in. diameter. Pump casing dia. 39 in.		month period in support of six batch transfers to Tank 7. Tank level ranged 60 in. to 72 in. (33 in. to 45 in. impeller submergence). Total of 800 kgal of well water added during campaign. In Tank 18, cleaning was effective to within 1.5 feet of tank wall. The ADMP requires a riser diameter of at least 40 in. to accommodate the 39 in. dia. pump casing.	when cavitation occurred. The conditions that must be controlled to prevent cavitation include tank level, fluid temperature, suction screen clogging, obstructions in fluid approach area to pump suction, and pump speed.
EMMA (Easily Maneuvered Manipulator Arm): Tubular arm consisting of up to five sections linked together by flexible joints. The sections are connected by a series of cables that control movement. The arm control user interface is a bank of joysticks. [13],[14],[15]	Different end effectors can be mounted on the arm for multiple purposes. The arm requires a 24 in. access opening. The payload is 250 lb at a maximum reach of 75 ft. The controls are made up of a large bank of hydraulically driven cables (thin width bands).	Same waste form as for SP above.	The system has never been used in an SRS HLW Tank. It utilizes a very complex set of hydraulic cylinders with associated controls. The operating controls do not allow for instantaneous stopping capability.	This technology can only be utilized if the end effector(s) can reach all areas of the 85 ft diameter, 40 ft deep tank for the purpose of sludge extraction. The arm does not have identified end effectors to date, so effectors would have to be developed. Articulation of the arm would be impeded by structural internal tank supports.

Table C-3: Waste Removal Technology Evaluated, But Not Yet Used At SRS

Technology Type	System Attributes	Waste Form	System Operability	Comments
Delphinus Manipulator Arm: Heavy-duty manipulator arm that utilizes an end effector. The early design incorporated a bull vacuum system. The technology is not off the shelf and was never fully developed. [16]	The arm has a 75 ft reach with a 3K lb payload. The arm requires a significant structural support system to manage its weight. A proposal from Delphinus was to utilize the existing slurry pump structural steel. It appears to be a very rugged piece of equipment. Different end effectors can be mounted on the arm. The arm requires a 36 in. access opening. The controls are hydraulic crane type controls. The end effector utilizes a tank top vacuum system in which the retrieved waste would have to be dispositioned to another location by either driving the material to another tank or transferring the waste above ground.	Same waste form as for SP above.	The system has never been used in an SRS HLW Tank. The infrastructure required to support the crane-like arm would be extensive.	This technology can only be utilized if the end effector(s) can reach all areas of the 85 ft diameter, 40 ft deep tank for the purpose of sludge extraction. The arm does not have identified end effectors to date, so effectors would have to be developed. Articulation of the arm would be impeded by structural internal tank supports.
Light Duty Utility Arm (LDUA): This is an articulating arm that can reach down into a tank and then be extended laterally to position a variety of tools (end effectors) to clean out a tank. Three arms were built for Hanford, and a Modified Light Duty Utility Arm (MLDUA) was also successfully used at ORNL. [17]	The arm has a 200 lb payload capacity. End effectors utilized with the MLDUA at Oak Ridge consisted of water jets and jet pumps to mobilize sludge deposits and then transport the sludge out of the tank.	Same waste form as for SP above.	The system has never been used in an SRS HLW Tank. The infrastructure required to support the crane-like arm would be extensive.	This technology could only be utilized if the end effectors can reach all areas of the 85 ft diameter, 40 ft deep tank for the purpose of sludge exrtaction. The arm does not have identified end effectors that can accomplish complete heel removal to date, so effectors would have to be identified and designed. Articulation of the arm would be impeded by structural internal tank supports.
Borehole Miner: This technology consists of a high-pressure water jet discharged from an extendable nozzle. The equipment was adapted for the ore mining industry. [18]	The nozzle is supplied at pressures up to 3.000 psi and can be remotely extended, rotated and angled. A Borehole Miner was designed for use on Tank 19 but was never used.	Same waste form as for SP above.	The system has never been used in an SRS HLW Tank. The system requires a large volume of new or recycled liquids to assist with eroding the material needing to be removed.	The movement of the extendable arm would be impeded by nternal structural support members in the HLW tanks.
ARD Environmental Services, Inc. (ARD): ARD provides equipment for operations, maintenance, and RadCon coverage. This equipment is a robotic crawler, installed on the tank floor, that utilizes an attached pumping system. [19]	Solids are removed by high pressure spraying, grinding, brushing and vacuuming/pumping out of the tank.	Same waste form as for SP above.	The system has never been used in an SRS HLW Tank. The system has been used at SRS in the Reactor Materials waste tanks to mobilize and transfer Low Level Waste.	The crawler is able to get the sluicing tool to the sludge/solids locations. Other crawlers are available commercially (e.g., Houdini). The crawler mobility would be limited by internal structural support members in the HLW tanks.

Technology Type	System Attributes	Waste Form	System Operability	Comments
Red Zone "Houdini": This is a remotely operated hydraulic work platform including a manipulator, plow blade, scoop, spray nozzles and a hydraulic shear. [20], [21]	The Houdini platform has been used at Fernald and in the gunite tanks at ORNL. It can be installed through a 22.5 in. tank opening. The entire system consists of the platform, tools, a power distribution control unit, a control console, and a tether management system.	Same waste form as for SP above.	The system has never been used in an SRS HLW Tank. The Houdini has had many upgrades during its development to met the requirements for removal of various waste forms. The design has been modified to correct hydraulic fluid leaks identified during operation.	The crawler mobility would be limited by internal structural support members in the HLW tanks.
Submersible Mixer Pump (SMP): This technology employs a slurry pump with a submersible product cooled motor that can be reused in multiple tanks. [22], [23] (Planned for use in Tanks 4-6)	Oscillating, centrifugal pump, with two diametrically opposed 4.4-in. discharge nozzles (nozzle centerlines located 24 in. above bottom of mounting foot). Driven by 3-phase 305-hp induction motor, hermetically sealed and cooled by process liquid. 7600 gpm total flow rate at 101 ft head at 1400 rpm. Maximum pump dia. 22.5 in. Total weight ~10,000 lb.	Same waste form as for SP above.	The system has never been used in an SRS HLW Tank. Pump length can be adjusted by inserting column sections. The SMP requires a 23 in. dia. riser opening to accommodate the 22.5 in. dia. pump casing.	To achieve prime, impeller must be fully submerged (tank level 44 in. above mounting foot) prior to pump startup.
Chemical Cleaning using Sluicers/Mixers: This technology involves using dilute (2-4 wt%) heated (90°C) oxalic acid (OA) to dissolve the sludge such that it can be pumped out as was done during Tank 16 acid cleaning. [24], [25]	OA is added via water monitors to provide agitation to prevent the formation of a boundary layer.	Same waste form as for SP above.	The system has not been used in a SRS HLW tank under the current authorization basis.	
Vacuum System: This technology was reviewed with the Delphinus articulating arm. [16]	The system uses a grinding mechanism on an end effector that sluices away waste material and then transfers it out of the tank into an above-tank vacuum collection vessel with filtered air discharge.	Same waste form as for SP above.	The system has never been used in an SRS HLW Tank.	This technology will be challenged by the requirements to manage the contaminated air pumped from the vacuum vessel. In addition, the collected waste must then be relocated to another location for continued disposition.
Confined Sluicing End Effector: This technology was utilized as a demonstration at Hanford and was successfully installed and operated at ORNL. [26]	The end effector used high pressure fluid to generate a sluicing action on hardened material , which is then discharged through a transfer hose to a receipt tank with the assistance of a vacuum system. The end effector must be kept in close contact with the material being removed, and therefore a high precision articulating arm with well-lighted remote viewing must be available.	Same waste form as for SP above.	The system has never been used in an SRS HLW Tank. The system requires close contact with the material being removed, therefore the CSEE moves at a very slow rate. Completing a waste removal campaign in an SRS HLW tank would require intensive operator support over an extensive time period for moving the end effector to cover the entire area of the tank.	This technology will be challenged by the requirements to manage the contaminated discharged air. In addition, the collected waste must then be relocated to another location for continued disposition. This technology can only be utilized if the end effector(s) can reach all areas of the 85 ft. diameter, 40 ft. deep tank for the purpose of sludge extraction. Movement of the arm would be impeded by structural internal tank supports.

Technology Type	System Attributes	Waste Form	System Operability	Comments
Removal of the Tank Contents through Excavation: This technology would require undefined D&D procedures and equipment resources that have not been evaluated to date. [27], [28]	Physical removal of tank and its residual contents employing D&D project techniques (excavation, segmentation, contamination control, packaging, shipping, etc.), with material processed and dispositioned elsewhere.	Same waste form as for SP above.	This process has never been used for dispositioning SRS HLW tanks.	This technology is challenged by the complexity and extensive scope associated with excavating in an operating tank farm. Significant occupational radiation dose to the workers would also be a factor.
Russian Pulsating Mixer Pump: Air-operated pump with the capability to sluice away tank solids using process fluid. Equipment commercially available and used in Russia, and tested by DOE at PNNL and in Tank TH-4 at ORNL. [29], [30]	The PM&TP tested produced 300 liters/minute velocity @ 2 pulses per minute, 18 to 25 cfm flowrate, 43 ft cleaning radius with four sluicing nozzles, 3 meter ECR with lower submersible nozzles. Entire head assembly rotates through a 90° arc to sweep a circular pattern.	2.5-ft layer of sludge, similar to SRS HLW tank sludge.	The system has not been used in an SRS HLW Tank. In Tank TH-4 at ORNL, system left a residual ring only 4 inches deep near the outside wall of the tank.	The technology has potential. Compared to centrifugal mixing pumps, The PM&TP provides excellent scouring action, low cost, mechanical simplicity, and ability to operate in much lower liquid levels.
Modified Sluicing and Acid Dissolution: This technology was used at Hanford's C-106 Tank to remove 2,800 gallons of heel. [31], [32]	In Hanford Tank C-106, this technology employed a sluice pump discharging through a manipulator-mounted nozzle to break up waste solids which were slurried and pumped out for storage in a double-walled tank. Oxalic acid was used to help dissolve solids. Remote viewing was used to aim the sluice nozzle.	Layers of supernate, sludge, and hardpan (from dissolution of aluminum clad fuel elements). 8% solids	97% of sludge removed by sluicing and total solids reduced to 0.3%.	Requires precision manipulator with CCTV viewing for directing sluice stream onto desired portion of waste. Works in conjunction with centrifugal slurry pump. Hanford technology viable baseline for SRS HLW tank application.
Video Camera Mapping/CAD Modeling for Residual Heel Volume Estimates: This technology was used by both SRS and Hanford to estimate heel volumes since the normal operating reel tapes become inoperable due to the lack of liquid for conductivity and tank level. [33]	Normal (not rad hardened) CCTV cameras can be employed; and CAD drawing or spreadsheet mapping software can be used.	Variable mounds of solids and liquids.	An available, vendor-supplied technology.	A viable technology for use in conjunction with tank cleaning and inspection equipment.
Mechanical Conveyance Systems: This technology is used in the mining industry to move material from one location to another. [34]	Electrically powered rail or pneumatic wheel conveyance for movement of materials.	Solids/liquids.	These systems are well suited for conveying bulk material, but there is no experience available in fitting them with the containment capability that would be needed for use in handling radioactive materials.	No practical application for cleaning SRS HLW tanks.

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APPENDIX D: WASTE CHARACTERIZATION SYSTEM

Appendix Purpose

The purpose of this appendix is to describe the site WCS.

Appendix Contents

This appendix contains a brief description of the WCS.

Key Points

- This system is a controlled database used to track waste data.
- It tracks concentrations of 41 radionuclides and 96 chemicals in each of the site underground waste tanks.
- System data are updated routinely under strict quality control and quality assurance requirements.

WCS is an electronic information system that tracks waste tank data, including projected radionuclide and chemical inventories, based on sample analyses, process histories, composition studies, and theoretical relationships. The system, initially developed in 1995, tracks the concentrations of 96 chemicals and 41 radioactive nuclides in each of the SRS waste tanks. The 41 nuclides were primarily selected based on their impact to waste tank source term, inhalation dose potential, or on the E-Area Vault Waste Acceptance Criteria. WCS consists of two functional areas: a sample data repository and waste characterization. The waste characterization contains inputs and calculations used to represent real-time conditions in the waste tanks. These real-time conditions are used to determine compliance (as input to Safety Basis calculations) with several nuclear safety administrative control programs such as the waste tank Flammability Control Program and the waste tank Corrosion Control Program. [1]

WCS is routinely updated to reflect changes in waste inventory resulting from the receipt of new waste from the SRS Canyon facilities, the Effluent Treatment Facility, or recycle waste from DWPF, the transfer of waste for processing at DWPF or SDF, or from the receipt of new characterization information in the form of sample analysis results, field measurements, waste tank video inspections, or technical reports. The waste tank contents in WCS are also adjusted to reflect the impacts of evaporator operation and to account for waste tank to waste tank transfers. [1]

Prior to new data being entered into WCS, a Data Integrity Review Team evaluates the new data to determine its technical validity. The core team members are engineers or scientists with expertise in relevant fields such as flammability, corrosion, organic chemistry, or criticality. The Data Integrity Review Team routinely meets to perform evaluations of new characterization information. As part of the review the new data is compared to previous measurements of the same parameter in the same waste tank, to previous measurements of the same parameter from other waste tanks, to the expected value for that parameter based on the waste tank's processing history, and to the relative concentrations of other parameters in the same data set to determine if the new data can be accepted as a valid representation of actual conditions. Once the Data

Integrity Review Team has deemed the data to be valid, it is approved for entry into the waste characterization portion of WCS and into the data repository. [1]

WCS is a controlled database that complies with strict Quality Assurance (QA) requirements. The QA requirements apply to not only data entry but also configuration control of the database software itself. All data entries into WCS must be independently verified. A Software Quality Assurance Plan governs the methods that must be utilized to maintain the database baseline configuration and to ensure the proper handling of the database. To address backup and recovery, including disaster recovery, WCS is hosted on a server that provides daily backup of all WCS files. [2]

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APPENDIX E: F-TANK FARM RESIDUAL MATERIAL INVENTORY

E.1 F-TANK FARM WASTE TANK RESIDUAL MATERIAL INVENTORY

Modeling of the residual material inventory in the remainder of the FTF waste tanks was performed for information as a part of the Tank 19 and Tank 18 tank closure process. The remainder of the FTF waste tanks will contain residual waste after waste removal. An estimate of the inventory of radionuclides in the residual material in the tanks after waste removal was prepared. The estimated inventories are extracted from the WCS. It is assumed that the radionuclide contribution from residual salt and supernate waste will be negligible compared to the radionuclide inventory of the residual sludge. The radioactive inventories are calculated to a reference date of May 30, 2005 for decay, which is conservative since the tanks will not be closed until the 2010 – 2020 timeframe. [1]

Tank 19 and Tank 18 were discussed in Section 2.5. Tank 17 and Tank 20 residual waste inventories upon closure were established and documented during the closure process for these tanks using similar techniques. [2], [3], [4] The data from Table E-1 and Table E-2 are used as the tank radiological inventories in fate and transport modeling. [5] Additional information on the FTF residual material inventory is described in the PODD. [6]

Table E-1: Radionuclide Inventory in FTF Tank Residual Material (Curies)

Tank	H-3 (Ci)	C-14 (Ci)	Co-60 (Ci)	Ni-59 (Ci)	Ni-63 (Ci)	Se-79 (Ci)	Sr-90 (Ci)	Y-90 (Ci)	Nb-94 (Ci)	Tc-99 (Ci)	Ru-106 (Ci)
1	0.00E+00	9.44E-05	1.45E+00	1.69E-01	1.39E+01	1.18E-01	4.07E+03	4.07E+03	8.82E-05	2.04E+00	3.32E-08
2	0.00E+00	1.60E-04	7.52E-01	1.89E-01	1.50E+01	1.29E-01	3.97E+03	3.97E+03	9.65E-05	2.23E+00	1.64E-10
3	0.00E+00	6.93E-05	4.29E-01	8.17E-02	6.56E+00	5.59E-02	1.80E+03	1.80E+03	4.18E-05	9.67E-01	5.59E-10
4	0.00E+00	4.51E-06	5.63E+00	1.35E-01	1.18E+01	9.66E-02	4.25E+03	4.25E+03	7.24E-05	1.67E+00	2.67E-04
5	0.00E+00	6.79E-05	2.03E+00	1.51E-01	1.26E+01	1.05E-01	3.95E+03	3.95E+03	7.89E-05	1.82E+00	3.04E-07
6	0.00E+00	0.00E+00	4.53E+00	2.12E-01	1.83E+01	1.52E-01	6.25E+03	6.25E+03	1.14E-04	2.63E+00	2.81E-06
7	0.00E+00	1.06E-04	5.96E-01	2.91E-02	2.51E+00	1.69E-02	6.90E+02	6.90E+02	1.26E-05	2.92E-01	4.17E-06
8	0.00E+00	9.64E-05	1.36E+00	5.04E-02	4.41E+00	3.25E-02	1.39E+03	1.39E+03	2.43E-05	5.62E-01	1.01E-05
17	2.22E+01	3.12E-03	2.29E+00	1.83E-01	0.00E+00	1.56E-02	8.34E+02	8.34E+02	0.00E+00	3.58E+00	2.02E-03
18	6.16E-02	5.61E-02	4.35E+00	9.12E-01	8.04E+01	5.05E-02	1.41E+03	1.41E+03	4.01E-05	1.48E+00	9.72E-05
19	3.78E-01	1.12E-01	7.28E+00	8.26E-01	7.62E+01	5.89E-02	4.00E+01	4.00E+01	3.63E-05	6.76E+00	2.70E-04
20	0.00E+00	6.60E-04	6.70E-01	3.90E-02	0.00E+00	3.20E-03	1.90E+02	1.90E+02	0.00E+00	8.50E-01	4.30E-04
25	0.00E+00	3.03E-03	5.11E+00	1.78E-01	1.74E+01	1.47E-02	9.50E+02	9.50E+02	7.81E-06	2.55E-01	3.45E-01
26	0.00E+00	3.03E-03	5.11E+00	1.78E-01	1.74E+01	1.47E-02	9.50E+02	9.50E+02	7.81E-06	2.55E-01	3.45E-01
27	0.00E+00	3.03E-03	5.11E+00	1.78E-01	1.74E+01	1.47E-02	9.50E+02	9.50E+02	7.81E-06	2.55E-01	3.45E-01
28	0.00E+00	3.03E-03	5.11E+00	1.78E-01	1.74E+01	1.47E-02	9.50E+02	9.50E+02	7.81E-06	2.55E-01	3.45E-01
33	0.00E+00	4.55E-05	1.80E+01	6.00E-02	5.94E+00	4.13E-02	2.77E+03	2.77E+03	3.09E-05	7.13E-01	1.36E+00
34	0.00E+00	0.00E+00	3.70E+01	1.90E-01	1.84E+01	1.36E-01	8.43E+03	8.43E+03	1.02E-04	2.36E+00	2.41E-01
44	0.00E+00	3.03E-03	5.11E+00	1.78E-01	1.74E+01	1.47E-02	9.50E+02	9.50E+02	7.81E-06	2.55E-01	3.45E-01
45	0.00E+00	3.03E-03	5.11E+00	1.78E-01	1.74E+01	1.47E-02	9.50E+02	9.50E+02	7.81E-06	2.55E-01	3.45E-01
46	0.00E+00	3.03E-03	5.11E+00	1.78E-01	1.74E+01	1.47E-02	9.50E+02	9.50E+02	7.81E-06	2.55E-01	3.45E-01
47	0.00E+00	3.03E-03	5.11E+00	1.78E-01	1.74E+01	1.47E-02	9.50E+02	9.50E+02	7.81E-06	2.55E-01	3.45E-01

Tank Group	H-3 (Ci)	C-14 (Ci)	Co-60 (Ci)	Ni-59 (Ci)	Ni-63 (Ci)	Se-79 (Ci)	Sr-90 (Ci)	Y-90 (Ci)	Nb-94 (Ci)	Tc-99 (Ci)	Ru-106 (Ci)
Tanks 1- 8	0.00E+00	5.99E-04	1.68E+01	1.02E+00	8.51E+01	7.06E-01	2.64E+04	2.64E+04	5.29E-04	1.22E+01	2.84E-04
Tanks 17-20	2.26E+01	1.72E-01	1.46E+01	1.96E+00	1.57E+02	1.28E-01	2.47E+03	2.47E+03	7.64E-05	1.27E+01	2.82E-03
Tanks 25-28, 33-34, 44-47	0.00E+00	2.43E-02	9.59E+01	1.67E+00	1.64E+02	2.95E-01	1.88E+04	1.88E+04	1.95E-04	5.11E+00	4.36E+00

Table E-1: Radionuclide Inventory in FTF Tank Residual Material (Curies) (Continued)

Tank	Rh-106 (Ci)	Sb-125 (Ci)	Sn-126 (Ci)	I-129 (Ci)	Cs-134 (Ci)	Cs-135 (Ci)	Cs-137 (Ci)	Ba-137m (Ci)	Ce-144 (Ci)
1	3.32E-08	8.74E-02	2.19E-01	9.71E-06	8.12E-05	1.36E-03	2.88E+02	2.73E+02	3.44E-11
2	1.64E-10	1.83E-02	2.40E-01	1.06E-05	8.68E-06	1.49E-03	2.83E+02	2.68E+02	2.26E-14
3	5.59E-10	1.42E-02	1.04E-01	4.61E-06	8.54E-06	6.47E-04	1.28E+02	1.21E+02	1.92E-13
4	2.67E-04	1.87E+00	1.80E-01	7.96E-06	5.54E-03	1.12E-03	2.97E+02	2.81E+02	4.72E-06
5	3.04E-07	1.87E-01	1.96E-01	8.68E-06	2.32E-04	1.22E-03	2.79E+02	2.64E+02	5.88E-10
6	2.81E-06	5.68E-01	2.83E-01	1.25E-05	8.61E-04	1.76E-03	4.39E+02	4.15E+02	1.00E-08
7	4.17E-06	1.12E-01	3.14E-02	1.39E-06	2.32E-04	1.96E-04	6.71E+02	6.35E+02	6.75E-08
8	1.01E-05	2.67E-01	6.05E-02	2.68E-06	5.57E-04	3.76E-04	9.75E+01	9.22E+01	1.65E-07
17	2.02E-03	1.47E+00	2.84E-02	1.28E-06	6.42E-03	1.74E-04	5.78E+01	5.46E+01	1.19E-04
18	9.72E-05	1.17E+00	1.40E-01	6.42E-06	3.29E-03	8.74E-04	1.23E+04	1.16E+04	1.44E-06
19	2.70E-04	2.59E+00	1.27E-01	4.95E-05	7.60E-03	7.91E-04	4.90E+04	4.64E+04	3.71E-06
20	4.30E-04	4.50E-01	5.90E-03	2.60E-07	2.00E-03	3.60E-05	4.10E-01	3.88E+01	1.80E-05
25	3.45E-01	6.75E+00	2.74E-02	1.21E-06	5.36E-02	1.71E-04	6.83E+03	6.46E+03	1.60E-01
26	3.45E-01	6.75E+00	2.74E-02	1.21E-06	5.36E-02	1.71E-04	6.50E+01	6.15E+01	1.60E-01
27	3.45E-01	6.75E+00	2.74E-02	1.21E-06	5.36E-02	1.71E-04	3.12E+04	2.95E+04	1.60E-01
28	3.45E-01	6.75E+00	2.74E-02	1.21E-06	5.36E-02	1.71E-04	6.50E+01	6.15E+01	1.60E-01
33	1.36E+00	2.94E+01	7.67E-02	3.40E-06	2.56E-01	4.78E-04	1.89E+02	1.79E+02	4.49E-01
34	2.41E-01	3.63E+01	2.53E-01	1.12E-05	2.19E-01	1.58E-03	5.79E+02	5.47E+02	3.15E-02
44	3.45E-01	6.75E+00	2.74E-02	1.21E-06	5.36E-02	1.71E-04	6.50E+01	6.15E+01	1.60E-01
45	3.45E-01	6.75E+00	2.74E-02	1.21E-06	5.36E-02	1.71E-04	6.50E+01	6.15E+01	1.60E-01
46	3.45E-01	6.75E+00	2.74E-02	1.21E-06	5.36E-02	1.71E-04	6.50E+01	6.15E+01	1.60E-01
47	3.45E-01	6.75E+00	2.74E-02	1.21E-06	5.36E-02	1.71E-04	6.50E+01	6.15E+01	1.60E-01

Tank Group	Rh-106 (Ci)	Sb-125 (Ci)	Sn-126 (Ci)	I-129 (Ci)	Cs-134 (Ci)	Cs-135 (Ci)	Cs-137 (Ci)	Ba-137m (Ci)	Ce-144 (Ci)
Tanks 1- 8	2.84E-04	3.12E+00	1.31E+00	5.81E-05	7.52E-03	8.17E-03	2.48E+03	2.35E+03	4.96E-06
Tanks 17-20	2.82E-03	5.68E+00	3.01E-01	5.75E-05	1.93E-02	1.88E-03	6.14E+04	5.81E+04	1.42E-04
Tanks 25-28, 33-34, 44-47	4.36E+00	1.20E+02	5.49E-01	2.43E-05	9.04E-01	3.43E-03	3.92E+04	3.71E+04	1.76E+00

Table E-1: Radionuclide Inventory in FTF Tank Residual Material (Curies) (Continued)

Tank	Pr-144 (Ci)	Pm-147 (Ci)	Eu-154 (Ci)	U-232 (Ci)	U-233 (Ci)	U-234 (Ci)	U-235 (Ci)	U-236 (Ci)	U-238 (Ci)	Np-237 (Ci)
1	3.44E-11	1.21E+00	6.77E+00	8.95E-05	0.00E+00	0.00E+00	1.38E-04	0.00E+00	3.44E-03	5.02E-03
2	2.26E-14	2.34E-01	4.90E+00	9.42E-05	0.00E+00	0.00E+00	4.54E-05	0.00E+00	1.13E-03	1.18E-02
3	1.92E-13	1.87E-01	2.48E+00	4.16E-05	0.00E+00	0.00E+00	5.87E-05	0.00E+00	1.46E-03	1.09E-02
4	4.72E-06	2.99E+01	1.32E+01	7.99E-05	0.00E+00	0.00E+00	1.05E-04	0.00E+00	4.51E-03	4.03E-03
5	5.88E-10	2.69E+00	7.92E+00	8.25E-05	0.00E+00	0.00E+00	1.61E-04	0.00E+00	3.80E-03	6.43E-03
6	1.00E-08	8.38E+00	1.51E+01	1.23E-04	0.00E+00	0.00E+00	1.52E-04	0.00E+00	5.68E-03	1.97E-03
7	6.75E-08	1.72E+00	1.76E+00	1.46E-05	0.00E+00	0.00E+00	4.99E-05	0.00E+00	2.05E-03	9.29E-04
8	1.65E-07	4.10E+00	3.83E+00	2.76E-05	0.00E+00	0.00E+00	7.30E-05	0.00E+00	3.36E-03	1.53E-03
17	1.19E-04	2.48E+01	3.94E+00	4.68E-05	0.00+00	0.00+00	3.03E-04	0.00+00	1.83E-02	1.38E-02
18	1.44E-06	1.85E+01	1.06E+01	2.18E-04	4.41E+00	2.53E+00	9.31E-03	3.15E-02	2.37E-01	1.18E-01
19	3.71E-06	4.15E+01	1.47E+01	2.10E-04	9.83E-01	8.96E-02	1.36E-03	3.27E-03	4.54E-02	1.12E-02
20	1.80E-05	7.70E+00	1.00E+00	1.00E-05	0.00E+00	0.00E+00	6.40E-05	2.70E-05	5.80E-03	7.16E-04
25	1.60E-01	1.22E+02	6.30E+00	4.89E-05	1.49E-10	4.75E-10	1.41E-04	1.82E-05	1.69E-02	4.35E-05
26	1.60E-01	1.22E+02	6.30E+00	4.89E-05	1.49E-10	4.75E-10	1.41E-04	1.82E-05	1.69E-02	4.35E-05
27	1.60E-01	1.22E+02	6.30E+00	4.89E-05	1.49E-10	4.75E-10	1.41E-04	1.82E-05	1.69E-02	4.35E-05
28	1.60E-01	1.22E+02	6.30E+00	4.89E-05	1.49E-10	4.75E-10	1.41E-04	1.82E-05	1.69E-02	4.35E-05
33	4.49E-01	5.40E+02	2.03E+01	4.10E-05	1.96E-10	4.05E-06	7.60E-05	4.21E-05	1.10E-02	3.46E-03
34	3.15E-02	6.37E+02	5.07E+01	1.29E-04	0.00E+00	0.00E+00	1.31E-04	0.00E+00	9.23E-03	7.13E-03
44	1.60E-01	1.22E+02	6.30E+00	4.89E-05	1.49E-10	4.75E-10	1.41E-04	1.82E-05	1.69E-02	4.35E-05
45	1.60E-01	1.22E+02	6.30E+00	4.89E-05	1.49E-10	4.75E-10	1.41E-04	1.82E-05	1.69E-02	4.35E-05
46	1.60E-01	1.22E+02	6.30E+00	4.89E-05	1.49E-10	4.75E-10	1.41E-04	1.82E-05	1.69E-02	4.35E-05
47	1.60E-01	1.22E+02	6.30E+00	4.89E-05	1.49E-10	4.75E-10	1.41E-04	1.82E-05	1.69E-02	4.35E-05

Tank Group	Pr-144 (Ci)	Pm-147 (Ci)	Eu-154 (Ci)	U-232 (Ci)	U-233 (Ci)	U-234 (Ci)	U-235 (Ci)	U-236 (Ci)	U-238 (Ci)	Np-237 (Ci)
Tanks 1- 8	4.96E-06	4.84E+01	5.60E+01	5.53E-04	0.00E+00	0.00E+00	7.83E-04	0.00E+00	2.54E-02	4.26E-02
Tanks 17-20	1.42E-04	9.25E+01	3.02E+01	4.85E-04	5.39E+00	2.62E+00	1.10E-02	3.48E-02	3.07E-01	1.44E-01
Tanks 25-28, 33-34, 44-47	1.76E+00	2.15E+03	1.21E+02	5.61E-04	1.39E-09	4.05E-06	1.34E-03	1.88E-04	1.55E-01	1.09E-02

Table E-1: Radionuclide Inventory in FTF Tank Residual Material (Curies) (Continued)

Tank	Pu-238 (Ci)	Pu-239 (Ci)	Pu-240 (Ci)	Pu-241 (Ci)	Pu-242 (Ci)	Am-242m (Ci)	Na-22 (Ci)	Al-26 (Ci)	Te-125m (Ci)	Sb-126 (Ci)	Sb-126m (Ci)
1	3.90E+00	1.04E+00	2.33E-01	1.27E+00	4.79E-05	1.90E-02	5.24E-04	2.71E-04	2.13E-02	3.07E-02	2.19E-01
2	7.73E+00	1.10E+00	2.47E-01	1.03E+00	5.08E-05	2.04E-02	5.01E-04	2.65E-04	4.47E-03	3.36E-02	2.40E-01
3	4.82E+00	6.88E-01	1.54E-01	7.74E-01	3.16E-05	8.90E-03	2.28E-04	1.20E-04	3.47E-03	1.46E-02	1.04E-01
4	8.17E-01	8.07E-01	1.80E-01	1.52E+00	3.71E-05	1.64E-02	5.38E-04	2.84E-04	4.56E-01	2.52E-02	1.80E-01
5	4.43E+00	7.39E-01	1.77E-01	1.18E+00	5.21E-05	1.73E-02	4.99E-04	2.64E-04	4.57E-02	2.74E-02	1.96E-01
6	0.00E+00	5.81E-01	2.02E-01	2.26E+00	3.98E-04	2.55E-02	7.90E-04	4.17E-04	1.39E-01	3.96E-02	2.83E-01
7	3.82E+00	7.38E-01	1.78E-01	2.31E+00	1.99E-04	2.73E-03	8.70E-05	4.59E-05	2.71E-02	4.37E-03	3.12E-02
8	4.37E+00	9.60E-01	2.26E-01	2.14E+00	2.85E-04	5.40E-03	1.76E-04	9.29E-05	6.53E-02	8.46E-03	6.05E-02
17	6.51E+01	1.48E+01	3.36E+00	2.93E+02	5.32E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	7.01E+01	1.40E+02	5.81E+01	2.52E+02	5.60E-01	8.03E-01	0.00E+00	0.00E+00	2.87E-01	1.96E-02	1.40E-01
19	2.45E+01	3.12E+01	1.11E+01	5.49E+01	3.72E-01	1.93E+00	0.00E+00	0.00E+00	6.32E-01	1.78E-02	1.27E-01
20	7.36E+00	3.50E+00	7.64E-01	5.00E+01	1.60E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	8.99E+01	1.41E+01	3.15E+00	7.38E+01	6.27E-04	0.00E+00	1.20E-04	6.34E-05	1.65E+00	3.83E-03	2.74E-02
26	8.99E+01	1.41E+01	3.15E+00	7.38E+01	6.27E-04	0.00E+00	1.20E-04	6.34E-05	1.65E+00	3.83E-03	2.74E-02
27	8.99E+01	1.41E+01	3.15E+00	7.38E+01	6.27E-04	0.00E+00	1.20E-04	6.34E-05	1.65E+00	3.83E-03	2.74E-02
28	8.99E+01	1.41E+01	3.15E+00	7.38E+01	6.27E-04	0.00E+00	1.20E-04	6.34E-05	1.65E+00	3.83E-03	2.74E-02
33	5.64E+00	3.08E+00	5.40E-01	1.60E+01	3.13E-05	7.54E-03	3.50E-04	1.85E-04	7.19E+00	1.07E-02	7.67E-02
34	0.00E+00	1.49E+00	3.33E-01	6.87E+00	6.91E-05	2.47E-02	1.07E-03	5.63E-04	8.87E+00	3.55E-02	2.53E-01
44	8.99E+01	1.41E+01	3.15E+00	7.38E+01	6.27E-04	0.00E+00	1.20E-04	6.34E-05	1.65E+00	3.83E-03	2.74E-02
45	8.99E+01	1.41E+01	3.15E+00	7.38E+01	6.27E-04	0.00E+00	1.20E-04	6.34E-05	1.65E+00	3.83E-03	2.74E-02
46	8.99E+01	1.41E+01	3.15E+00	7.38E+01	6.27E-04	0.00E+00	1.20E-04	6.34E-05	1.65E+00	3.83E-03	2.74E-02
47	8.99E+01	1.41E+01	3.15E+00	7.38E+01	6.27E-04	0.00E+00	1.20E-04	6.34E-05	1.65E+00	3.83E-03	2.74E-02

Tank Group	Pu-238 (Ci)	Pu-239 (Ci)	Pu-240 (Ci)	Pu-241 (Ci)	Pu-242 (Ci)	Am-242m (Ci)	Na-22 (Ci)	Al-26 (Ci)	Te-125m (Ci)	Sb-126 (Ci)	Sb-126m (Ci)
Tanks 1- 8	2.99E+01	6.65E+00	1.60E+00	1.25E+01	1.10E-03	1.16E-01	3.34E-03	1.76E-03	7.62E-01	1.84E-01	1.31E+00
Tanks 17-20	1.67E+02	1.90E+02	7.33E+01	6.50E+02	9.39E-01	2.73E+00	0.00E+00	0.00E+00	9.19E-01	3.74E-02	3.74E-02
Tanks 25-28, 33-34, 44-47	7.25E+02	1.17E+02	2.61E+01	6.13E+02	5.12E-03	3.22E-02	2.38E-03	1.26E-03	2.93E+01	7.68E-02	5.49E-01

Table E-1: Radionuclide Inventory in FTF Tank Residual Material (Curies) (Continued)

Tank	Sm-151 (Ci)	Eu-152 (Ci)	Eu-155 (Ci)	Ra-226 (Ci)	Ac-227 (Ci)	Th-229 (Ci)	Th-230 (Ci)	Pa-231 (Ci)	Pu-244 (Ci)	Bk-249 (Ci)
1	3.40E+02	7.48E-01	3.81E+00	0.00E+00	6.07E-08	0.00E+00	0.00E+00	1.29E-07	2.23E-07	1.84E-31
2	3.59E+02	6.31E-01	1.86E+00	0.00E+00	2.39E-08	0.00E+00	0.00E+00	4.76E-08	2.37E-07	3.14E-34
3	1.58E+02	3.03E-01	1.08E+00	0.00E+00	2.85E-08	0.00E+00	0.00E+00	5.86E-08	1.47E-07	1.72E-33
4	3.00E+02	1.06E+00	1.65E+01	0.00E+00	3.80E-08	0.00E+00	0.00E+00	8.68E-08	1.73E-07	6.72E-27
5	3.12E+02	8.00E-01	5.49E+00	0.00E+00	6.45E-08	0.00E+00	0.00E+00	1.42E-07	2.43E-07	2.37E-30
6	4.64E+02	1.40E+00	1.25E+01	0.00E+00	5.19E-08	0.00E+00	0.00E+00	1.22E-07	1.85E-06	3.02E-29
7	5.10E+01	1.55E-01	1.67E+00	0.00E+00	1.38E-08	0.00E+00	0.00E+00	3.09E-08	7.02E-07	9.61E-29
8	1.00E+02	3.31E-01	3.86E+00	0.00E+00	2.76E-08	0.00E+00	0.00E+00	6.19E-08	1.33E-06	2.36E-28
17	0.00E+00	0.00E+00	8.07E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	4.58E+01	1.98E-01	2.67E+00	4.90E-07	1.64E-06	2.27E-03	6.00E-05	4.57E-06	3.39E-04	1.89E-28
19	1.30E+00	5.62E-03	7.57E-02	1.06E-07	2.85E-07	2.61E-03	1.29E-05	7.90E-07	1.66E-03	5.36E-30
20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	5.21E+01	3.55E-01	1.63E+01	3.03E-17	1.18E-08	2.45E-14	1.28E-14	4.77E-08	2.92E-06	3.38E-23
26	5.21E+01	3.55E-01	1.63E+01	3.03E-17	1.18E-08	2.45E-14	1.28E-14	4.77E-08	2.92E-06	3.38E-23
27	5.21E+01	3.55E-01	1.63E+01	3.03E-17	1.18E-08	2.45E-14	1.28E-14	4.77E-08	2.92E-06	3.38E-23
28	5.21E+01	3.55E-01	1.63E+01	3.03E-17	1.18E-08	2.45E-14	1.28E-14	4.77E-08	2.92E-06	3.38E-23
33	1.47E+02	1.08E+00	5.68E+01	3.24E-13	4.05E-09	9.69E-14	1.26E-10	1.83E-08	1.46E-07	1.09E-22
34	4.73E+02	3.01E+00	1.17E+02	0.00E+00	1.49E-08	0.00E+00	0.00E+00	5.56E-08	3.22E-07	1.74E-23
44	5.21E+01	3.55E-01	1.63E+01	3.03E-17	1.18E-08	2.45E-14	1.28E-14	4.77E-08	2.92E-06	3.38E-23
45	5.21E+01	3.55E-01	1.63E+01	3.03E-17	1.18E-08	2.45E-14	1.28E-14	4.77E-08	2.92E-06	3.38E-23
46	5.21E+01	3.55E-01	1.63E+01	3.03E-17	1.18E-08	2.45E-14	1.28E-14	4.77E-08	2.92E-06	3.38E-23
47	5.21E+01	3.55E-01	1.63E+01	3.03E-17	1.18E-08	2.45E-14	1.28E-14	4.77E-08	2.92E-06	3.38E-23

Tank Group	Sm-151 (Ci)	Eu-152 (Ci)	Eu-155 (Ci)	Ra-226 (Ci)	Ac-227 (Ci)	Th-229 (Ci)	Th-230 (Ci)	Pa-231 (Ci)	Pu-244 (Ci)	Bk-249 (Ci)
Tanks 1- 8	2.08E+03	5.43E+00	4.68E+01	0.00E+00	3.09E-07	0.00E+00	0.00E+00	6.79E-07	4.91E-06	7.08E-27
Tanks 17-20	4.71E+01	2.04E-01	2.83E+00	5.96E-07	1.93E-06	4.88E-03	7.29E-05	5.36E-06	2.00E-03	1.94E-28
Tanks 25-28, 33-34, 44-47	1.04E+03	6.93E+00	3.04E+02	3.24E-13	1.13E-07	2.93E-13	1.26E-10	4.56E-07	2.38E-05	3.97E-22

Table E-1: Radionuclide Inventory in FTF Tank Residual Material (Curies) (Continued)

Tank	Cf-249 (Ci)	Am-243 (Ci)	Cm-242 (Ci)	Cm-243 (Ci)	Cm-247 (Ci)	Cm-248 (Ci)	Cm-244 (Ci)	Cm-245 (Ci)	Am-241 (Ci)
1	1.13E-19	5.11E-05	2.76E-26	5.44E-04	1.81E-17	4.17E-18	2.74E-03	3.35E-09	1.55E+01
2	1.22E-19	5.60E-05	1.87E-32	5.31E-04	1.98E-17	4.57E-18	2.49E-03	3.66E-09	1.68E+01
3	5.32E-20	2.43E-05	3.96E-30	2.41E-04	8.59E-18	1.98E-18	1.16E-03	1.59E-09	7.39E+00
4	9.42E-20	8.78E-02	2.25E-17	5.69E-04	1.48E-17	3.42E-18	2.15E+01	9.32E-04	1.35E+01
5	1.01E-19	4.57E-05	2.33E-24	5.29E-04	1.62E-17	3.73E-18	2.80E-03	2.99E-09	1.40E+01
6	1.48E-19	4.57E-02	2.92E-22	8.37E-04	2.33E-17	5.38E-18	3.15E+00	4.32E-09	2.05E+01
7	1.63E-20	2.69E-02	5.49E-19	9.16E-05	2.58E-18	5.94E-19	4.32E+00	6.63E-05	2.64E+00
8	3.16E-20	6.69E-02	1.36E-18	1.86E-04	4.99E-18	1.15E-18	8.98E+00	1.65E-04	4.99E+00
17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.24E-04	4.40E-10	3.94E+01
18	1.39E-20	6.15E-06	3.19E-20	9.52E-05	2.17E-18	5.02E-19	1.69E+02	2.17E-09	7.41E+01
19	3.96E-22	1.75E-07	9.07E-22	2.70E-06	6.16E-20	1.42E-20	3.05E-03	1.97E-09	1.17E+01
20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.70E-04	9.10E-11	1.70E+00
25	1.49E-20	6.42E-06	1.20E-08	1.27E-04	2.27E-18	5.25E-19	9.07E-04	4.22E-10	3.19E+00
26	1.49E-20	6.42E-06	1.20E-08	1.27E-04	2.27E-18	5.25E-19	9.07E-04	4.22E-10	3.19E+00
27	1.49E-20	6.42E-06	1.20E-08	1.27E-04	2.27E-18	5.25E-19	9.07E-04	4.22E-10	3.19E+00
28	1.49E-20	6.42E-06	1.20E-08	1.27E-04	2.27E-18	5.25E-19	9.07E-04	4.22E-10	3.19E+00
33	4.17E-20	1.79E-05	1.16E-08	3.69E-04	6.33E-18	1.46E-18	2.74E-03	1.17E-09	8.74E+00
34	1.37E-19	5.93E-05	1.06E-10	5.94E-03	2.09E-17	4.83E-18	7.93E-03	3.87E-09	1.72E+02
44	1.49E-20	6.42E-06	1.20E-08	1.27E-04	2.27E-18	5.25E-19	9.07E-04	4.22E-10	3.19E+00
45	1.49E-20	6.42E-06	1.20E-08	1.27E-04	2.27E-18	5.25E-19	9.07E-04	4.22E-10	3.19E+00
46	1.49E-20	6.42E-06	1.20E-08	1.27E-04	2.27E-18	5.25E-19	9.07E-04	4.22E-10	3.19E+00
47	1.49E-20	6.42E-06	1.20E-08	1.27E-04	2.27E-18	5.25E-19	9.07E-04	4.22E-10	3.19E+00

Tank Group	Cf-249 (Ci)	Am-243 (Ci)	Cm-242 (Ci)	Cm-243 (Ci)	Cm-247 (Ci)	Cm-248 (Ci)	Cm-244 (Ci)	Cm-245 (Ci)	Am-241 (Ci)
Tanks 1- 8	6.79E-19	2.27E-01	2.44E-17	3.53E-03	1.08E-16	2.50E-17	3.80E+01	1.16E-03	9.53E+01
Tanks 17-20	1.43E-20	6.33E-06	3.28E-20	9.79E-05	2.23E-18	5.16E-19	1.69E+02	4.67E-09	1.27E+02
Tanks 25-28, 33-34, 44-47	2.98E-19	1.29E-04	1.08E-07	7.33E-03	4.54E-17	1.05E-17	1.79E-02	8.42E-09	2.06E+02

Table E-2: Radionuclide Concentration in FTF Tank Residual Material (Curies/Gram)

Tank	H-3 (Ci/g)	C-14 (Ci/g)	Co-60 (Ci/g)	Ni-59 (Ci/g)	Ni-63 (Ci/g)	Se-79 (Ci/g)	Sr-90 (Ci/g)	Y-90 (Ci/g)	Nb-94 (Ci/g)	Tc-99 (Ci/g)	Ru-106 (Ci/g)
1	0.00E+00	1.12E-09	1.72E-05	2.00E-06	1.65E-04	1.40E-06	4.82E-02	4.82E-02	1.05E-09	2.42E-05	3.93E-13
2	0.00E+00	1.90E-09	8.91E-06	2.24E-06	1.78E-04	1.53E-06	4.70E-02	4.70E-02	1.14E-09	2.64E-05	1.94E-15
3	0.00E+00	8.21E-10	5.08E-06	9.68E-07	7.77E-05	6.62E-07	2.13E-02	2.13E-02	4.95E-10	1.15E-05	6.62E-15
4	0.00E+00	5.34E-11	6.67E-05	1.60E-06	1.40E-04	1.14E-06	5.04E-02	5.04E-02	8.58E-10	1.98E-05	3.16E-09
5	0.00E+00	8.05E-10	2.41E-05	1.79E-06	1.49E-04	1.24E-06	4.68E-02	4.68E-02	9.35E-10	2.16E-05	3.60E-12
6	0.00E+00	0.00E+00	5.37E-05	2.51E-06	2.17E-04	1.80E-06	7.41E-02	7.41E-02	1.35E-09	3.12E-05	3.33E-11
7	0.00E+00	1.26E-09	7.06E-06	3.45E-07	2.97E-05	2.00E-07	8.18E-03	8.18E-03	1.49E-10	3.46E-06	4.94E-11
8	0.00E+00	1.14E-09	1.61E-05	5.97E-07	5.23E-05	3.85E-07	1.65E-02	1.65E-02	2.88E-10	6.66E-06	1.20E-10
17	1.10E-05	1.54E-09	1.13E-06	9.03E-08	0.00E+00	7.70E-09	4.12E-04	4.12E-04	0.00E+00	1.77E-06	9.97E-10
18	3.69E-09	3.36E-09	2.61E-07	5.47E-08	4.82E-06	3.03E-09	8.45E-05	8.45E-05	2.40E-12	8.87E-08	5.83E-12
19	6.83E-09	2.02E-09	1.32E-07	1.49E-08	1.38E-06	1.06E-09	7.23E-07	7.23E-07	6.56E-13	1.22E-07	4.88E-12
20	0.00E+00	7.82E-10	7.94E-07	4.62E-08	0.00E+00	3.79E-09	2.25E-04	2.25E-04	0.00E+00	1.01E-06	5.09E-10
25	0.00E+00	3.59E-09	6.05E-06	2.11E-07	2.06E-05	1.74E-08	1.13E-03	1.13E-03	9.25E-12	3.02E-07	4.09E-07
26	0.00E+00	3.59E-09	6.05E-06	2.11E-07	2.06E-05	1.74E-08	1.13E-03	1.13E-03	9.25E-12	3.02E-07	4.09E-07
27	0.00E+00	3.59E-09	6.05E-06	2.11E-07	2.06E-05	1.74E-08	1.13E-03	1.13E-03	9.25E-12	3.02E-07	4.09E-07
28	0.00E+00	3.59E-09	6.05E-06	2.11E-07	2.06E-05	1.74E-08	1.13E-03	1.13E-03	9.25E-12	3.02E-07	4.09E-07
33	0.00E+00	5.39E-10	2.13E-04	7.11E-07	7.04E-05	4.89E-07	3.28E-02	3.28E-02	3.66E-10	8.45E-06	1.61E-05
34	0.00E+00	0.00E+00	4.38E-04	2.25E-06	2.18E-04	1.61E-06	9.99E-02	9.99E-02	1.21E-09	2.80E-05	2.86E-06
44	0.00E+00	3.59E-09	6.05E-06	2.11E-07	2.06E-05	1.74E-08	1.13E-03	1.13E-03	9.25E-12	3.02E-07	4.09E-07
45	0.00E+00	3.59E-09	6.05E-06	2.11E-07	2.06E-05	1.74E-08	1.13E-03	1.13E-03	9.25E-12	3.02E-07	4.09E-07
46	0.00E+00	3.59E-09	6.05E-06	2.11E-07	2.06E-05	1.74E-08	1.13E-03	1.13E-03	9.25E-12	3.02E-07	4.09E-07
47	0.00E+00	3.59E-09	6.05E-06	2.11E-07	2.06E-05	1.74E-08	1.13E-03	1.13E-03	9.25E-12	3.02E-07	4.09E-07

Tank Group	H-3 (Ci/g)	C-14 (Ci/g)	Co-60 (Ci/g)	Ni-59 (Ci/g)	Ni-63 (Ci/g)	Se-79 (Ci/g)	Sr-90 (Ci/g)	Y-90 (Ci/g)	Nb-94 (Ci/g)	Tc-99 (Ci/g)	Ru-106 (Ci/g)
Tanks 1- 8	0.00E+00	8.86E-10	2.485E-05	1.51E-06	1.26E-04	1.05E-06	3.91E-02	3.91E-02	7.83E-10	1.81E-05	4.21E-10
Tanks 17-20	3.02E-07	2.30E-09	1.95E-07	2.62E-08	2.09E-06	1.71E-09	3.30E-05	3.30E-05	1.02E-12	1.70E-07	3.76E-11
Tanks 25-28, 33-34, 44-47	0.00E+00	3.51E-09	1.39E-05	2.42E-07	2.36E-05	4.26E-08	2.72E-03	2.72E-03	2.82E-11	7.39E-07	6.30E-07

Table E-2: Radionuclide Concentration in FTF Tank Residual Material (Curies/Gram) (Continued)

Tank	Rh-106 (Ci/g)	Sb-125 (Ci/g)	Sn-126 (Ci/g)	I-129 (Ci/g)	Cs-134 (Ci/g)	Cs-135 (Ci/g)	Cs-137 (Ci/g)	Ba-137m (Ci/g)	Ce-144 (Ci/g)
1	3.93E-13	1.04E-06	2.59E-06	1.15E-10	9.62E-10	1.61E-08	3.41E-03	3.23E-03	4.08E-16
2	1.94E-15	2.17E-07	2.84E-06	1.26E-10	1.03E-10	1.77E-08	3.35E-03	3.18E-03	2.68E-19
3	6.62E-15	1.68E-07	1.23E-06	5.46E-11	1.01E-10	7.67E-09	1.52E-03	1.43E-03	2.27E-18
4	3.16E-09	2.22E-05	2.13E-06	9.43E-11	6.56E-08	1.33E-08	3.52E-03	3.33E-03	5.59E-11
5	3.60E-12	2.22E-06	2.32E-06	1.03E-10	2.75E-09	1.45E-08	3.31E-03	3.13E-03	6.97E-15
6	3.33E-11	6.73E-06	3.35E-06	1.48E-10	1.02E-08	2.09E-08	5.20E-03	4.92E-03	1.18E-13
7	4.94E-11	1.33E-06	3.72E-07	1.65E-11	2.75E-09	2.32E-09	7.95E-03	7.52E-03	8.00E-13
8	1.20E-10	3.16E-06	7.17E-07	3.18E-11	6.60E-09	4.45E-09	1.16E-03	1.09E-03	1.95E-12
17	9.97E-10	7.26E-07	1.40E-08	6.32E-13	3.17E-09	8.59E-11	2.85E-05	2.70E-05	5.87E-11
18	5.83E-12	7.01E-08	8.39E-09	3.85E-13	1.97E-10	5.24E-11	7.37E-04	6.95E-04	8.63E-14
19	4.88E-12	4.68E-08	2.30E-09	8.95E-13	1.37E-10	1.43E-11	8.86E-04	8.39E-04	6.71E-14
20	5.09E-10	5.33E-07	6.99E-09	3.08E-13	2.37E-09	4.27E-11	4.86E-07	4.60E-05	2.13E-11
25	4.09E-07	8.00E-06	3.25E-08	1.43E-12	6.35E-08	2.03E-10	8.09E-03	7.65E-03	1.90E-07
26	4.09E-07	8.00E-06	3.25E-08	1.43E-12	6.35E-08	2.03E-10	7.70E-05	7.29E-05	1.90E-07
27	4.09E-07	8.00E-06	3.25E-08	1.43E-12	6.35E-08	2.03E-10	3.70E-02	3.50E-02	1.90E-07
28	4.09E-07	8.00E-06	3.25E-08	1.43E-12	6.35E-08	2.03E-10	7.70E-05	7.29E-05	1.90E-07
33	1.61E-05	3.48E-04	9.09E-07	4.03E-11	3.03E-06	5.66E-09	2.24E-03	2.12E-03	5.32E-06
34	2.86E-06	4.30E-04	3.00E-06	1.33E-10	2.59E-06	1.87E-08	6.86E-03	6.48E-03	3.73E-07
44	4.09E-07	8.00E-06	3.25E-08	1.43E-12	6.35E-08	2.03E-10	7.70E-05	7.29E-05	1.90E-07
45	4.09E-07	8.00E-06	3.25E-08	1.43E-12	6.35E-08	2.03E-10	7.70E-05	7.29E-05	1.90E-07
46	4.09E-07	8.00E-06	3.25E-08	1.43E-12	6.35E-08	2.03E-10	7.70E-05	7.29E-05	1.90E-07
47	4.09E-07	8.00E-06	3.25E-08	1.43E-12	6.35E-08	2.03E-10	7.70E-05	7.29E-05	1.90E-07

Tank Group	Rh-106 (Ci/g)	Sb-125 (Ci/g)	Sn-126 (Ci/g)	I-129 (Ci/g)	Cs-134 (Ci/g)	Cs-135 (Ci/g)	Cs-137 (Ci/g)	Ba-137m (Ci/g)	Ce-144 (Ci/g)
Tanks 1- 8	4.21E-10	4.63E-06	1.95E-06	8.61E-11	1.11E-08	1.21E-08	3.68E-03	3.48E-03	7.35E-12
Tanks 17-20	3.77E-11	7.59E-08	4.02E-09	7.67E-13	2.58E-10	2.50E-11	8.19E-04	7.76E-04	1.90E-12
Tanks 25-28, 33-34, 44-47	6.30E-07	1.73E-05	7.93E-08	3.51E-12	1.31E-07	4.95E-10	5.66E-03	5.35E-03	2.54E-07

Table E-2: Radionuclide Concentration in FTF Tank Residual Material (Curies/Gram) (Continued)

Tank	Pr-144 (Ci/g)	Pm-147 (Ci/g)	Eu-154 (Ci/g)	U-232 (Ci/g)	U-233 (Ci/g)	U-234 (Ci/g)	U-235 (Ci/g)	U-236 (Ci/g)	U-238 (Ci/g)	Np-237 (Ci/g)
1	4.08E-16	1.43E-05	8.02E-05	1.06E-09	0.00E+00	0.00E+00	1.64E-09	0.00E+00	4.08E-08	5.95E-08
2	2.68E-19	2.77E-06	5.81E-05	1.12E-09	0.00E+00	0.00E+00	5.38E-10	0.00E+00	1.34E-08	1.40E-07
3	2.27E-18	2.22E-06	2.94E-05	4.93E-10	0.00E+00	0.00E+00	6.95E-10	0.00E+00	1.73E-08	1.29E-07
4	5.59E-11	3.54E-04	1.56E-04	9.47E-10	0.00E+00	0.00E+00	1.24E-09	0.00E+00	5.34E-08	4.77E-08
5	6.97E-15	3.19E-05	9.38E-05	9.77E-10	0.00E+00	0.00E+00	1.91E-09	0.00E+00	4.50E-08	7.62E-08
6	1.18E-13	9.93E-05	1.79E-04	1.46E-09	0.00E+00	0.00E+00	1.80E-09	0.00E+00	6.73E-08	2.33E-08
7	8.00E-13	2.04E-05	2.09E-05	1.73E-10	0.00E+00	0.00E+00	5.91E-10	0.00E+00	2.43E-08	1.10E-08
8	1.95E-12	4.86E-05	4.54E-05	3.27E-10	0.00E+00	0.00E+00	8.65E-10	0.00E+00	3.98E-08	1.81E-08
17	5.87E-11	1.22E-05	1.95E-06	2.31E-11	0.00+00	0.00+00	1.50E-10	0.00+00	9.03E-09	6.81E-09
18	8.63E-14	1.11E-06	6.35E-07	1.31E-11	2.64E-07	1.52E-07	5.58E-10	1.89E-09	1.42E-08	7.07E-09
19	6.71E-14	7.50E-07	2.66E-07	3.80E-12	1.78E-08	1.62E-09	2.46E-11	5.91E-11	8.21E-10	2.02E-10
20	2.13E-11	9.12E-06	1.18E-06	1.18E-11	0.00E+00	0.00E+00	7.58E-11	3.20E-11	6.87E-09	8.48E-10
25	1.90E-07	1.45E-04	7.46E-06	5.79E-11	1.77E-16	5.63E-16	1.67E-10	2.16E-11	2.00E-08	5.15E-11
26	1.90E-07	1.45E-04	7.46E-06	5.79E-11	1.77E-16	5.63E-16	1.67E-10	2.16E-11	2.00E-08	5.15E-11
27	1.90E-07	1.45E-04	7.46E-06	5.79E-11	1.77E-16	5.63E-16	1.67E-10	2.16E-11	2.00E-08	5.15E-11
28	1.90E-07	1.45E-04	7.46E-06	5.79E-11	1.77E-16	5.63E-16	1.67E-10	2.16E-11	2.00E-08	5.15E-11
33	5.32E-06	6.40E-03	2.41E-04	4.86E-10	2.32E-15	4.80E-11	9.00E-10	4.99E-10	1.30E-07	4.10E-08
34	3.73E-07	7.55E-03	6.01E-04	1.53E-09	0.00E+00	0.00E+00	1.55E-09	0.00E+00	1.09E-07	8.45E-08
44	1.90E-07	1.45E-04	7.46E-06	5.79E-11	1.77E-16	5.63E-16	1.67E-10	2.16E-11	2.00E-08	5.15E-11
45	1.90E-07	1.45E-04	7.46E-06	5.79E-11	1.77E-16	5.63E-16	1.67E-10	2.16E-11	2.00E-08	5.15E-11
46	1.90E-07	1.45E-04	7.46E-06	5.79E-11	1.77E-16	5.63E-16	1.67E-10	2.16E-11	2.00E-08	5.15E-11
47	1.90E-07	1.45E-04	7.46E-06	5.79E-11	1.77E-16	5.63E-16	1.67E-10	2.16E-11	2.00E-08	5.15E-11

Tank Group	Pr-144 (Ci/g)	Pm-147 (Ci/g)	Eu-154 (Ci/g)	U-232 (Ci/g)	U-233 (Ci/g)	U-234 (Ci/g)	U-235 (Ci/g)	U-236 (Ci/g)	U-238 (Ci/g)	Np-237 (Ci/g)
Tanks 1- 8	7.35E-12	7.17E-05	8.29E-05	8.19E-10	0.00E+00	0.00E+00	1.16E-09	0.00E+00	3.77E-08	6.31E-08
Tanks 17-20	1.90E-12	1.24E-06	4.04E-07	6.47E-12	7.20E-08	3.50E-08	1.47E-10	4.65E-10	4.10E-09	1.92E-09
Tanks 25-28, 33-34, 44-47	2.54E-07	3.11E-04	1.75E-05	8.11E-11	2.01E-16	5.86E-13	1.93E-10	2.71E-11	2.25E-08	1.58E-09

Table E-2: Radionuclide Concentration in FTF Tank Residual Material (Curies/Gram) (Continued)

Tank	Pu-238 (Ci/g)	Pu-239 (Ci/g)	Pu-240 (Ci/g)	Pu-241 (Ci/g)	Pu-242 (Ci/g)	Am-242m (Ci/g)	Na-22 (Ci/g)	Al-26 (Ci/g)	Te-125m (Ci/g)	Sb-126 (Ci/g)	Sb-126m (Ci/g)
1	4.62E-05	1.23E-05	2.76E-06	1.50E-05	5.68E-10	2.25E-07	6.21E-09	3.21E-09	2.52E-07	3.64E-07	2.59E-06
2	9.16E-05	1.30E-05	2.93E-06	1.22E-05	6.02E-10	2.42E-07	5.94E-09	3.14E-09	5.30E-08	3.98E-07	2.84E-06
3	5.71E-05	8.15E-06	1.82E-06	9.17E-06	3.74E-10	1.05E-07	2.70E-09	1.42E-09	4.11E-08	1.73E-07	1.23E-06
4	9.68E-06	9.56E-06	2.13E-06	1.80E-05	4.40E-10	1.94E-07	6.37E-09	3.36E-09	5.40E-06	2.99E-07	2.13E-06
5	5.25E-05	8.76E-06	2.10E-06	1.40E-05	6.17E-10	2.05E-07	5.91E-09	3.13E-09	5.41E-07	3.25E-07	2.32E-06
6	0.00E+00	6.88E-06	2.39E-06	2.68E-05	4.72E-09	3.02E-07	9.36E-09	4.94E-09	1.65E-06	4.69E-07	3.35E-06
7	4.53E-05	8.74E-06	2.11E-06	2.74E-05	2.36E-09	3.23E-08	1.03E-09	5.44E-10	3.21E-07	5.18E-08	3.70E-07
8	5.18E-05	1.14E-05	2.68E-06	2.54E-05	3.38E-09	6.40E-08	2.09E-09	1.10E-09	7.74E-07	1.00E-07	7.17E-07
17	3.21E-05	7.31E-06	1.66E-06	1.45E-04	2.63E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	4.20E-06	8.39E-06	3.48E-06	1.51E-05	3.36E-08	4.81E-08	0.00E+00	0.00E+00	1.72E-08	1.17E-09	8.39E-09
19	4.43E-07	5.64E-07	2.01E-07	9.92E-07	6.72E-09	3.49E-08	0.00E+00	0.00E+00	1.14E-08	3.22E-10	2.30E-09
20	8.72E-06	4.15E-06	9.05E-07	5.92E-05	1.90E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	1.07E-04	1.67E-05	3.73E-06	8.74E-05	7.43E-10	0.00E+00	1.42E-10	7.51E-11	1.95E-06	4.54E-09	3.25E-08
26	1.07E-04	1.67E-05	3.73E-06	8.74E-05	7.43E-10	0.00E+00	1.42E-10	7.51E-11	1.95E-06	4.54E-09	3.25E-08
27	1.07E-04	1.67E-05	3.73E-06	8.74E-05	7.43E-10	0.00E+00	1.42E-10	7.51E-11	1.95E-06	4.54E-09	3.25E-08
28	1.07E-04	1.67E-05	3.73E-06	8.74E-05	7.43E-10	0.00E+00	1.42E-10	7.51E-11	1.95E-06	4.54E-09	3.25E-08
33	6.68E-05	3.65E-05	6.40E-06	1.90E-04	3.71E-10	8.93E-08	4.15E-09	2.19E-09	8.52E-05	1.27E-07	9.09E-07
34	0.00E+00	1.77E-05	3.95E-06	8.14E-05	8.19E-10	2.93E-07	1.27E-08	6.67E-09	1.05E-04	4.21E-07	3.00E-06
44	1.07E-04	1.67E-05	3.73E-06	8.74E-05	7.43E-10	0.00E+00	1.42E-10	7.51E-11	1.95E-06	4.54E-09	3.25E-08
45	1.07E-04	1.67E-05	3.73E-06	8.74E-05	7.43E-10	0.00E+00	1.42E-10	7.51E-11	1.95E-06	4.54E-09	3.25E-08
46	1.07E-04	1.67E-05	3.73E-06	8.74E-05	7.43E-10	0.00E+00	1.42E-10	7.51E-11	1.95E-06	4.54E-09	3.25E-08
47	1.07E-04	1.67E-05	3.73E-06	8.74E-05	7.43E-10	0.00E+00	1.42E-10	7.51E-11	1.95E-06	4.54E-09	3.25E-08

Tank Group	Pu-238 (Ci/g)	Pu-239 (Ci/g)	Pu-240 (Ci/g)	Pu-241 (Ci/g)	Pu-242 (Ci/g)	Am-242m (Ci/g)	Na-22 (Ci/g)	Al-26 (Ci/g)	Te-125m (Ci/g)	Sb-126 (Ci/g)	Sb-126m (Ci/g)
Tanks 1- 8	4.43E-05	9.85E-06	2.37E-06	1.85E-05	1.63E-09	1.71E-07	4.95E-09	2.61E-09	1.13E-06	2.72E-07	1.95E-06
Tanks 17-20	2.23E-06	2.53E-06	9.79E-07	8.68E-06	1.25E-08	3.65E-08	0.00E+00	0.00E+00	1.23E-08	4.99E-10	4.99E-10
Tanks 25-28, 33-34, 44-47	1.05E-04	1.70E-05	3.77E-06	8.86E-05	7.39E-10	4.66E-09	3.44E-10	1.81E-10	4.23E-06	1.11E-08	7.93E-08

Table E-2: Radionuclide Concentration in FTF Tank Residual Material (Curies/Gram) (Continued)

Tank	Sm-151 (Ci/g)	Eu-152 (Ci/g)	Eu-155 (Ci/g)	Ra-226 (Ci/g)	Ac-227 (Ci/g)	Th-229 (Ci/g)	Th-230 (Ci/g)	Pa-231 (Ci/g)	Pu-244 (Ci/g)	Bk-249 (Ci/g)
1	4.03E-03	8.86E-06	4.51E-05	0.00E+00	7.19E-13	0.00E+00	0.00E+00	1.53E-12	2.64E-12	2.18E-36
2	4.25E-03	7.48E-06	2.20E-05	0.00E+00	2.83E-13	0.00E+00	0.00E+00	5.64E-13	2.81E-12	3.72E-39
3	1.87E-03	3.59E-06	1.28E-05	0.00E+00	3.38E-13	0.00E+00	0.00E+00	6.94E-13	1.74E-12	2.04E-38
4	3.55E-03	1.26E-05	1.95E-04	0.00E+00	4.50E-13	0.00E+00	0.00E+00	1.03E-12	2.05E-12	7.96E-32
5	3.70E-03	9.48E-06	6.50E-05	0.00E+00	7.64E-13	0.00E+00	0.00E+00	1.68E-12	2.88E-12	2.81E-35
6	5.50E-03	1.66E-05	1.48E-04	0.00E+00	6.15E-13	0.00E+00	0.00E+00	1.45E-12	2.19E-11	3.58E-34
7	6.04E-04	1.84E-06	1.98E-05	0.00E+00	1.64E-13	0.00E+00	0.00E+00	3.66E-13	8.32E-12	1.14E-33
8	1.18E-03	3.92E-06	4.57E-05	0.00E+00	3.27E-13	0.00E+00	0.00E+00	7.33E-13	1.58E-11	2.80E-33
17	0.00E+00	0.00E+00	3.98E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
18	2.74E-06	1.19E-08	1.60E-07	2.94E-14	9.83E-14	1.36E-10	3.60E-12	2.74E-13	2.03E-11	1.13E-35
19	2.35E-08	1.02E-10	1.37E-09	1.92E-15	5.15E-15	4.72E-11	2.33E-13	1.43E-14	3.00E-11	9.69E-38
20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
25	6.17E-05	4.21E-07	1.93E-05	3.59E-23	1.40E-14	2.90E-20	1.52E-20	5.65E-14	3.46E-12	4.00E-29
26	6.17E-05	4.21E-07	1.93E-05	3.59E-23	1.40E-14	2.90E-20	1.52E-20	5.65E-14	3.46E-12	4.00E-29
27	6.17E-05	4.21E-07	1.93E-05	3.59E-23	1.40E-14	2.90E-20	1.52E-20	5.65E-14	3.46E-12	4.00E-29
28	6.17E-05	4.21E-07	1.93E-05	3.59E-23	1.40E-14	2.90E-20	1.52E-20	5.65E-14	3.46E-12	4.00E-29
33	1.74E-03	1.28E-05	6.73E-04	3.84E-18	4.80E-14	1.15E-18	1.49E-15	2.17E-13	1.73E-12	1.29E-27
34	5.60E-03	3.57E-05	1.39E-03	0.00E+00	1.77E-13	0.00E+00	0.00E+00	6.59E-13	3.82E-12	2.06E-28
44	6.17E-05	4.21E-07	1.93E-05	3.59E-23	1.40E-14	2.90E-20	1.52E-20	5.65E-14	3.46E-12	4.00E-29
45	6.17E-05	4.21E-07	1.93E-05	3.59E-23	1.40E-14	2.90E-20	1.52E-20	5.65E-14	3.46E-12	4.00E-29
46	6.17E-05	4.21E-07	1.93E-05	3.59E-23	1.40E-14	2.90E-20	1.52E-20	5.65E-14	3.46E-12	4.00E-29
47	6.17E-05	4.21E-07	1.93E-05	3.59E-23	1.40E-14	2.90E-20	1.52E-20	5.65E-14	3.46E-12	4.00E-29

Tank Group	Sm-151 (Ci)/g	Eu-152 (Ci/g)	Eu-155 (Ci/g)	Ra-226 (Ci/g)	Ac-227 (Ci/g)	Th-229 (Ci/g)	Th-230 (Ci/g)	Pa-231 (Ci/g)	Pu-244 (Ci/g)	Bk-249 (Ci/g)
Tanks 1- 8	3.09E-03	8.04E-06	6.93E-05	0.00E+00	4.57E-13	0.00E+00	0.00E+00	1.01E-12	7.26E-12	1.05E-32
Tanks 17-20	6.29E-07	2.72E-09	3.77E-08	7.96E-15	2.57E-14	6.52E-11	9.74E-13	7.16E-14	2.67E-11	2.60E-36
Tanks 25-28, 33-34, 44-47	1.50E-04	1.00E-06	4.393E-05	4.69E-20	1.64E-14	4.23E-20	1.82E-17	6.59E-14	3.44E-12	5.73E-29

Table E-2: Radionuclide Concentration in FTF Tank Residual Material (Curies/Gram) (Continued)

Tank	Cf-249 (Ci/g)	Am-243 (Ci/g)	Cm-242 (Ci/g)	Cm-243 (Ci/g)	Cm-247 (Ci/g)	Cm-248 (Ci/g)	Cm-244 (Ci/g)	Cm-245 (Ci/g)	Am-241 (Ci/g)
1	1.34E-24	6.05E-10	3.27E-31	6.45E-09	2.14E-22	4.94E-23	3.25E-08	3.97E-14	1.84E-04
2	1.45E-24	6.64E-10	2.22E-37	6.29E-09	2.35E-22	5.41E-23	2.95E-08	4.34E-14	1.99E-04
3	6.30E-25	2.88E-10	4.69E-35	2.86E-09	1.02E-22	2.35E-23	1.37E-08	1.88E-14	8.76E-05
4	1.12E-24	1.04E-06	2.67E-22	6.74E-09	1.75E-22	4.05E-23	2.55E-04	1.10E-08	1.60E-04
5	1.20E-24	5.41E-10	2.76E-29	6.27E-09	1.92E-22	4.42E-23	3.32E-08	3.54E-14	1.66E-04
6	1.75E-24	5.41E-07	3.46E-27	9.92E-09	2.76E-22	6.37E-23	3.73E-05	5.12E-14	2.43E-04
7	1.93E-25	3.19E-07	6.50E-24	1.09E-09	3.06E-23	7.04E-24	5.12E-05	7.86E-10	3.13E-05
8	3.74E-25	7.93E-07	1.61E-23	2.20E-09	5.91E-23	1.36E-23	1.06E-04	1.95E-09	5.91E-05
17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.57E-10	2.17E-16	1.95E-05
18	8.33E-28	3.69E-13	1.91E-27	5.71E-12	1.30E-25	3.01E-26	1.01E-05	1.30E-16	4.44E-06
19	7.16E-30	3.16E-15	1.64E-29	4.88E-14	1.11E-27	2.57E-28	5.51E-11	3.56E-17	2.11E-07
20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.01E-10	1.08E-16	2.01E-06
25	1.77E-26	7.61E-12	1.42E-14	1.50E-10	2.69E-24	6.22E-25	1.07E-09	5.00E-16	3.78E-06
26	1.77E-26	7.61E-12	1.42E-14	1.50E-10	2.69E-24	6.22E-25	1.07E-09	5.00E-16	3.78E-06
27	1.77E-26	7.61E-12	1.42E-14	1.50E-10	2.69E-24	6.22E-25	1.07E-09	5.00E-16	3.78E-06
28	1.77E-26	7.61E-12	1.42E-14	1.50E-10	2.69E-24	6.22E-25	1.07E-09	5.00E-16	3.78E-06
33	4.94E-25	2.12E-10	1.37E-13	4.37E-09	7.50E-23	1.73E-23	3.25E-08	1.39E-14	1.04E-04
34	1.62E-24	7.03E-10	1.26E-15	7.04E-08	2.48E-22	5.72E-23	9.40E-08	4.59E-14	2.04E-03
44	1.77E-26	7.61E-12	1.42E-14	1.50E-10	2.69E-24	6.22E-25	1.07E-09	5.00E-16	3.78E-06
45	1.77E-26	7.61E-12	1.42E-14	1.50E-10	2.69E-24	6.22E-25	1.07E-09	5.00E-16	3.78E-06
46	1.77E-26	7.61E-12	1.42E-14	1.50E-10	2.69E-24	6.22E-25	1.07E-09	5.00E-16	3.78E-06
47	1.77E-26	7.61E-12	1.42E-14	1.50E-10	2.69E-24	6.22E-25	1.07E-09	5.00E-16	3.78E-06

Tank Group	Cf-249 (Ci/g)	Am-243 (Ci/g)	Cm-242 (Ci/g)	Cm-243 (Ci/g)	Cm-247 (Ci/g)	Cm-248 (Ci/g)	Cm-244 (Ci/g)	Cm-245 (Ci/g)	Am-241 (Ci/g)
Tanks 1- 8	1.01E-24	3.37E-07	3.62E-23	5.23E-09	1.60E-22	3.70E-23	5.62E-05	1.72E-09	1.41E-04
Tanks 17-20	1.91E-28	8.45E-14	4.38E-28	1.31E-12	2.98E-26	6.89E-27	2.26E-06	6.24E-17	1.69E-06
Tanks 25-28, 33-34, 44-47	4.30E-26	1.86E-11	1.56E-14	1.06E-09	6.56E-24	1.52E-24	2.59E-09	1.22E-15	2.98E-05

E.2 FTF ANCILLARY EQUIPMENT RESIDUAL MATERIAL INVENTORY

Modeling of the residual material inventory in the FTF ancillary equipment was performed and this ancillary equipment will be addressed in a future Section 3116 Determination. FTF contains ancillary equipment with a residual radiological inventory that must be accounted for upon facility closure. This ancillary equipment includes buried pipe (transfer lines), pump tanks, and evaporators, all of which have been in contact with radioactive waste over the operating life of the facility. The amount of contamination on these components depends on such factors as the service life of the component, its materials of construction, and the contaminating medium in contact with the component. The ancillary equipment residual material inventory is small when compared to the inventory associated with the residual material in waste tanks. [6]

The contaminating medium is radioactive waste, and a representative inventory for the FTF waste stream was derived based on the types of waste that have historically been transferred through the systems. [7] The total inventory assigned to FTF ancillary equipment is listed in Table E-3. Additional information on the FTF ancillary equipment residual material inventory is described in the PODD. [6]

Table E-3: Radionuclide Inventory for FTF Ancillary Equipment

Isotope	Total Curies	Curies per Total Mass (Ci/g)
H-3	2.61E+01	8.38E-06
C-14	1.97E-01	6.34E-08
Al-26	1.66E-02	5.32E-09
Co-60	5.47E+01	1.76E-05
Ni-59	1.16E+01	3.73E-06
Ni-63	1.05E+03	3.38E-04
Se-79	7.57E-02	2.43E-08
Sr-90	1.05E+04	3.38E-03
Y-90	1.05E+04	3.38E-03
Nb-94	5.71E-05	1.83E-11
Tc-99	1.81E+00	5.81E-07
Rh-106	1.40E+00	4.48E-07
Ru-106	1.40E+00	4.48E-07
Te-125m	1.33E+01	4.28E-06
Sb-125	5.46E+01	1.75E-05
Sb-126	3.27E-02	1.05E-08
Sb-126m	2.33E-01	7.49E-08
Sn-126	2.33E-01	7.49E-08
I-129	7.98E-02	2.56E-08
Cs-134	3.94E+01	1.27E-05
Cs-135	1.19E-02	3.82E-09
Cs-137	1.38E+04	4.42E-03
Ba-137m	1.30E+04	4.18E-03
Ce-144	1.97E-01	6.31E-08
Pr-144	1.97E-01	6.31E-08
Pm-147	9.19E+02	2.95E-04
Sm-151	2.75E+02	8.82E-05
Eu-152	2.14E+00	6.86E-07
Eu-154	1.15E+02	3.70E-05
Eu-155	1.05E+02	3.37E-05
Ra-226	1.14E-04	3.66E-11

Isotope	Total Curies	Curies per Total Mass (Ci/g)
Ra-228	0.00E+00	0.00E+00
Ac-227	3.44E-06	1.10E-12
Th-229	9.82E+00	3.15E-06
Th-230	1.50E-04	4.82E-11
Th-232	0.00E+00	0.00E+00
Pa-231	3.46E-06	1.11E-12
U-232	7.32E-03	2.35E-09
U-233	1.58E+01	5.06E-06
U-234	1.82E-02	5.85E-09
U-235	5.78E-05	1.86E-11
U-236	5.94E-04	1.91E-10
U-238	3.26E-03	1.05E-09
Np-237	5.47E-03	1.76E-09
Pu-238	4.12E+02	1.32E-04
Pu-239	7.59E+00	2.44E-06
Pu-240	3.92E+00	1.26E-06
Pu-241	3.10E+02	9.95E-05
Pu-242	2.33E-02	7.48E-09
Pu-244	1.71E-05	5.50E-12
Am-241	1.58E+01	5.08E-06
Am-242m	1.86E-01	5.97E-08
Am-243	3.57E-02	1.15E-08
Cm-242	1.14E-02	3.65E-09
Cm-243	2.43E-03	7.81E-10
Cm-244	3.28E-01	1.05E-07
Cm-245	1.43E-04	4.60E-11
Cm-247	2.30E-09	7.39E-16
Cm-248	4.26E-13	1.37E-19
Bk-249	2.94E-15	9.44E-22
Cf-249	2.43E-12	7.80E-19

The inadvertent intruder scenario assumes that an eight-inch well drill intersects and passes through a four-inch waste transfer line and brings the radioactive contents to the surface. A drilling diameter of eight-inches is based on the use of a four-inch diameter pipe for the well, which is a standard pipe size for domestic water use. The well cuttings include material that is equivalent to a blend of piping material, evaporator contents, and material from other ancillary facilities. The radionuclide inventory for this intruder scenario is provided in Table E-4. [6],[8]

Table E-4: Intruder-Agricultural Scenario Radionuclide Inventory from Drill Cuttings

Radionuclide	Curies	Radionuclide	Curies	Radionuclide	Curies
H-3	2.80E−03	Cs-137	3.18E−01	U-238	2.56E−08
C-14	3.69E−08	Ba-137m	3.01E−01	Np-237	7.13E−08
Al-26	3.37E−07	Ce-144	4.77E−06	Pu-238	9.95E−03
Co-60	1.32E−03	Pr-144	4.77E−06	Pu-239	8.86E−05
Ni-59	2.55E−06	Pm-147	2.23E−02	Pu-240	6.20E−05
Ni-63	2.55E−04	Sm-151	6.63E−03	Pu-241	7.21E−03
Se-79	1.83E−06	Eu-152	5.17E−05	Pu-242	2.51E−07
Sr-90	2.54E−01	Eu-154	2.79E−03	Pu-244	4.15E−10
Y-90	2.54E−01	Eu-155	2.54E−03	Am-241	2.18E−04
Nb-94	1.37E−09	Ra-226	2.76E−09	Am-242m	2.89E−07
Tc-99	3.58E−05	Ra-228	0.00E+00	Am-243	8.73E−08
Rh-106	3.39E−05	Ac-227	8.33E−11	Cm-242	2.76E−07
Ru-106	3.39E−05	Th-229	2.38E−04	Cm-243	5.89E−08
Te-125m	3.17E−04	Th-230	3.53E−09	Cm-244	6.37E−06
Sb-125	1.30E−03	Th-232	0.00E+00	Cm-245	2.93E−09
Sb-126	4.76E−07	Pa-231	8.35E−11	Cm-247	9.91E−18
Sb-126m	3.40E−06	U-232	2.35E−08	Cm-248	4.74E−14
Sn-126	3.40E−06	U-233	3.81E−04	Bk-249	7.13E−20
I-129	3.40E−08	U-234	3.38E−08	Cf-249	5.89E−17
Cs-134	9.34E−04	U-235	5.54E−10		
Cs-135	2.12E−08	U-236	1.01E−08		

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APPENDIX F: MANAGEMENT CONTROL SYSTEMS

Appendix Purpose

The purpose of this appendix is to describe management control systems used to ensure quality in the process of determining whether Tank 19 and Tank 18 will comply with Section 3116 of the NDAA when they are closed in place.

Appendix Contents

This appendix identifies applicable DOE and contractor requirements, describes the contractor quality assurance program, explains how quality was assured in data collection and analysis, describes qualifications of the waste determination team, and outlines how documents and records related to the waste determination are controlled. This appendix also describes peer reviews of the computer modeling related to determining compliance with the performance objectives specified in 10 CFR 61, Subpart C.

Key Points

- The process used was consistent with DOE quality standards in 10 CFR 830, Subpart A, and DOE Order 414.1B, *Quality Assurance*.
- The process used was consistent with contractor quality standards included in the Westinghouse Savannah River Company Quality Assurance Manual, the company Conduct of Engineering Manual, and related company implementing procedures.
- The process used in estimating amounts of residual radionuclides in Tanks 19 and 18 included statistical analyses to determine the optimum sampling strategy and analyses of the samples by the Savannah River National Laboratory.
- Implementation of the quality standards and procedures used provides assurance that information used in the determination is accurate and supportable and that conclusions reached are appropriate.
- The determination of compliance with Section 3116 was accomplished by a Westinghouse Savannah River Company team of experienced waste management professionals under the direction and oversight of the DOE-SR.
- Two separate independent peer reviews of the performance assessment modeling described in the PODD provide additional assurance that analysis results used in the determination are reasonable.

The management control systems used in this waste determination have been formalized in a series of related standards and written procedures. They are designed to ensure that primary project objectives are met and that an optimum margin of safety for protection of personnel, the public, and the environment is achieved. Implementation of these systems by DOE and contractor managers and engineers provides assurance that data and engineering analyses which form the basis for determination of compliance with Section 3116 of the NDAA are accurate and supportable, and that the proper conclusions were reached.

REQUIREMENTS AND PROCEDURES

The QA process followed in this waste determination was based on DOE requirements contained in (1) Code of Federal Regulations 10 CFR 830, Subpart A [1], which establishes quality requirements for DOE contractors conducting activities including providing items and services that affect, or may affect, nuclear safety of DOE facilities and (2) DOE Order 414.1B, *Quality Assurance* [2] and its associated manuals.

The DOE-Savannah River Quality Assurance Program Manual [3] (SRM 414.1.1.C) describes the SRS Quality Assurance Program required by DOE Order 414.1B. This program is based primarily on the commercial consensus standard American Society of Mechanical Engineering (ASME) Nuclear Quality Assurance (NQA) NQA-1-2000, *Quality Assurance Requirements for Nuclear Facility Applications* [4].

DOE contractors are required by the SRID process to identify and incorporate the requirements of 10 CFR 830 and DOE Order 414.1B in their company-level procedures and processes. WSRC incorporates these DOE requirements in policy, manuals, and procedures such as the following:

- WSRC-1-01, *Management Policies (MP)*, Policy 4.2, *Quality Assurance* [5]
- WSRC-RP-92-225, *Quality Assurance Management Plan* [6];
- WSRC Manual 1Q, *Quality Assurance Manual* [7];
- WSRC Manual E7, *Conduct of Engineering Manual* [8]; and
- Appropriate implementing procedures.

The next section briefly discusses each of these.

QUALITY ASSURANCE

WSRC Manual 1Q [7] describes general WSRC requirements for QA as follows:

> **WSRC-1-01, Management Policies (MP), Policy 4.2, *Quality Assurance* [5].** This document contains the WSRC President's policy statement regarding the company's commitment to provide products and services that meet or exceed the requirements and expectations of WSRC customers. This policy requires the WSRC QA Program is to be implemented in a manner that supports implementation of WSRC imperatives of safety, disciplined operations, cost-effectiveness, continuous improvement, and teamwork.
>
> WSRC has established and implemented an Integrated Safety Management System (ISMS). The QA program is consistent with, and an integral part, of the WSRC ISMS. The policy requires that the program include appropriate procedures to comply with legal, regulatory, contractual, and corporate requirements related to quality. The policy also requires that the WSRC QA Program comply with DOE Order 414.1B [2], 10 CFR 830, Subpart A [1] and the WSRC Quality Assurance Management Plan [6].

The WSRC QA Program incorporates a graded approach commensurate with risk in the definition and application of QA/Quality Control (QC) requirements. The QA Program also provides for the prevention of errors, as well as the detection and correction of deficient conditions, and incorporates an assessment process for identifying opportunities for continuous improvement.

WSRC-RP-92-225, *Quality Assurance Management Plan*. The WSRC Quality Assurance Management Plan [6] describes the requirements and responsibilities for execution of the WSRC QA Program for implementing DOE Order 414.1B [2] and 10 CFR 830, Subpart A [1]. ASME NQA-1, *Quality Assurance Requirements for Nuclear Facilities* [4] and other consensus standards were used in the development of the QA Management Plan. The plan has been jointly approved by WSRC and DOE-SR and serves as the basis for the establishment of the procedures contained in the WSRC 1Q Manual.

WSRC Manual 1Q, *Quality Assurance Manual*. This manual provides the structure and procedures for achieving and verifying the WSRC requirements for quality. The manual consists of a series of QA procedures that describe applicable QA requirements. The WSRC Manual 1Q states:

> "The WSRC QA Program has been developed to be responsive to the requirements of DOE O 414.1B, Quality Assurance [2] and DOE Safety Rule Title 10 CFR 830, Subpart A, Quality Assurance Requirements [1]. Because of the size and complexity of the Savannah River Site (SRS) and its varied products, services, and missions, the program has been defined in a standard framework of company policy, procedures, and instructions to be used by the implementing organizations to perform quality-related activities. These documents shall, as a minimum, include all of the requirements of WSRC-RP-92-225, *Quality Assurance Management Plan (QAMP)* [6] criteria for which the implementing organizations have responsibility."

WSRC Manual E7, *Conduct of Engineering Manual*. For WSRC, the QA implementing procedure for performing reviews of technical work is found in WSRC Manual E7, *Conduct of Engineering Manual*, Procedure 2.60, *Technical Reviews* [9].

In this procedure, the end use of data drives the level of review required. Design Verification, the highest level review, must be performed for work affecting safety significant/safety class systems. Design Check is the next lower level of review and is required for all production support and general service design output documents. Because engineering work associated with the waste determination, the PODD, and the associated documents is not associated with safety significant or safety class systems, the Design Check represents the appropriate level of rigor. WSRC therefore used the Design Check process in this determination.

Under Procedure 2.60, a design checker assures the technical accuracy of the design document by performing the following Design Check activities:

- A mathematical check, when appropriate;
- A review for correct use of technical input, including quality requirements;
- A review of the approach used and reasonableness of the output; and

- An administrative check (page numbers, etc.).

A design checker must meet the following criteria to perform a Design Check:

- Did not participate in the development of the portion of the document being checked;
- Is knowledgeable in the area of the design or analysis for which they review;
- Is capable of performing similar design or analysis activities; and
- Has security clearance for access to sufficient information to perform the Design Check.

The Savannah River National Laboratory (SRNL) uses a similar procedure for Design Checks and Design Verification [10]. This procedure meets the requirements for review of Type 2 Calculations contained in WSRC Manual E7, Procedure 2.31, *Engineering Calculations* [11]. The SRNL procedure includes a flowchart depicting the SRNL technical review process, lines of inquiry for performing reviews, a checklist for communicating instructions and best management practices to set a benchmark for management expectations.

The next section discusses SRNL QA associated with laboratory analyses.

DATA COLLECTION AND ANALYSIS

WSRC used formal procedures for collecting and analyzing data used in connection with determination of compliance with Section 3116 of the NDAA. These procedures were consistent with QA requirements outlined above.

For example, to provide a technical basis for sampling residual waste inside Tanks 19 and 18, the team had the Statistical Consulting Section of SRNL perform a statistical study. The results were used in an analysis prepared to determine the optimum sampling strategy [12]. This strategy was incorporated into plans for the sampling [13]. These samples were analyzed by the SRNL as described below. The PODD provides details of the sample collection and residual mapping processes.

For laboratory work, SRNL followed its L1 procedures, which establish responsibilities and requirements for the preparation, review, approval, revision, cancellation, and use of division-level procedures and sub-division level procedures and work instructions for the laboratory. The L1 procedures are based on the WSRC company-level procedures and are tailored to the specific activities of the laboratory in an environment that meets the expectations of the WSRC Manual 1Q [7] and WSRC Manual E7 [8].

The laboratory's procedures address matters such as:

- Sample chain of custody;
- Calibration of laboratory instruments using standards traceable to the National Institute of Standards and Technology;
- Appropriate operational checks such as background counts and reproducibility checks;
- Administrative procedures;
- Radiological work practices;

- Control of chemicals, materials, and equipment;
- Storage of archived samples;
- Waste disposition;
- Technical work control; and
- Quality assurance.

PERSONNEL QUALIFICATIONS

Members of the WSRC team dedicated to this waste determination are experienced waste management professionals. They brought to this project extensive knowledge and experience in radioactive waste management and related technical standards such as:

- DOE Order 435.1, Radioactive *Waste Management* [14]
- DOE Manual 435.1-1, *Radioactive Waste Management Manual* [15]
- DOE Guide 435.1-1, *Implementation Guide for Use With DOE M 435.1-1* [16]
- 10 CFR 61, *Licensing Requirements for Land Disposal Of Radioactive Waste* [17]
- NUREG-0782, *Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste* [18]
- NUREG-1573, *A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities: Recommendations of NRC's Performance Assessment Working Group* [19].

Team members included engineers and scientists who are WSRC employees, along with subcontractors, and independent consultants. Biographical information on key team members who participated in evaluating long-term performance of the closed tanks appears in the PODD.

OVERSIGHT AND INDEPENDENT REVIEW

A key element in implementation of the quality standards and procedures outlined above is individuals being responsible for their own work. WSRC management also provides direction and oversight to ensure that these standards and procedures are closely followed, and regularly assesses compliance with quality requirements. In addition, DOE-SR also provides oversight and review to this end.

Evaluation of the long-term performance of the closed tanks compared to the standards of 10 CFR 61, Part C is given particular emphasis because of its importance in ensuring protection of the public and the environment. The PODD describes quality assurance provisions related to engineering design, calculations, and computer modeling – including code validation and modeling quality assurance reviews – that were implemented by WSRC to provide assurance that the analysis results were accurate.

WSRC also arranged for an independent peer review of the PODD by outside experts in this field. WSRC incorporated recommendations of the review team.

DOE-SR arranged for a second independent review of the PODD by another team of independent experts in performance assessment of nuclear facilities. WSRC also incorporated recommendations from this team into the PODD.

These two independent peer reviews provide additional assurance that the conclusion reached by DOE that the long-term performance of the closed tanks will meet the performance objectives of 10 CFR 61, Part C is appropriate.

DOCUMENT AND RECORD CONTROL

Documents and records associated with this waste determination are controlled in accordance with the SRS Records Management System. This system is used to ensure that records important to safety and quality are generated, reviewed, approved, collected, and maintained so they accurately reflect completed work and facility conditions, and comply with applicable statutory or contractual requirements.

The SRS Records Management System incorporates the requirements of DOE Order 200.1, *Information Management Program* [20], and DOE Order 414.1B, *Quality Assurance* [2]. Schedules for records retention and disposition comply with the General Records Schedule of the National Archives and Records Administration and other approved records schedules. The Record Management procedures include instructions for retention, protection, preservation, changes, traceability, accountability, and retrievability of records. They also provide controls to ensure records are legible, accurate, complete, retrievable, and validated by authorized personnel.

Records are stored and maintained to minimize the risk of damage, larceny, vandalism, or deterioration. Active records are not sent to records holding facilities, but are stored in a facility where the records may be readily accessed.

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