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Draft Basis for Section 3116 Determination for Closure of F-Tank Farm at the Savannah River Site

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APPENDICES

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Appendix A Liquid Waste System Description A-1

ACRONYMS / ABREVIATIONS

ADMP	Advanced Design Mixer Pump
AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
ARP	Actinide Removal Process
CFR	Code of Federal Regulations
Ci	curie(s)
CLM	Central Climatology
cm	centimeter(s)
CSRA	Central Savannah River Area
CSSX	Caustic Side Solvent Extraction
DDA	Deliquification, Dissolution, and Adjustment
DOE	United States Department of Energy
DSA	Documented Safety Analysis
DWPF	Defense Waste Processing Facility
ECC	Enhanced Chemical Cleaning
E_h	Measure of reduction (or oxidation) potential
EIS	Environmental Impact Statement
EPA	United States Environmental Protection Agency
ETF	Effluent Treatment Facility
FDB	F-Tank Farm Diversion Box
FFA	Federal Facility Agreement
FPP	F-Tank Farm Pump Pit
FPT	F-Tank Farm Pump Tank
FTF	F-Tank Farm
FTF PA	F-Tank Farm Performance Assessment

ACRONYMS / ABREVIATIONS (Continued)

g	gram(s)
GCL	Geosynthetic Clay Liner
GCP	General Closure Plan
gpm	gallons per minute
GSA	General Separations Area
GSAD	General Separations Area Database
GWSB	Glass Waste Storage Building
HA	Hazard Analysis
HAW	High Activity Waste
HDPE	High Density Polyethylene
HEU	Highly Enriched Uranium
HM	H-Modified
hr	hour(s)
HRR	Highly Radioactive Radionuclide
HTF	H-Tank Farm
lb	pound(s)
ISWLF	Industrial Solid Waste Landfill Facility
K_d	Distribution Coefficient
LAW	Low Activity Waste
LDB	Leak Detection Box
m	meter(s)
MCi	million curie(s)
MCU	Modular Caustic Side Solvent Extraction Unit
MEP	Maximum Extent Practical
mg	milligram(s)
Mgal	million gallon(s)
mrem	millirem(s)
MSL	Mean Sea Level
mSv	milliSievert(s)
NASA	National Aeronautics and Space Administration
NCRP	National Council on Radiation Protection and Measurements
NDAA	Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005
NPDES	National Pollutant Discharge Elimination System
NRC	United States Nuclear Regulatory Commission
NRMP	Natural Resources Management Plan
OSHA	Occupational Safety and Health Administration
OUO	Official Use Only
PA	Performance Assessment
PEL	Permissible Exposure Limit
pH	Measure of acidity or alkalinity of a solution
psi	pounds per square inch
PUREX	Plutonium Uranium Extraction
RBA	Radiological Buffer Area
ROD	Record of Decision
rpm	revolutions per minute
RPP	Radiation Protection Program

ACRONYMS / ABREVIATIONS (Continued)

ROI	Region of Influence
SA	Special Analysis
SCDHEC	South Carolina Department of Health and Environmental Control
SDF	Saltstone Disposal Facility
SMA	Strong Motion Accelerometer
SMP	Submersible Mixer Pump
SPF	Saltstone Production Facility
S/RID	Standards/Requirement Identification Document
SREL	Savannah River Ecology Laboratory
SRNL	Savannah River National Laboratory
SRS	Savannah River Site
STP	Submersible Transfer Pump
SWPF	Salt Waste Processing Facility
TED	Total Effective Dose
TEDE	Total Effective Dose Equivalent
TNX	Training and Experimental Test Facility
UTR	Upper Three Runs
WAC	Waste Acceptance Criteria
WCS	Waste Characterization System
WMC	Waste Mixing Chamber
yr	year(s)

1.0 INTRODUCTION AND PURPOSE

Section Purpose

This section provides the purpose and scope of this document, titled *Draft Basis for Section 3116 Determination for Closure of F-Tank Farm at the Savannah River Site* (hereinafter referred to as: Draft FTF 3116 Basis Document).

Section Contents

This section contains a brief introduction to the Savannah River Site (SRS) and F-Tank Farm (FTF) and describes the purpose and scope of this Draft FTF 3116 Basis Document.

Key Points

- The Department of Energy (DOE) is issuing this Draft FTF 3116 Basis Document to provide a draft basis for a potential determination by the Secretary of Energy, in consultation with the Nuclear Regulatory Commission (NRC), pursuant to Section 3116 of the Ronald W. Reagan National Defense Authorization Act (NDAA) for Fiscal Year 2005 (hereinafter referred to as: NDAA Section 3116). [NDAA_3116]
- This Draft FTF 3116 Basis Document applies to stabilized residuals in waste tanks[1] and ancillary structures, those waste tanks, and the ancillary structures (including integral equipment) at the SRS FTF at the time of closure.
- The FTF is a 22-acre site consisting of underground radioactive waste storage tanks and supporting ancillary structures.
- The FTF tank waste storage and removal operations are performed in accordance with a State-issued industrial wastewater construction permit. Removal from service and stabilization of the FTF waste tanks and ancillary structures will be carried out pursuant to a State-approved closure plan and will be consistent with the SRS Federal Facility Agreement (FFA). [WSRC-OS-94-42]
- After completion of waste removal activities, the FTF waste tanks will be stabilized by filling the tanks with grout. Ancillary structures will be filled, as necessary, to prevent subsidence.
- Stabilization of individual FTF waste tanks and ancillary structures is anticipated to take place after individual component cleaning is complete.
- The Final FTF 3116 Basis Document will be issued by DOE following consultation with the NRC.

1.1 Introduction

In accordance with NDAA Section 3116, certain waste from reprocessing of spent nuclear fuel is not high-level waste if the Secretary of Energy, in consultation with the NRC, determines that the criteria in NDAA Section 3116(a) are met. This Draft FTF 3116 Basis Document shows that those criteria are satisfied, to support a potential determination by the Secretary pursuant Section 3116. This Draft FTF 3116 Basis Document concerns the stabilized residuals in waste tanks and ancillary structures, those waste tanks, and the ancillary structures (including integral equipment) at the SRS FTF at the time of closure.

The DOE is issuing this Draft FTF 3116 Basis Document for NRC consultative review, as part of DOE's consultation with NRC. Although not required by NDAA Section 3116, DOE will issue this Draft FTF 3116 Basis Document for public review and comment. This Draft FTF 3116 Basis Document will be finalized after DOE consults with the NRC and considers public comments.

[1] The FTF has 22 waste tanks. Two of those waste tanks, Tanks 17 and 20, were cleaned and operationally closed in 1997, prior to enactment of NDAA Section 3116. Accordingly, Tanks 17 and 20 are not within the scope of this draft FTF 3116 Basis Document. Rather, the remaining 20 tanks in FTF (and their residuals) are addressed by this Draft FTF 3116 Basis Document, and, unless otherwise specified, references in this document to "the waste tanks" or "FTF waste tanks" refer to those remaining tanks. Nevertheless, the experience gained from Tanks 17 and 20 is discussed in certain sections of this document to provide insight and additional information. Also, the *Performance Assessment for the F-Tank Farm at the Savannah River Site* [SRS-REG-2007-00002], cited in several sections of this Draft FTF 3116 Basis Document, takes into account radionuclide residuals from all FTF waste tanks, including Tanks 17 and 20, for accuracy and completeness.

1.2 Purpose and Scope

The purpose of this Draft FTF 3116 Basis Document is to demonstrate and document that, after final stabilization activities are complete, the stabilized residuals in the FTF waste tanks and ancillary structures, the waste tanks, and the ancillary structures (including integral equipment) at the time of closure meet the NDAA Section 3116(a) criteria and, therefore are not high-level waste.

The scope of this Draft FTF 3116 Basis Document specifically addresses the stabilized residuals in the FTF waste tanks and ancillary structures, the waste tanks, and the ancillary structures (including integral equipment) at the time of FTF closure[2]. This Draft FTF 3116 Basis Document does not include other SRS facilities or systems, or waste removed from the waste tanks and ancillary structures.

1.3 Schedule and Plans for Closing Tanks

The FTF waste tanks[3] are closed in accordance with the SRS FFA, a formal agreement between DOE, Region 4 of the United States Environmental Protection Agency (EPA) and the South Carolina Department of Health and Environmental Control (SCDHEC). The FFA establishes that, among other things, the SRS waste tanks that do not meet secondary containment standards (old style tanks, specifically the Type I and Type IV tanks in FTF) must be removed from service according to the FFA schedule. The current FFA calls for operational closure of Tanks 18 and 19 by December 2012, and staggered operational closure of the other eight FTF (Type I) waste tanks (tanks numbers not specified in the SRS FFA) by September 2022[4]. [WSRC-OS-94-42] DOE addresses the closure of the remaining FTF tanks (Type III and IIIA tanks) and ancillary structures in the SRS Liquid Waste System Plan[5]. [SRR-LWP-2009-00001]

The DOE will, pursuant to its authority, including that under the Atomic Energy Act of 1954, as amended, and applicable DOE Orders, manuals and policies, pursue closure of the FTF and monitor its activities to ensure compliance with all requirements. Furthermore, DOE uses a documented process to review and resolve any disposal questions and develop any mitigation measures, as appropriate.

Tank waste storage and removal operations are governed by an SCDHEC industrial wastewater construction permit. [DHEC_03-03-1993] Removal from service and stabilization of the FTF waste tanks and ancillary structures will be carried out pursuant to a State-approved closure plan, the FTF General Closure Plan, which contains the overall plan for removing from service and stabilizing the FTF waste tanks and ancillary structures. [LWO-RIP-2009-00009] A specific Closure Module for each tank or ancillary structure or groupings of tanks and ancillary structures will be developed and submitted to the State of South Carolina for approval. Final tank stabilization activities shall not proceed until the State grants approval. Stabilization of individual FTF waste tanks and ancillary structures is anticipated to take place after individual component cleaning is complete.

In the *Savannah River Site High-Level Waste Tank Closure Environmental Impact Statement Record of Decision*, DOE selected the alternative to fill the waste tanks with reducing grout to stabilize the residual

[2] For the purpose of this Draft FTF 3116 Basis Document, the residual waste remaining in a waste tank or ancillary structure following successful completion of waste removal activities and removal of the highly radioactive radionuclides to the maximum extent practical is referred to as "residuals." Stabilization of these residuals within the FTF waste tanks will be carried out by filling the tanks with grout after completion of waste removal activities. Ancillary structures will be filled, as necessary, to prevent subsidence of the structure or final closure cap. The DOE does not plan to add fill material to the FTF transfer lines.

[3] The FTF contains four designs (types) of tanks, called Type I, Type IV, Type III and Type IIIA tanks. The FTF tanks are also numbered, although not sequentially. The tank types are discussed in further detail in Section 2.0 of this Draft FTF 3116 Basis Document.

[4] The FFA includes procedures for revision of the FFA schedule.

[5] Under the current SRS Liquid Waste System Plan, DOE anticipates closure of the remaining FTF tanks (Type III and IIIA tanks) by 2029. The Liquid Waste System Plan is updated periodically as appropriate.

According to the *Performance Assessment for the F-Tank Farm at the Savannah River Site* [SRS-REG-2007-00002], DOE's planned sequence for closure of FTF is as follows:

- Closure of Type IV tanks, the Type I tanks and finally the Type III and IIIA tanks. The ancillary structures (such as transfer lines) are planned to be closed as appropriate with a goal of closing FTF in stages.
- Following closure of a geographic section (such as Type IV tanks and evaporator area), the section may be left in an interim closure state in preparation for final closure.
- Following closure of all FTF waste tanks and ancillary structures, FTF will undergo final closure in accordance with the FFA.

material. This method was chosen as the most preferred environmental alternative and the least hazardous for closure of the waste tanks and associated equipment. [DOE/EIS-0303 ROD] The stabilized grout form will provide a chemical environment to reduce migration of contaminants into the environment, prevent inadvertent intrusion and minimize free-standing liquids and void spaces in the waste tanks.

1.4 Outline of Draft FTF 3116 Basis Document

To support closure of the FTF at the SRS, this Draft FTF 3116 Basis Document demonstrates that the stabilized residuals within the FTF waste tanks and ancillary structures, those waste tanks, and the ancillary structures (including integral equipment) at the time of closure resulting from, in part, prior reprocessing of spent nuclear fuel meet the criteria in NDAA Section 3116(a) and thus are not high-level waste.

Section 2.0 of this Draft FTF 3116 Basis Document provides an overview of SRS and the FTF. In addition, extensive descriptions of the FTF and waste processing facilities are provided in the *Performance Assessment for the F-Tank Farm at the Savannah River Site* (hereinafter referred to as: FTF PA). [SRS-REG-2007-00002] Section 3.0 of this Draft FTF 3116 Basis Document provides the specific language and criteria of NDAA Section 3116(a). Subsequently, Section 4.0 through Section 8.0 provides the basis for the Secretary of Energy, in consultation with the NRC, to make a determination that the NDAA Section 3116(a) criteria are met and, thus, the waste is not high-level waste.

2.0 BACKGROUND

Section Purpose

The purpose of this section is to provide background information to support discussions in later sections which demonstrate that the provisions in NDAA Section 3116(a) are met.

Section Contents

Section 2.1 provides an overview of FTF with descriptions of the different waste tank designs and ancillary structures. Section 2.2 identifies the sources of the waste managed in FTF and summarizes the history of each of the waste tanks. Section 2.3 describes waste tank closure activities and status. Section 2.4 describes the residual characterization process. Section 2.5 discusses stabilization of the waste tanks. Section 2.6 describes the FTF Closure Cap.

Key Points

- The FTF occupies 22 acres in the General Separations Area (GSA) near the center of the SRS.
- The FTF contains 22 carbon steel waste tanks of three different basic designs, eight with a nominal capacity of 750,000 gallons per tank and fourteen with a nominal capacity of 1,300,000 gallons per tank.
- Most of the waste in these waste tanks originated in the SRS F-Canyon Facility, which recovered nuclear material produced in the site's nuclear production reactors.
- Two of the FTF waste tanks, Tank 17 and Tank 20, have been operationally closed in place with the approval of the South Carolina Department of Health and Environmental Control prior to enactment of NDAA Section 3116, and are not within the scope of this Draft FTF 3116 Basis Document.
- In addition to the waste tanks, FTF contains ancillary structures with a residual radiological inventory that is accounted for as part of FTF closure.
- Estimated radionuclide concentrations for residual material in FTF at closure were determined by sample analysis, process knowledge data maintained in the Waste Characterization System (WCS), and special analysis; the associated risks were assessed in the FTF PA.
- After waste removal, FTF waste tanks will be filled with grout to provide long-term stability and minimize the mobility and migration of radionuclides.

2.1 Savannah River Site and F-Tank Farm Facility Overview

This section provides brief descriptions of the site and the FTF [6].

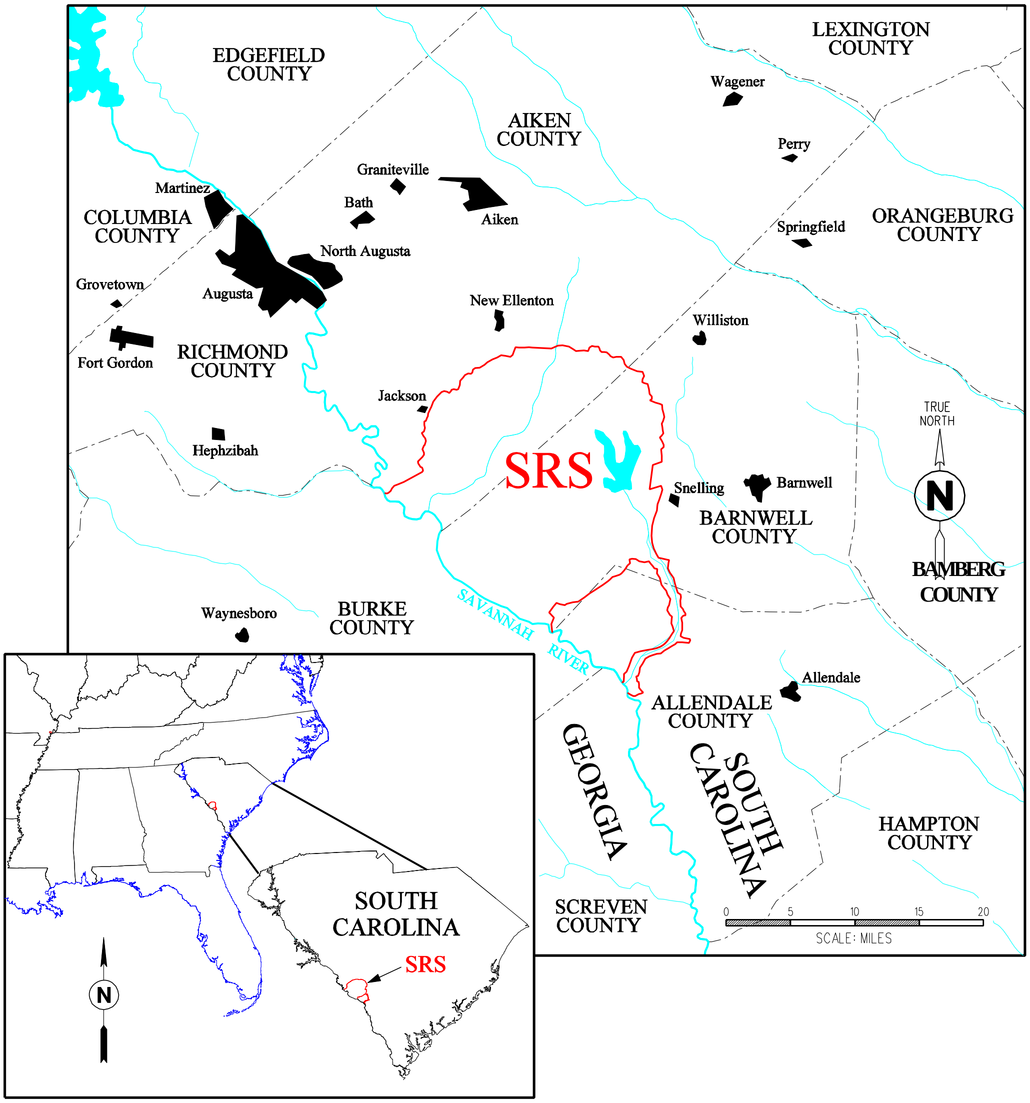
2.1.1 Geography and Demography

2.1.1.1 SRS Site Description

The SRS, one of the sites in the DOE complex, was constructed starting in the early 1950s to produce nuclear materials (such as Pu-239 and tritium). The site covers approximately 310 square miles in South Carolina and borders the Savannah River. The SRS encompasses 198,344 acres in Aiken, Allendale, and Barnwell counties of South Carolina. The site is approximately 12 miles south of Aiken, South Carolina, and 15 miles southeast of Augusta, Georgia, as shown in Figure 2.1-1. [WSRC-STI-2008-00057]

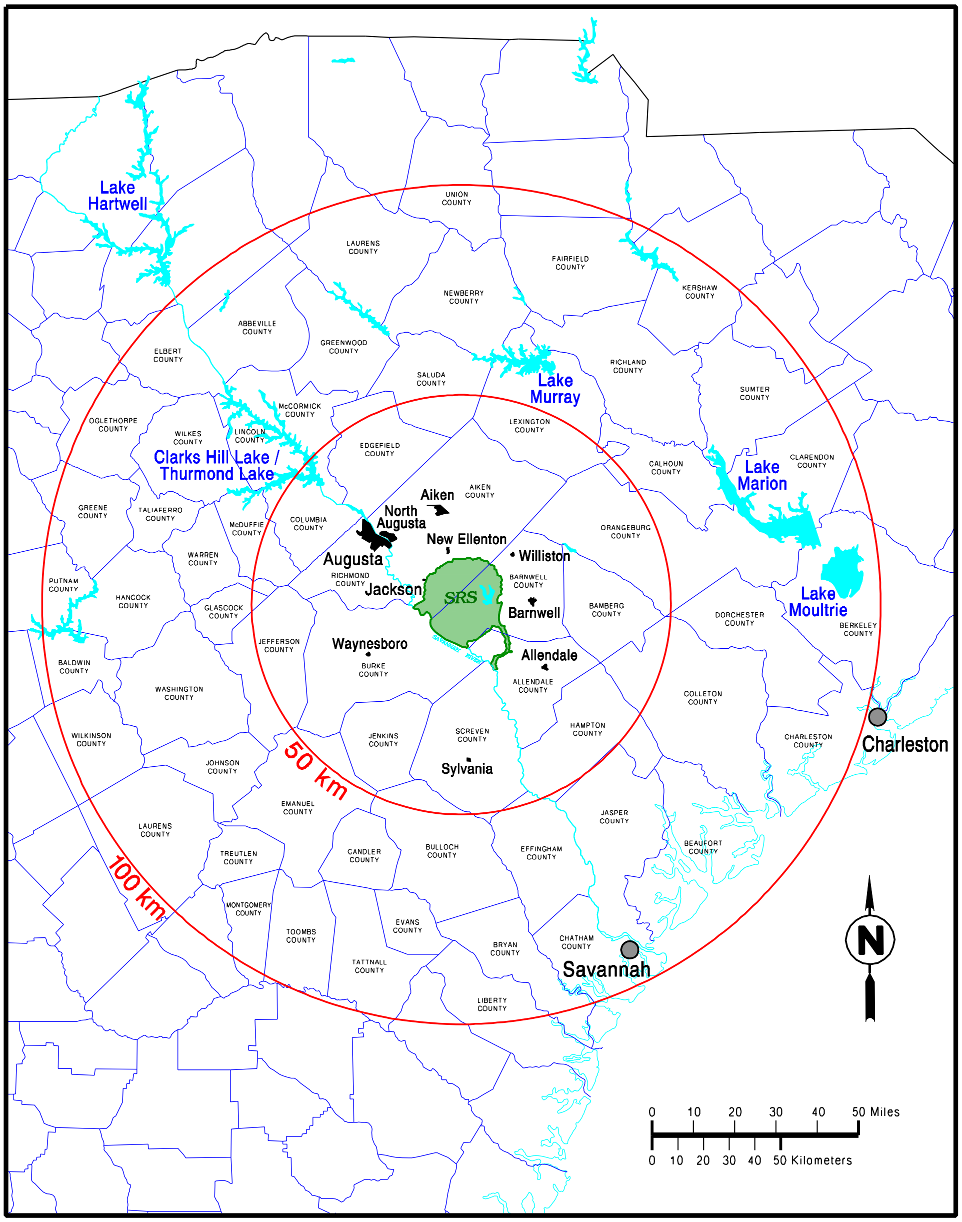
[6] Sections 1.0 and 2.0, as well as Appendix A, of this Draft FTF 3116 Basis Document, contain information to further inform the reader. DOE views such information, to the extent it is not otherwise relied upon in this Draft FTF 3116 Basis Document, as outside the scope of this Draft FTF 3116 Basis Document and not included as NDAA 3116 requirements or criteria for purposes of NRC consultation.

Figure 2.1-1: Physical Location of Savannah River Site



Prominent geographic features within 30 miles of SRS include the Savannah River and Clarks Hill Lake (also known as Thurmond Lake), shown in Figure 2.1-2. The Savannah River forms the southwest boundary of SRS. Clarks Hill Lake is the largest nearby public recreational area. This reservoir is located on the Savannah River, approximately 40 miles upstream of the center of SRS.

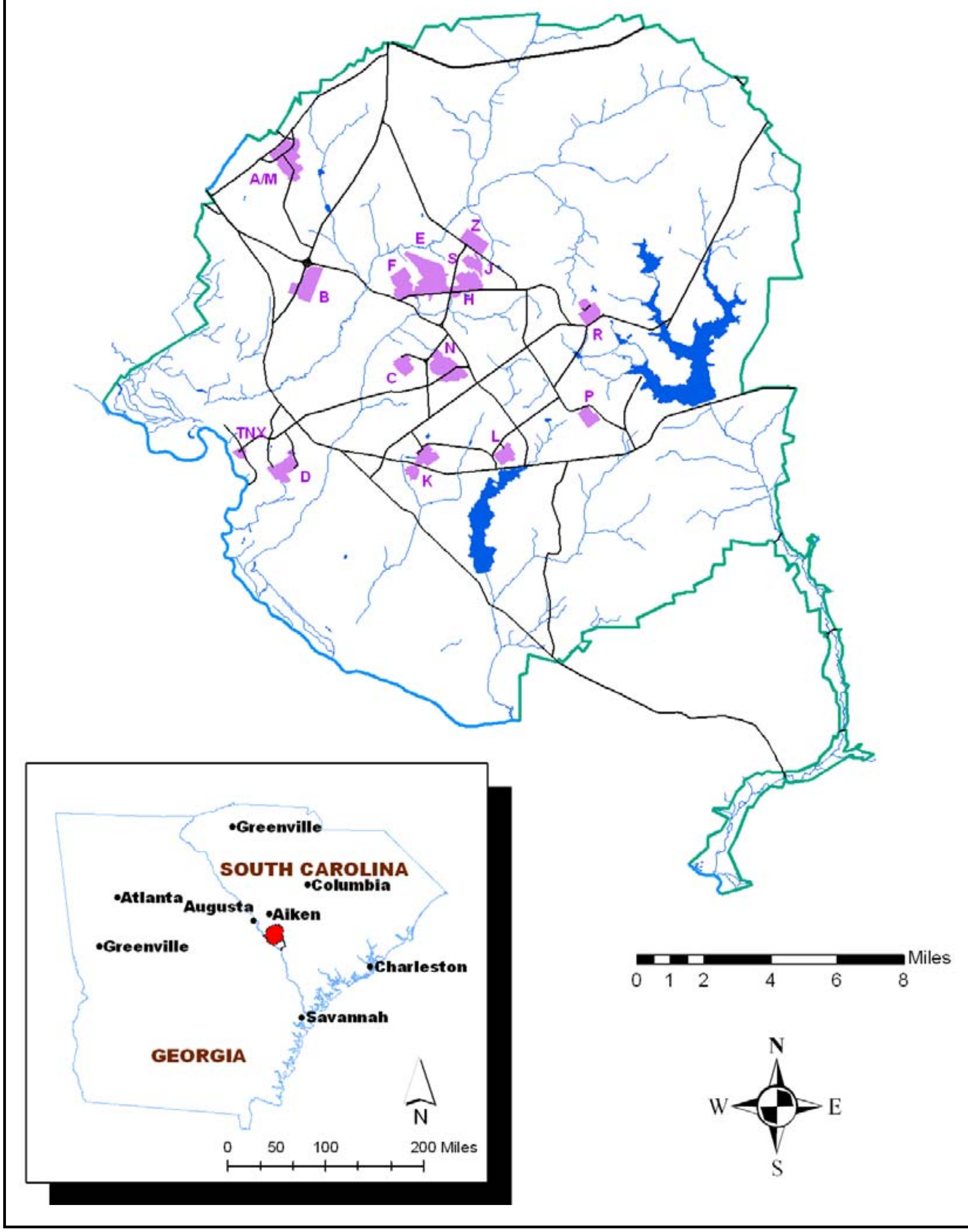
Figure 2.1-2: Location of Savannah River Site and Adjacent Areas



Within the SRS boundary, prominent water features include Par Pond and L Lake, shown in Figure 2.1-3. Par Pond, a former reactor cooling water impoundment, covers approximately 2,700 acres and lies in the eastern sector of SRS. L Lake, another former reactor cooling water impoundment, covers approximately 1,000 acres and lies in the southern sector of SRS. [WSRC-IM-2004-00008, pg 1.4-11]

Figure 2.1-3 also shows the major operational areas at SRS. Prominent operational areas, both past and present, include, Separations (F and H Areas), Waste Management Operations (E Area), Liquid Waste (F, H, J, S and Z Areas) and the Reactor Areas (C, K, L, P, R). Savannah River National Laboratory (SRNL) and Savannah River Ecology Laboratory (SREL) are located in A Area. Administrative and support services are located in B Area and construction administration activities are located in N Area. D Area is the coal-fired powerhouse that provides steam to SRS. M Area and Training and Experimental Test Facility (TNX) have undergone Deactivation and Decommissioning.

Figure 2.1-3: SRS Operational Area Location Map



2.1.1.2 Closure Site Description

The FTF is in F Area which is located in the central region of SRS. Figure 2.1-4 presents the area known as the General Separations Area (GSA). The GSA is located atop a ridge running southwest-northeast that forms the drainage divide between Upper Three Runs (UTR) to the north and Fourmile Branch to the south. The GSA contains the F and H Area Separations Facilities, the S-Area Defense Waste Processing Facility (DWPF), the Z-Area Saltstone Facility and the E-Area Low-Level Waste Disposal Facilities. The FTF is a liquid waste storage facility consisting of 22 carbon steel waste tanks, shown in Figure 2.1-5, which store liquid radioactive waste generated primarily from the F-Canyon PUREX process. Note that two tanks, 17 and 20, have been cleaned, removed from service and filled with grout. The FTF design features (e.g., waste tanks, transfer lines, evaporator systems) are discussed in more detail in Sections 2.1.11 and 2.1.12.

2.1.1.3 Population Distribution

According to U.S. Census Bureau data, the estimated 2008 population in the eight-county region of influence (ROI) was 550,675. Four of the counties lie in South Carolina and include: Aiken, Allendale, Bamberg and Barnwell. The other four counties lie in Georgia and include Burke, Columbia, Richmond and Screven (Figure 2.1-2). The ROI includes the counties immediately adjacent to SRS and the counties where the majority of SRS workers reside. Approximately 84% of the ROI population live in the following three counties: Aiken (28%), Richmond (36.2%) and Columbia (20.1%). Only about 16% of the ROI population live in the remaining counties as shown in Table 2.1-1. [*http://factfinder.census.gov*]

Figure 2.1-4: Layout of the GSA

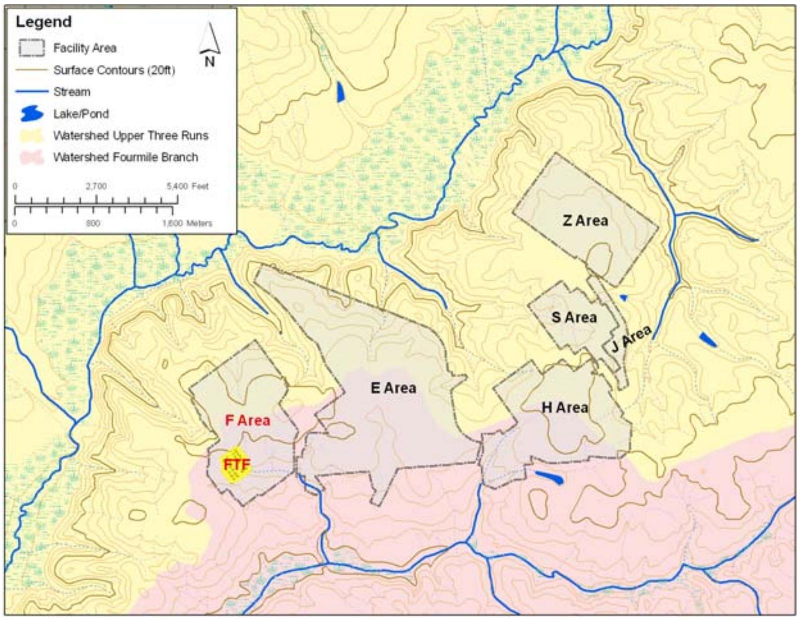
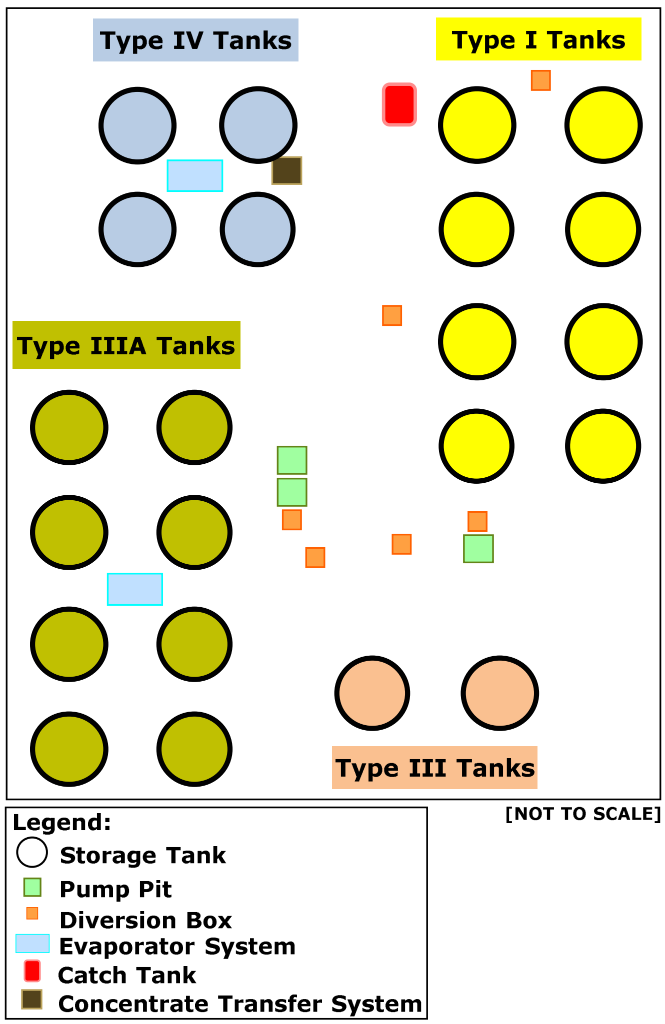


Figure 2.1-5: General Layout of FTF



From 2000 to 2008 the population in the eight-county region grew an estimated 5.8%. Columbia County had the highest estimated growth at approximately 23.9% followed by Aiken County with an estimated growth of approximately 8.1% and Burke County with an estimated growth of 2.2%. Allendale, Bamberg, Barnwell and Screven counties experienced a net population loss. Calculations are based on information obtained from the U. S. Census Bureau website: [*http://factfinder.census.gov*]

Table 2.1-1: Population Distribution and Percent of Region of Influence (% ROI) for Counties and Selected Communities

Jurisdiction	2008 Population Estimate[1]	2008 % ROI
SOUTH CAROLINA		
Aiken County	154,071	28
Aiken, City	29,434	5.4
Jackson, Town	1,647	0.3
New Ellenton, Town	2,227	0.4
North Augusta, City	20,712	3.8
Allendale County	10,447	1.9
Allendale, Town	3,659	0.7
Bamberg County	15,307	2.8
Bamberg, Town	3,432	0.6
Barnwell County	22,872	4.2
Barnwell, City	4,783	0.9
GEORGIA		
Burke County	22,732	4.1
Columbia County	110,627	20.1
Richmond County	199,486	36.2
Screven County	15,133	2.7
Eight-County Total	**550,675**	

[1] 2008 Population estimates based on 2000 population census and are provided by the U.S. Census Bureau, Population Estimates Program; data for births, deaths, and domestic and international migration were used by the U.S. Census Bureau to update the 2000 base counts. [*http://factfinder.census.gov*]

Population projections and further information regarding the region around SRS can be found in the *High-Level Waste Tank Closure Final Environmental Impact Statement*. [DOE/EIS-0303]

2.1.1.4 Land Use – Present and Planned

Land within a five-mile radius of FTF is entirely within SRS boundaries and is currently used either for industrial purposes or as forested land. Current land use within the entire GSA is classified as heavy nuclear industrial. Plans for the future of SRS are addressed in two key planning documents identified below.

- The *SRS End State Vision* [PIT-MISC-0089]
- The *Savannah River Site (SRS) Long Range Comprehensive Plan* [PIT-MISC-0041]

The Long Range Comprehensive Plan assumes that the entire site will be owned and controlled by the Federal Government in perpetuity[7].

2.1.2 Meteorology and Climatology

2.1.2.1 General SRS Climate

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

Average rainfall during the fall is usually the least of the four seasons. In the winter months, mid-latitude low pressure systems and associated fronts often migrate through the region. As a result, conditions frequently alternate between warm, moist, subtropical air from the Gulf of Mexico region and cool, dry polar air. The Appalachian Mountains to the north and northwest of SRS help moderate the extremely cold temperatures that are associated with occasional outbreaks of Arctic air. Consequently, less than one-third of winter days have minimum temperatures below freezing on average, and days with temperatures below 20°F are infrequent. Measurable snowfall occurs an average of once every two years. Tornadoes occur more frequently in spring than the other seasons of the year. Although spring

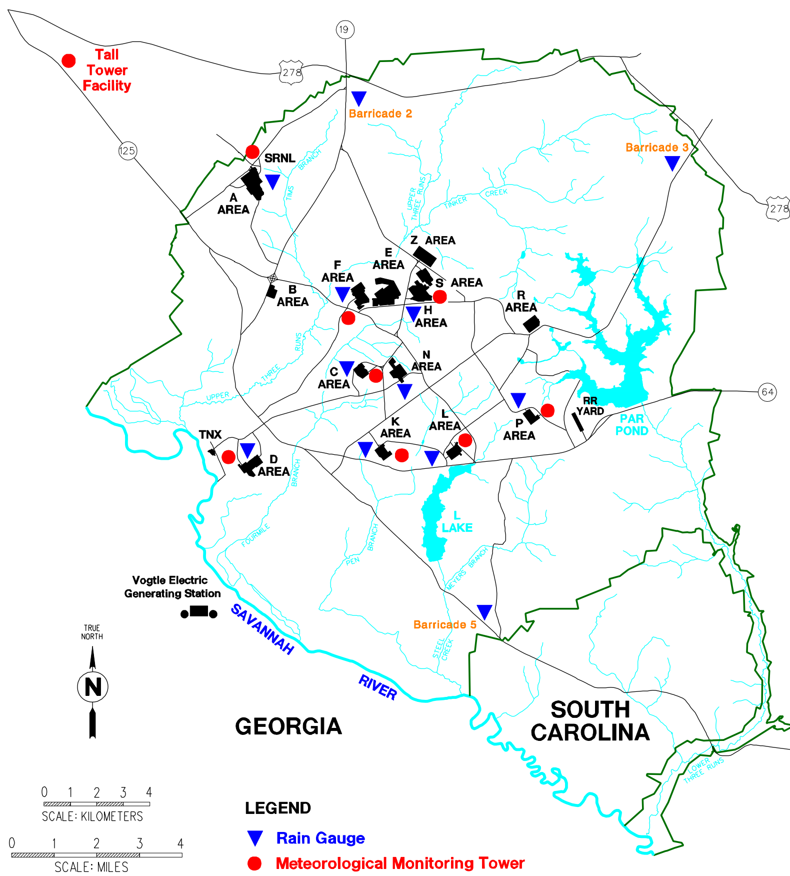
[7] For the purposes of the FTF PA, no federal protection is assumed beyond a 100-year period of institutional control. The 100-year period of institutional control is assumed to begin in the year 2020.

weather is somewhat windy, temperatures are usually mild and humidity is relatively low. [WSRC-TR-2007-00118]

2.1.2.2 Meteorological Data Collection

Meteorological data are collected at SRS from a network of nine primary monitoring stations (Figure 2.1-6). Towers located adjacent to each of eight areas (A, C, D, F, H, K, L and P Areas) are equipped to measure wind direction and wind speed at 201.3 feet above ground and to measure temperature and dew point at both 6.6 feet (2 meters) and 201.3 feet above ground. Temperature and dew point are also measured at 2 meters. A ninth tower near N Area, known as the Central Climatology (CLM) Site, is instrumented with wind, temperature and dew point sensors at four levels: 6.6 feet (13.2 feet for wind), 59.4 feet, 118.8 feet and 201.3 feet. The CLM site is also equipped with an automated tipping bucket rain gauge, a barometric pressure sensor and a solar radiometer near the tower at ground level. Data acquisition units at each station record a measurement from each instrument at 1-second intervals. Every 15 minutes, 900 data points are processed to generate statistical summaries for each variable, including averages and instantaneous maxima. The results are then uploaded to a relational database for permanent archival.

Figure 2.1-6: SRS Meteorological Monitoring Network



[WSRC-TR-2007-00118]

In addition, the Tall Tower facility near Beech Island, South Carolina, provides a set of high-quality meteorological measurements that is unique to the southeast United States. This facility utilizes fast-response sonic anemometers, water vapor sensors, barometric pressure sensors, slow-response temperature sensors and relative humidity sensors. Data are collected at 100 feet, 200 feet, and 1,000 feet above ground level. Spread-spectrum modems at each measurement level transmit raw data to a redundant set of personal computers at the SRNL. Data processing software on the personal computers determine mean values and other statistical quantities every 15 minutes and uploads the results to the relational database.

Precipitation measurements are collected from a network of 13 rain gauges across SRS (Figure 2.1-6). Twelve of these gauges are read manually by security or operations personnel once per day, usually around 6 am. The daily data are reported to the SRNL Atmospheric Technologies Center, where it is technically reviewed and manually entered into a permanent electronic database. The other is an automated rain gauge at the CLM previously addressed above. [WSRC-TR-2007-00118]

2.1.2.3 Data Pertinent to FTF PA Modeling

Weather data pertinent to the FTF PA modeling are atmospheric dispersion, precipitation, and air temperature. Each is discussed below.

2.1.2.3.1 Atmospheric Dispersion

Since the mid-1970s, a five-year database of meteorological conditions at SRS is updated in order to support dose calculations for accident or routine release scenarios for onsite and offsite populations. The meteorological database includes wind speed, wind direction, temperature, dew point and horizontal and vertical turbulence intensities. The most recent database is for the time period January 1, 2002 through December 31, 2006, and consists of one-hour time averages of temperature and dew-point; wind speed, direction, and turbulence. [WSRC-STI-2007-00613] This data is used to determine dose release factors in the evaluation for air pathway dose modeling described in Section 4.0 of the FTF PA, and reported in WSRC-STI-2007-00343. [SRS-REG-2007-00002]

2.1.2.3.2 Precipitation

Compilations of rainfall data obtained from meteorological data collection described above for years 1952 through 2006 for the site and for years 1961 through 2006 obtained from the 200-F weather station are provided in WSRC-STI-2007-00184. An average precipitation of 48.5 inches per year results from the 55 year monitoring period for the site and 49 inches per year from the 46 year monitoring period for F Area. This data is used to determine appropriate rainfall assumptions for the performance evaluation of infiltration through the closure cap described in Section 2.6 and evaluated in WSRC-STI-2007-00184.

2.1.2.3.3 Air Temperature

A compilation of air temperature data obtained from meteorological data collection described above for years 1968 through 2005 is provided in WSRC-STI-2007-00184. For this 37-year period the annual average air temperature is approximately 64°F with an average monthly air temperature from a low of approximately 46°F to a high of approximately 81°F. This data is used to determine appropriate assumptions for the performance evaluation of infiltration through the closure cap described in Section 2.6 and evaluated in WSRC-STI-2007-00184.

2.1.3 Ecology

Comprehensive descriptions of SRS ecological resources and wildlife can be found in *SRS Ecology Environmental Information Document* and are briefly discussed in this section. [WSRC-TR-2005-00201]

The SRS supports abundant terrestrial and semi-aquatic wildlife, as well as a number of species considered threatened or endangered. Since the early 1950s, the SRS has changed from 67% forest and 33% agriculture to 94% forest, with the remainder in aquatic habitats and developed areas. Wildlife populations correspondingly shifted from forest-farm edge-utilizing species to a predominance of forest-dwelling species. The SRS now supports 44 species of amphibians, 60 species of reptiles, 255 species of birds, and 55 species of mammals. These populations include urban wildlife, several commercially and recreationally important species, and a few threatened and endangered species. Protection and restoration of all flora and fauna to a point where their existence is not jeopardized are principal goals of federal and state environmental programs. Those species of plants and animals afforded governmental protection are collectively referred to as “species of concern.” [WSRC-TR-2005-00201, page 3-1]

The SRS has extensive, widely distributed wetlands, most of which are associated with floodplains, creeks, or impoundments. In addition, approximately 200 Carolina bays occur on SRS. Carolina bays are unique wetland features of the southeastern United States. They are isolated wetland habitats dispersed throughout the uplands of SRS. The approximately 200 Carolina bays on SRS exhibit extremely variable hydrogeology and a range of plant communities from herbaceous marsh to forested wetland. [DOE-EIS-0303, pg 3-26]

The Savannah River bounds SRS to the southwest for approximately 20 miles. The river floodplain supports an extensive swamp, covering approximately 15 square miles of SRS; with a natural levee separating the swamp from the river. Timber was cut in the swamp from the turn of the century until 1951, when the Atomic Energy Commission assumed control of the area. At present, the swamp forest is comprised of two kinds of forested wetland communities. Areas that are slightly elevated and well-drained are characterized by a mixture of oak species, as well as red maple, sweetgum and other hardwood species. Low-lying areas that are continuously flooded are dominated by second-growth bald cypress and water tupelo. [DOE-EIS-0303, page 3-26, 3-29]

The SRS supports abundant herpetofauna because of its temperate climate and diverse habitats. The species of herpetofauna include 17 salamanders, 27 frogs and toads, one crocodilian, 13 turtles, nine lizards and 36 snakes. The class Amphibia is represented on site by two orders, 11 families, 16 genera and 44 species. The Reptilia are represented by three orders, 12 families, 41 genera and 59 species. [WSRC-TR-2005-00201, page 3-2]

Waterfowl and wading birds, as well as many upland species, use SRS aquatic habitats year round. Sixty-seven percent use Carolina bays and emergent marshes. Sixty-eight percent of the upland species use this habitat type. Edge or shoreline areas accounted for high numbers of upland birds at Carolina bays and emergent marshes, stream, and small drainage corridors and river swamp habitats. The aquatic birds are most common in large and small open water habitat. [WSRC-TR-2005-00201, pages 3-10]

More than 255 species of birds are found in the SRS. Large mammals inhabiting the site include white-tailed deer and feral hogs. Raccoon, beaver and otter are relatively common throughout the wetlands of SRS. In addition, the gray fox, opossum, bobcat, gray squirrel, fox squirrel, eastern cottontail, mourning dove, northern bobwhite and eastern wild turkey are common at SRS. Threatened and endangered plant and animal species known to occur or that might occur on the overall SRS include the smooth purple coneflower, wood stork, red-cockaded woodpecker and shortnose sturgeon.

The FTF is located within a densely developed, industrialized area of SRS. The immediate area provides habitat for only those animal species typically classified as urban wildlife. Species commonly encountered in this type of urban landscape include the Southern toad, green anole, rat snake, rock dove, European starling, house mouse, opossum and feral cats and dogs. Grasses and landscaped areas within F Area also provide some marginal terrestrial wildlife habitat. A number of ground-foraging bird species (e.g., American robin, killdeer and mourning dove) and small mammals (e.g., cotton mouse, cotton rat and Eastern cottontail) that use lawns and landscaped areas around buildings may be present at certain times of the year, depending on the level of human activity (e.g., frequency of mowing). Pine plantations managed for timber production by the U.S. Forest Service (under an interagency agreement with DOE) occupy surrounding areas.

The Fourmile Branch seepline area is located in a bottomland hardwood forest community. The canopy layer of this bottomland forest is dominated by sweetgum, red maple and red bay. Sweet bay is also common. The understory consists largely of saplings of these same species, as well as a herbaceous layer of smilax, dog hobble, giant cane, poison ivy, chain fern and hepatica. At the seepline's upland edge, scattered American holly and white oak occur. Dominant along Fourmile Branch in this area are tag alder, willow, sweetgum and wax myrtle. The UTR seepline is located in a similar bottomland hardwood forest community. [DOE-EIS-0303, page 3-30]

No endangered or threatened fish or wildlife species have been recorded near the UTR and Fourmile Branch seeplines. The seeplines and associated bottomland community do not provide habitat favored by endangered or threatened fish and wildlife species known to occur at SRS. The American alligator is the only Federally protected species that could potentially occur in the area of the seeplines. Fourmile Branch does support a small population of American alligator in its lower reaches, where the stream enters the Savannah River swamp. [DOE-EIS-0303, pg 3-30]

According to summaries of studies on UTR documented in the *SRS Ecology Environmental Information Document*, the macroinvertebrate communities of UTR drainage are unusual. [WSRC-TR-2005-00201] They include many rare species and contain species not often found living together in the same freshwater system. Since UTR is a spring-fed stream and is colder and generally clearer than most surface water at its low elevation, species typical of unpolluted streams in northern North America or the Southern Appalachian Mountains are found here along with lowland (Atlantic Coastal Plain) species.

The fish community of UTR is typical of third- and higher-order streams on SRS that have not been greatly affected by industrial operations, with shiners and sunfish dominating collections. The smaller tributaries to UTR are dominated by shiners and other small-bodied species (i.e., pirate perch, madtoms and darters) indicative of unimpacted streams in the Atlantic Coastal Plain. In the 1970s, the United States Geological Survey designated UTR as a National Hydrological Benchmark Stream due to its high water quality and rich fauna. However, this designation was rescinded in 1992 due to increased development of the UTR watershed north of SRS site boundaries. [DOE-EIS-0303, page 3-31]

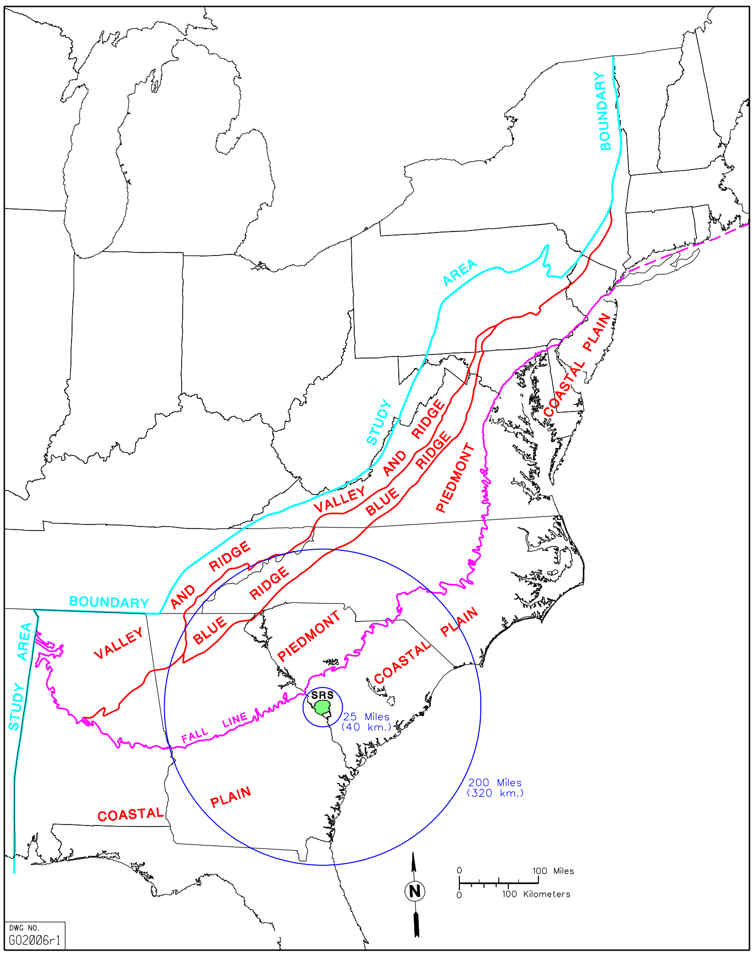
2.1.4 Geology, Seismology, and Volcanology

Regional and local information on the geologic and seismic characteristics of the FTF are presented in this section. Because SRS is not located within a region of active plate tectonics characterized by volcanism, volcanology is not an issue of concern in the FTF PA, and thus further discussion of this topic is omitted from the following discussion. [WSRC-IM-2004-00008, page 1.5-7]

2.1.4.1 Regional and Site-Specific Topography

The SRS is on the Atlantic Coastal Plain, Physiographic Province approximately 25 miles southeast of the Fall Line that separates the relatively unconsolidated Coastal Plain sediments. Beneath the Coastal Plain sedimentary sequence are two geologic terranes: 1) the Dunbarton basin, a Triassic-Jurassic Rift basin, filled with lithified terrigenous and lacustrine sediments; and 2) a crystalline terrane of metamorphosed sedimentary and igneous rock that may range in age from Precambrian to late Paleozoic from the crystalline igneous and metamorphic rocks of possibly late Precambrian to late Paleozoic age in the Piedmont Province. Early to middle Mesozoic (Triassic to Jurassic) rocks occur in isolated fault-bounded valleys either exposed within the crystalline belts or buried beneath the Coastal Plain sediments. The Coastal Plain sediments were derived from erosion of the crystalline rocks during late Mesozoic (Cretaceous) in stream and river valleys and are represented locally by gravel deposits adjacent to present-day streams and by sediments filling upland depressions (sinks and Carolina Bays). The Cretaceous and younger sediments are not significantly indurated. The total thickness of the sediment package at SRS varies between approximately 700 feet at the northwest boundary and 1,200 feet at the southeast boundary. [WSRC-TR-95-0046]

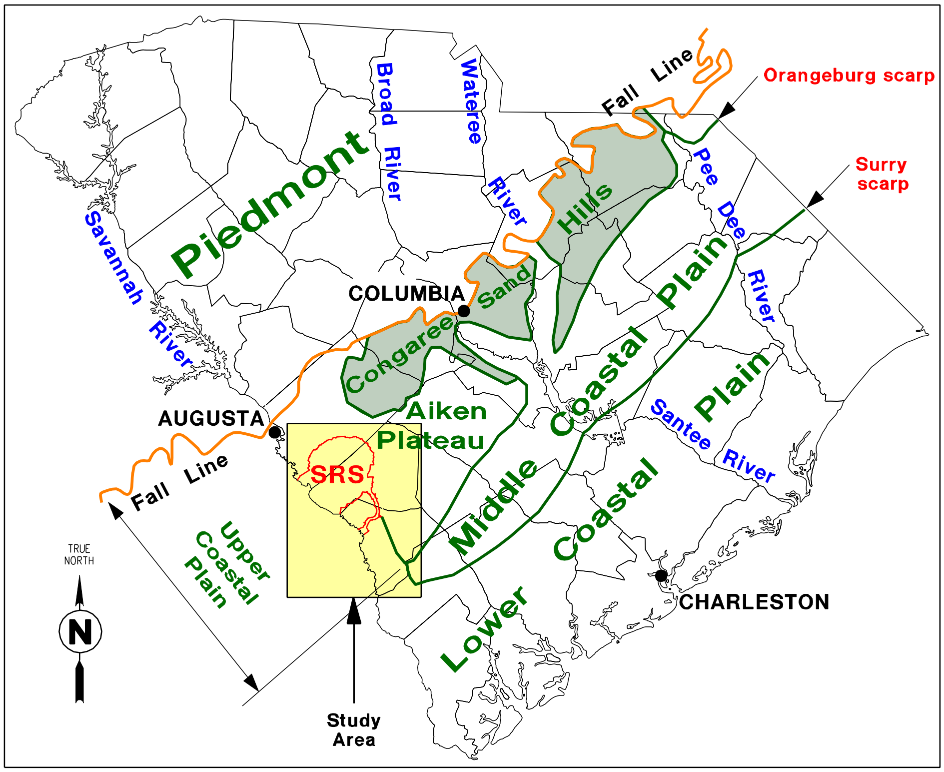
Figure 2.1-7: Regional Geological Provinces of Eastern United States



[WSRC-TR-95-0046, Figure 4-1]

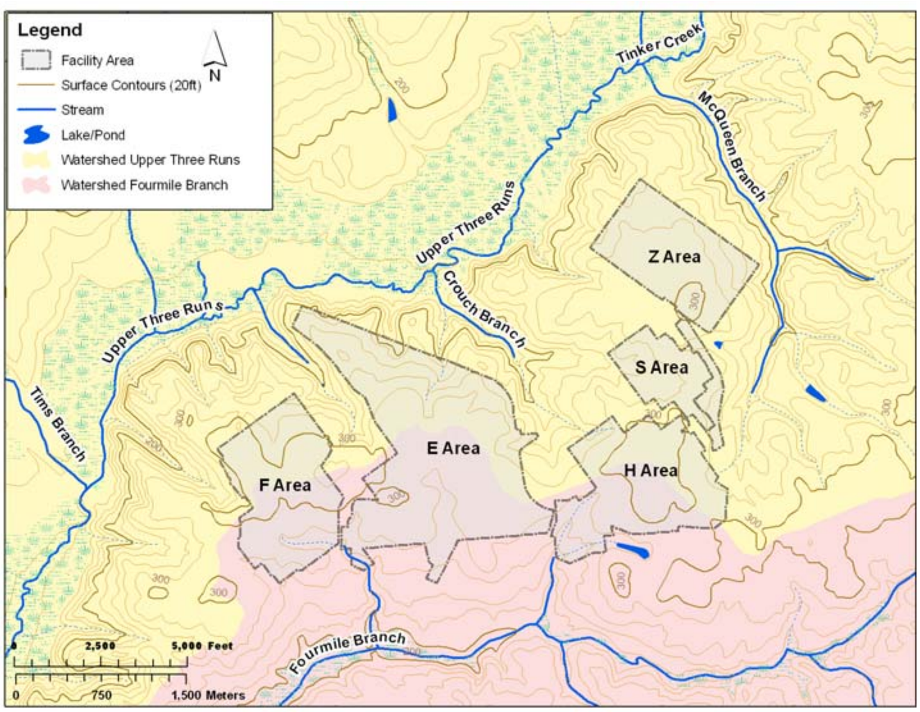
Figure 2.1-7 shows the relationship of SRS to overall regional geological provinces, and Figure 2.1-8 details the regional physiographic provinces in South Carolina. As can be seen on Figure 2.1-8, much of SRS lies within the Aiken Plateau, and this Plateau slopes to the southeast approximately 5 feet 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. [WSRC-TR-95-0046] Figure 2.1-9 shows the topography and 10-foot contour lines of the GSA. [SRS-REG-2007-00002]

Figure 2.1-8: Regional Geologic Provinces of South Carolina



[WSRC-TR-95-0046, Figure 2-3]

Figure 2.1-9: GSA Topography



Currently, FTF storm water drainage is directed to an outfall, which will be unaffected by FTF operations and tank closure activities. The installation of the FTF closure cap (Section 2.6) will necessitate changes to the FTF drainage system which will be designed later as part of the overall closure of FTF.

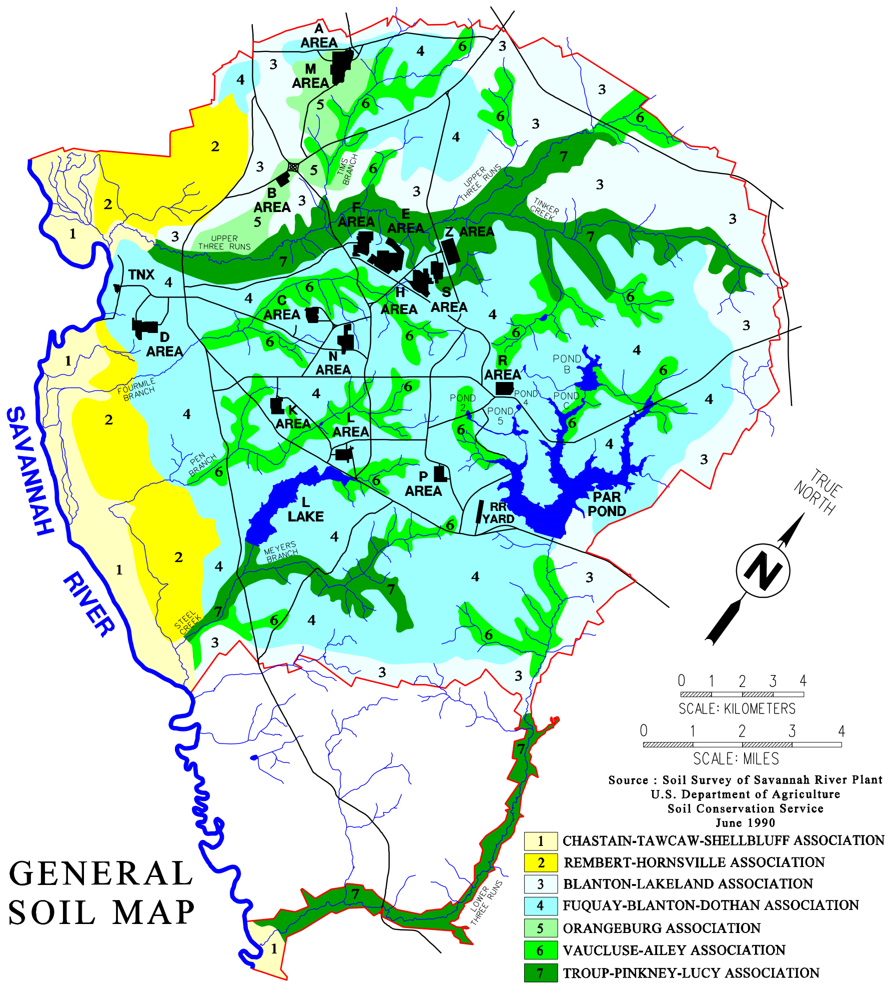
2.1.4.2 Local Geology and Soils

The vadose zone is comprised of the middle to late Miocene-age "Upland Unit," that extends over much of SRS. The term "Upland Unit" is an informal name used to describe sediments at higher elevations located in the Upper Coastal Plain in southwestern South Carolina. This area has also been referred to as the Aiken Plateau. The occurrence of cross-bedded, poorly sorted sands with clay lenses in the Aiken Plateau indicates fluvial deposition (high-energy channel deposits to channel-fill deposits) with occasional transitional marine influence. This depositional environment results in wide differences in lithology and presents a very complex system of transmissive and confining beds or zones. The lower surface of the "Upland Unit" is very irregular due to erosion of the underlying formations.

A notable feature of the "Upland Unit" is its compositional variability. This formation predominantly consists of red-brown to yellow-orange, gray and tan colored, coarse to fine grained sand, pebbly sand with lenses and beds of sandy clay and clay. Generally vertically upward through the unit, sorting of grains becomes poorer, clay beds become more abundant and thicker, and sands become more argillaceous and indurated. In some areas, small-scale joints and fractures, both of which are commonly filled with sand or silt, traverse the unit. The mineralogy of the sands and pebbles primarily consists of quartz, with some feldspars. In areas to the east-southeast, sediments may become more phosphatic and dolomitic. The soils at F Area may contain as much as 20% to 40% clay. [DOE-EIS-0303, pages 3-1 and 3-5]

The SRS is comprised of seven major soil associations: Chastain-Tawcaw-Shellbluff; Rembert-Hornsville; Blanton-Lakeland; Fuquay-Blanton-Dothan; Orangeburg; Vaucluse-Ailey; and Troup-Pinkney-Lucy. Figure 2.1-10 delineates the general soil associations for SRS. Details regarding these associations may be found in the *Soil Survey of the Savannah River Plant, U.S. Department of Agriculture, Soil Conservation Service*. [*http://soildatamart.nrcs.usda.gov/Manuscripts/SC696/0/savannah.pdf*]

Figure 2.1-10: General Soil Associations for SRS



[*http://soildatamart.nrcs.usda.gov/Manuscripts/SC696/0/savannah.pdf*]

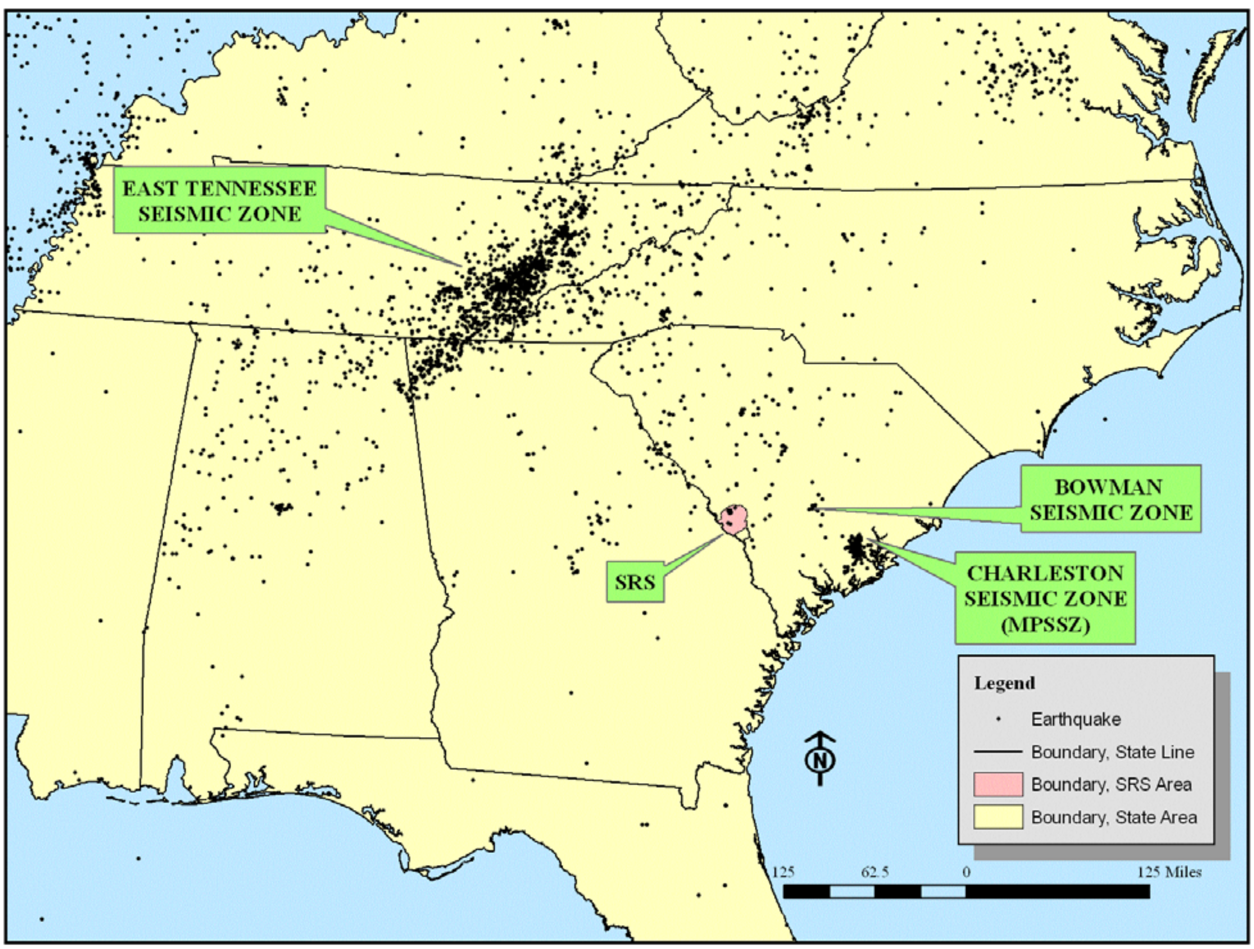
The overall general soil association for F Area is the Fuquay-Blanton-Dothan. The most predominant soil types within F Area are classified as Udorthents. Udorthents consist of well-drained soils that formed in heterogeneous materials, which are the spoil or refuse from excavations and major construction operations. The soil material has been removed, mixed and moved. Udorthents range from sandy to clayey, depending upon the source of material or geologic parent material. Udorthents are most commonly associated with well-drained to excessively drained upland soils. A few small, poorly drained areas that have spoil are also included. Typical profiles for Udorthents are not shown due to the unconsolidation within short distances. Clayey soil has demonstrated good retention for most radionuclides. There are also areas that consist of cross-bedded, poorly sorted sand with lenses and layers of silt and clay.

A more detailed description of the geology and soils of the F Area can be found in a report titled *Hydrogeologic Framework of West-Central South Carolina*. [PIT-MISC-0112]

2.1.4.3 Seismology

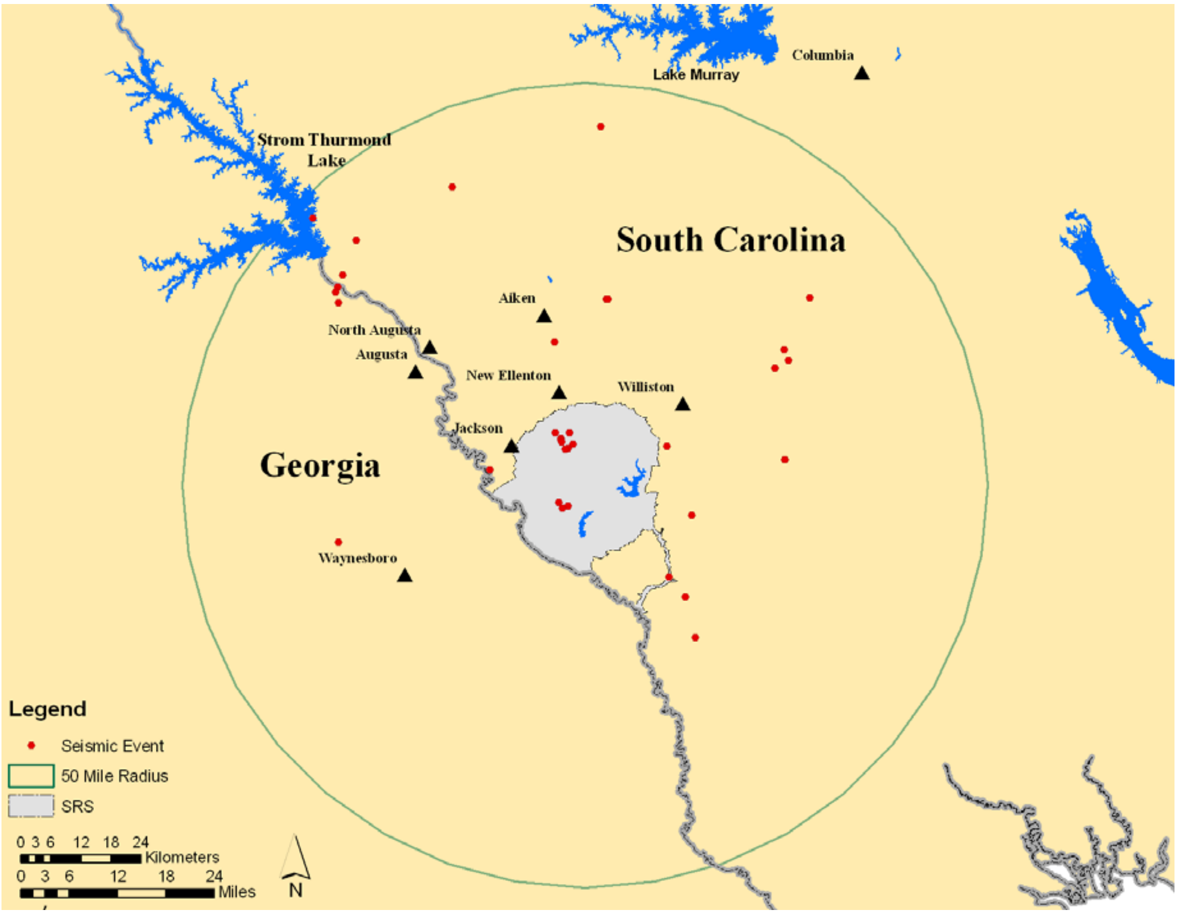
The seismic history of the southeastern U.S. (of which SRS is a part) spans a period of nearly three centuries, and is dominated by the Charleston earthquake of August 31, 1886 (estimated magnitude of 7.0). The historical database for the region is essentially composed of two data sets extending back to as early as 1698. The first set is comprised of pre-network, mostly qualitative data (1698-1974), and the second set covers the relatively recent period of instrumentally recorded or post-network seismicity, 1974 through April 2009. Figure 2.1-11 shows the locations of historical seismic events in the southeast. Figure 2.1-12 denotes the epicenter locations of seismic events within a 50-mile radius of SRS. [WSRC-MS-2003-00617, *http://neic.cr.usgs.gov/neis/last_event_states/*]

Figure 2.1-11: Historical Seismic Events in the Southeast



[*http://neic.cr.usgs.gov/neis/last_event_states/*]

Figure 2.1-12: Seismic Events within a 50-Mile Radius of SRS



[WSRC-MS-2003-00617, *http://neic.cr.usgs.gov/neis/last_event_states/*]

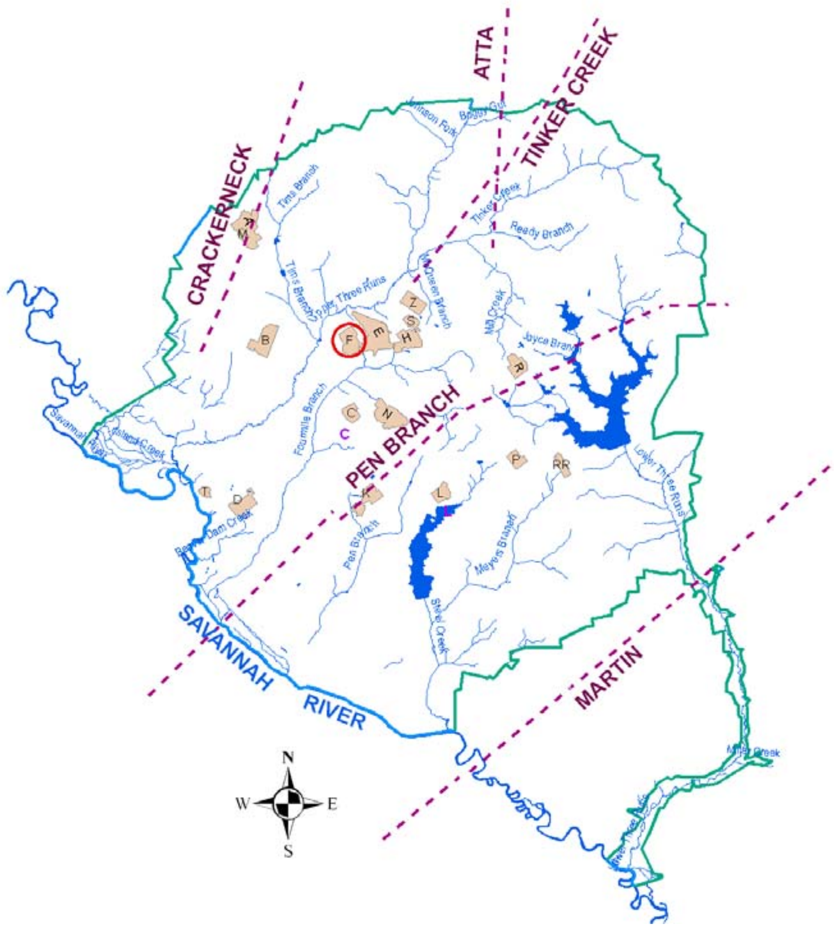
The most recent seismic event occurring within a 50-mile radius of SRS was on March 27, 2009, with a magnitude of 2.6. No damage to SRS was recorded. There have, however, been four earthquakes with epicenter locations within SRS. They occurred on June 9, 1985 (magnitude of 2.6); August 5, 1988 (magnitude of 2.0); May 17, 1997 (magnitude of 2.3), and October 8, 2001 (magnitude of 2.6). No Strong Motion Accelerometers (SMAs) were triggered as a result of these earthquakes. Note that additional seismic events with epicenter locations within SRS occurred shortly after the October 2001 earthquake, however, these seismic events were attributed to aftershocks and not actual earthquakes. [WSRC-MS-2003-00617]

The regional faults within SRS and vicinity are shown in Figure 2.1-13, a study entititled *Comparison of Cenozoic Fauling at the Savannah River Site to Fault Characteristics of the Atlantic Coast Fault Province: Implications for Fault Capability* (WSRC-TR-2000-00310) provides additional data. The study concludes that these regional faults exhibit the same general characteristics, and are closely associated with the faults of the Atlantic Coastal Fault Province, and thus are part of the Atlantic Coastal Fault Province. Several faults of the Atlantic Coastal Fault Province have been the subject of detailed investigations. In all cases, the conclusion has been reached that these faults have not had a movement within the past 35,000 years and no movement of a recurring nature within the past 500,000 years. Inclusion in the Atlantic Coastal Fault Province means that the historical precedent established by decades of previous studies on the seismic hazard potential for the Atlantic Coastal Fault Province is relevant to faulting at the SRS.

In 1976, a short-period seismic network was established. This network continues to be upgraded and in 1999 a 10-station SMA network was installed throughout the complex. Specific to F Area, one SMA is located near the tanks in the tank farm. Detailed information regarding seismic characteristics at SRS can be found in the Documented Safety Analysis document, WSRC-IM-2004-00008.

Seismic considerations are included in the design of the FTF closure cap to ensure seismic induced degradation mechanisms are addressed. Section 2.6 discusses the design of the FTF closure cap which will appropriately consider and handle static loading induced settlement, seismic induced liquefaction and subsequent settlement, and seismic induced slope instability.

Figure 2.1-13: Regional Scale Faults for SRS and Vicinity



[WSRC-TR-2000-00310, Figure 10]

2.1.5 Hydrogeology

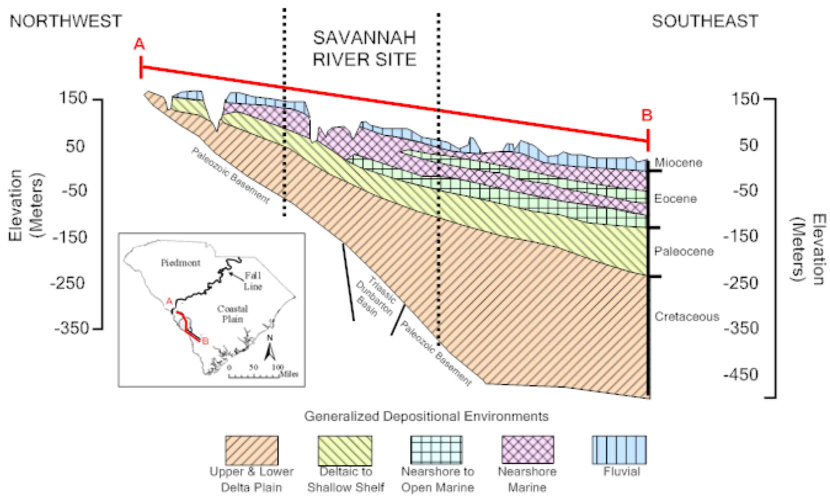
An understanding of the hydrogeology of the FTF is required in order for an estimate of the fate and transport of the residual FTF contaminants to be modeled. Characterization and monitoring data in the SRS GSA is extensive and provides a clear understanding of hydrogeology containing the FTF, and permitted generation of the General Separations Area Database (GSAD). Additional background information supporting this conclusion is presented in Section 2.1.5.2.

2.1.5.1 Regional Hydrogeology

The SRS lies in the Atlantic Coastal Plain, a southeast-dipping wedge of unconsolidated and semi-consolidated sediment, which extends from its contact with the Piedmont Province at the Fall Line to the continental shelf edge. Sediments range in geologic age from Late Cretaceous to Recent and include sands, clays, limestones and gravels. This sedimentary sequence ranges in thickness from essentially zero at the Fall Line to more than 1,219 meters (4,000 feet) at the Atlantic Coast. At SRS, coastal plain sediments thicken from approximately 213 meters (700 feet) at the northwestern boundary to approximately 430 meters (1,410 feet) at the southeastern boundary of the site and form a series of aquifers and confining or semi-confining units. Aquifer systems include the Floridan, Dublin and Midville systems. [WSRC-TR-96-0399-Vol. 1]

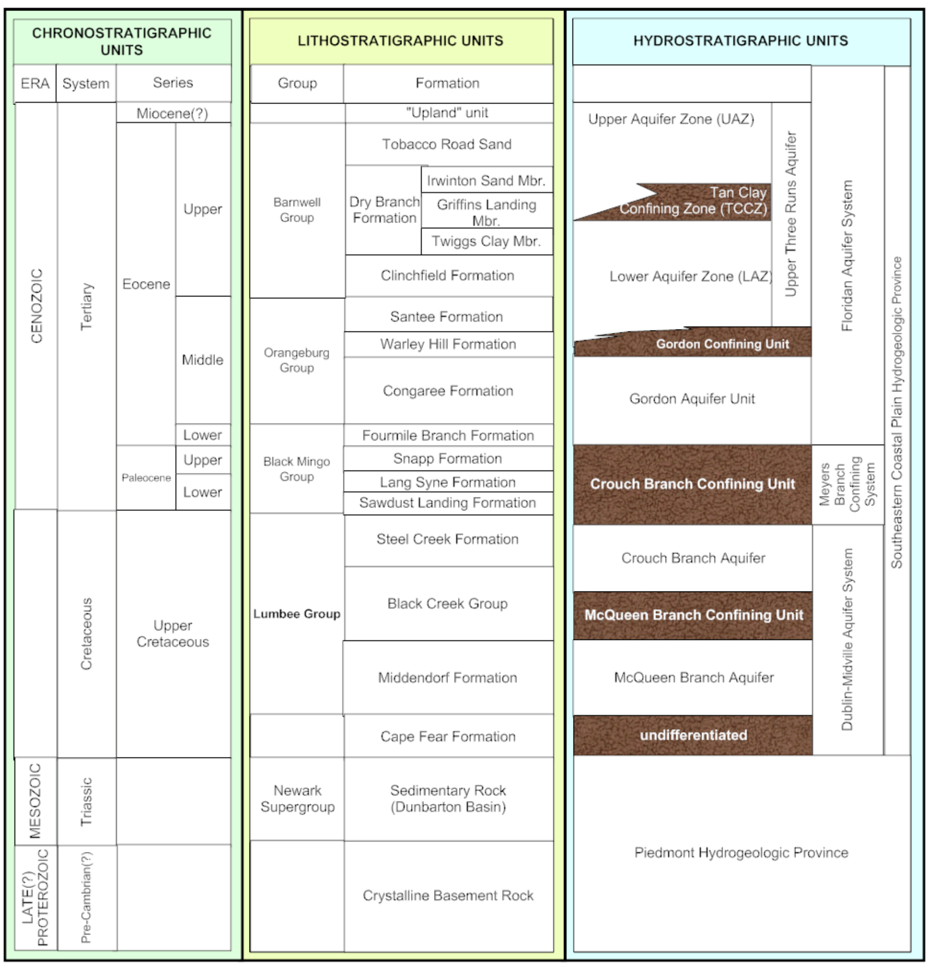
Figure 2.1-14 shows a generalized cross section of the sedimentary strata and their corresponding depositional environments for the Upper Coastal Plain down-dip through SRS into the Lower Coastal Plain. Figure 2.1-15 shows the regional lithologic units and their corresponding hydrostratigraphic units at SRS. This classification system is consistent with the established system, and is now widely used as SRS standard. [WSRC-TR-96-0399-Vol. 1, WSRC-TR-95-0046, WSRC-STI-2008-00057, Page 7-1]

Figure 2.1-14: Regional NW to SE Cross Section



[WSRC-TR-95-0046]

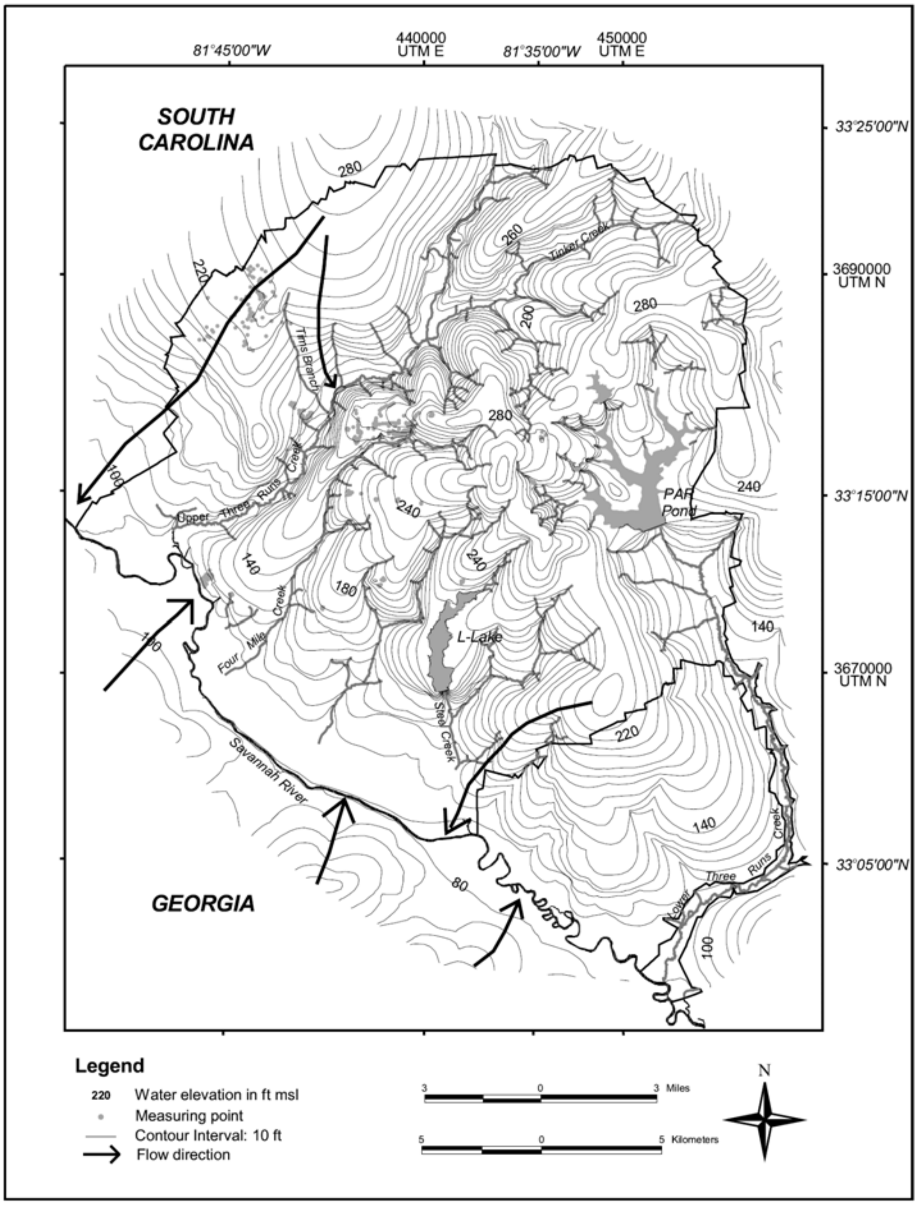
Figure 2.1-15: Comparison of Chronostratigraphic, Lithostratigraphic, and Hydrostratigraphic Units in the SRS Region



[WSRC-TR-95-0046]

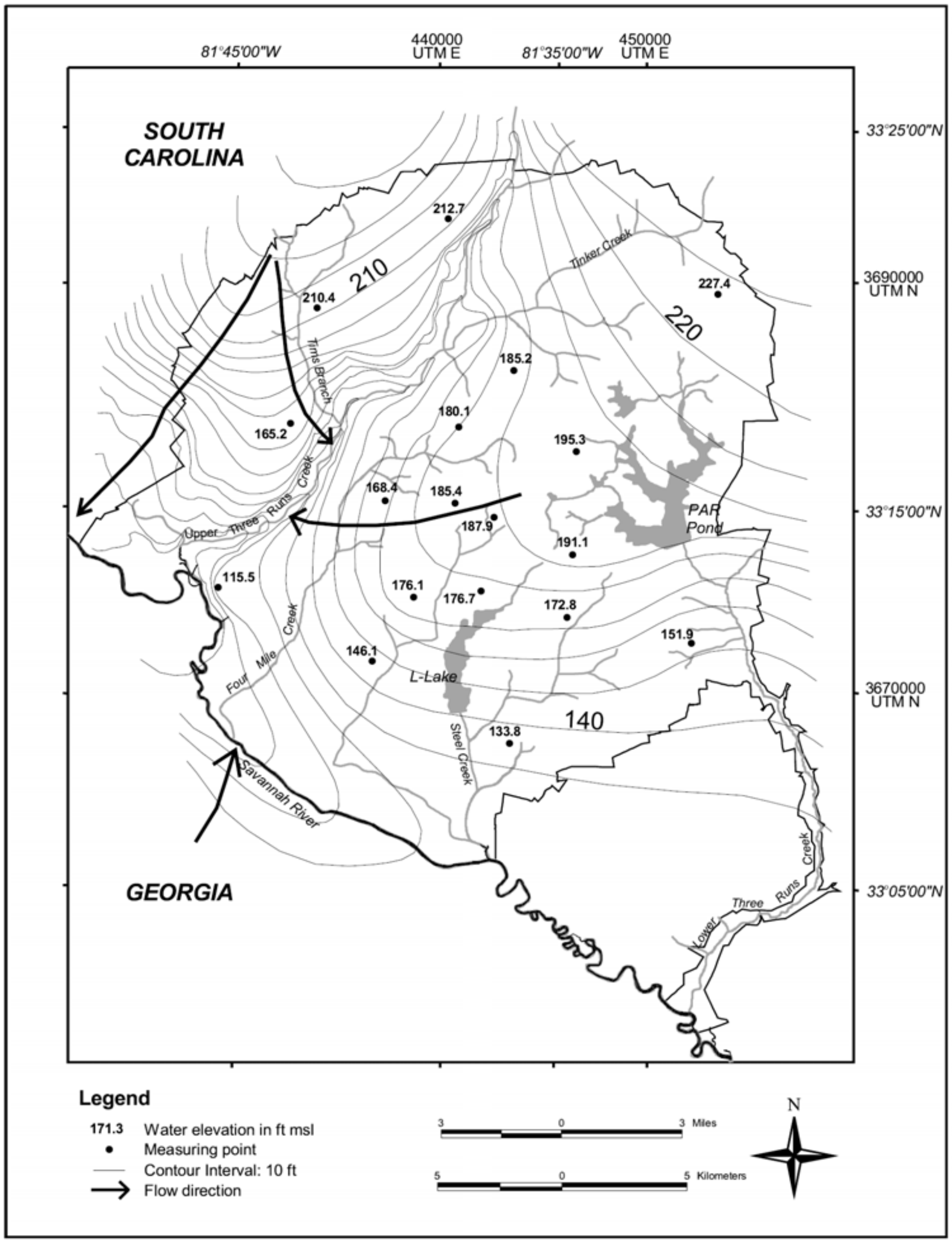
Figure 2.1-16 and Figure 2.1-17 illustrate potentiometric maps of the UTR and Gordon Aquifers. Groundwater within the Floridan Aquifer system flows toward streams and swamps and into the Savannah River at rates ranging from inches to several hundred feet per year. The depth to which nearby streams cut into sediments, the lithology of the sediments and the orientation of the sediment formations control the horizontal and vertical movement of the groundwater. The valleys of smaller perennial streams, such as Fourmile Branch, McQueen Branch and Crouch Branch in the GSA, allow discharge from the shallow saturated geologic formations. The valleys of major tributaries of the Savannah River (e.g., UTR) drain formations of greater depth. With the release of water to the streams, the hydraulic head of the aquifer unit releasing the water can become less than that of the underlying unit. If this occurs, groundwater has the potential to migrate upward from the lower unit to the overlying unit. [DOE-EIS-0303, page 3-10]

Figure 2.1-16: Potentiometric Surface of the Upper Three Runs Aquifer



[WSRC-STI-2008-00057-SRS Maps, Page 18]

Figure 2.1-17: Potentiometric Surface of the Gordon Aquifer

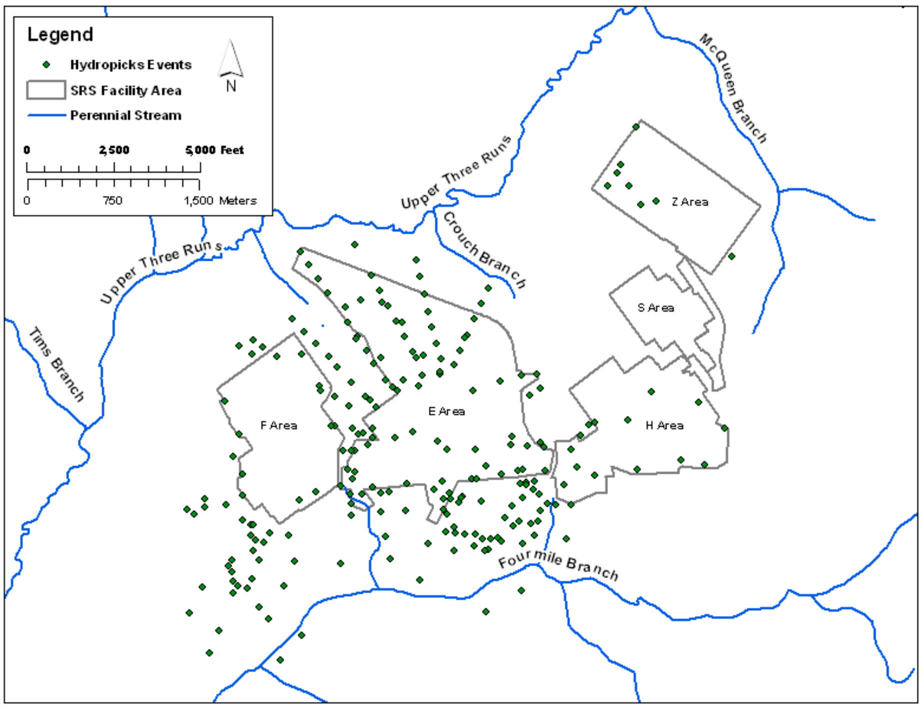


[WSRC-STI-2008-00057-SRS Maps, Page 19]

2.1.5.2 Characterization of Local Hydrogeology

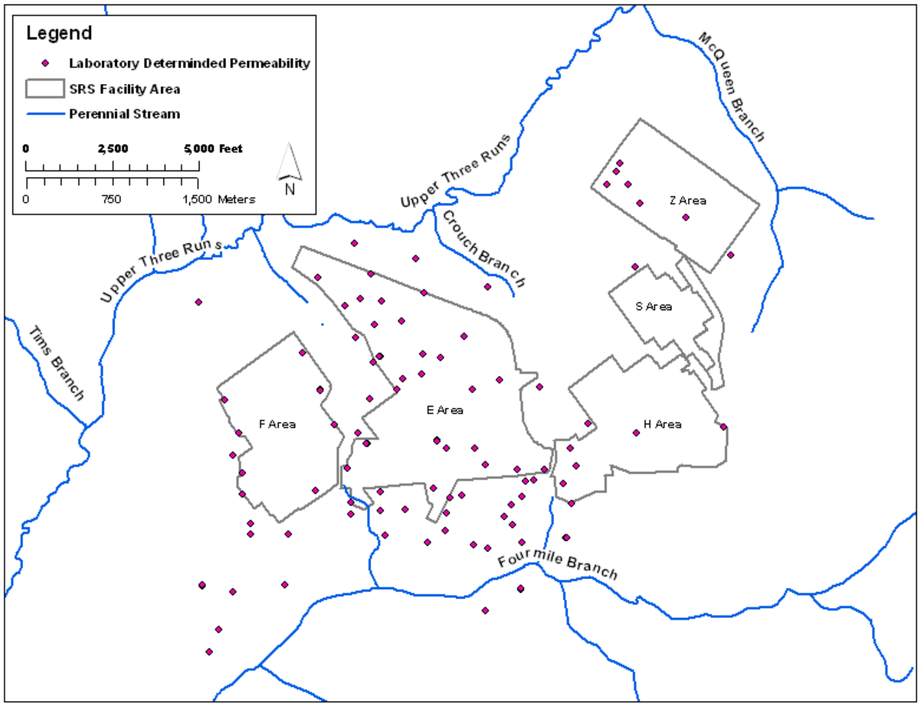
The GSA has been the focus of numerous geological and hydrogeologic investigations. Early work included installation of monitoring wells in the 1950s and 1960s. Further characterization and monitoring were conducted in the area during the 1970s through present time, largely to support groundwater monitoring and decommissioning activities. The GSAD was developed using field data and interpretations for the GSA and vicinity through 1996. Although characterization and monitoring have been ongoing, the additional data has not altered fundamental understanding of groundwater flow patterns and gradients in the GSA. The GSAD is a subset of site-wide data sets of soil lithology and groundwater information. Figure 2.1-18 shows the location of all hydrostratigraphic picks used in the GSAD. Picks were made based on a combination of geophysical logs, cone penetration test logs and core descriptions. Figures 2.1-19 through 2.1-22 show locations of laboratory permeability data, multiple well pump tests, single well pump test and slug test data used in the GSAD. Table 2.1-2 presents a summary of the characterization and monitoring data in the GSAD. These data provide detailed understanding of local hydrogeology beneath the FTF. See WSRC-TR-96-0399, Volumes 1 and 2 for a more comprehensive discussion of the data set. The GSAD, comprising SRS characterization and monitoring data and interpretations is used as the basis of hydrogeologic input values into the computational model for groundwater flow and contaminant transport as described in Section 4.0 of the FTF PA. [SRS-REG-2007-00002]

Figure 2.1-18: Hydrostratigraphic Picks in GSAD Database



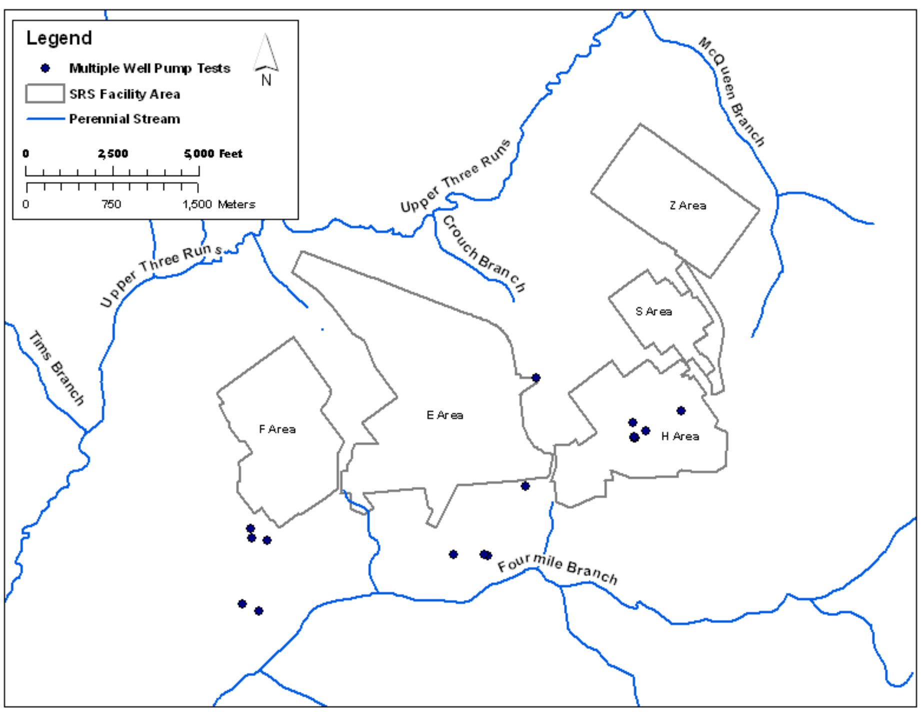
[WSRC-TR-96-0399-Vol. 2]

Figure 2.1-19: Laboratory Determined Permeability Data in GSAD Database



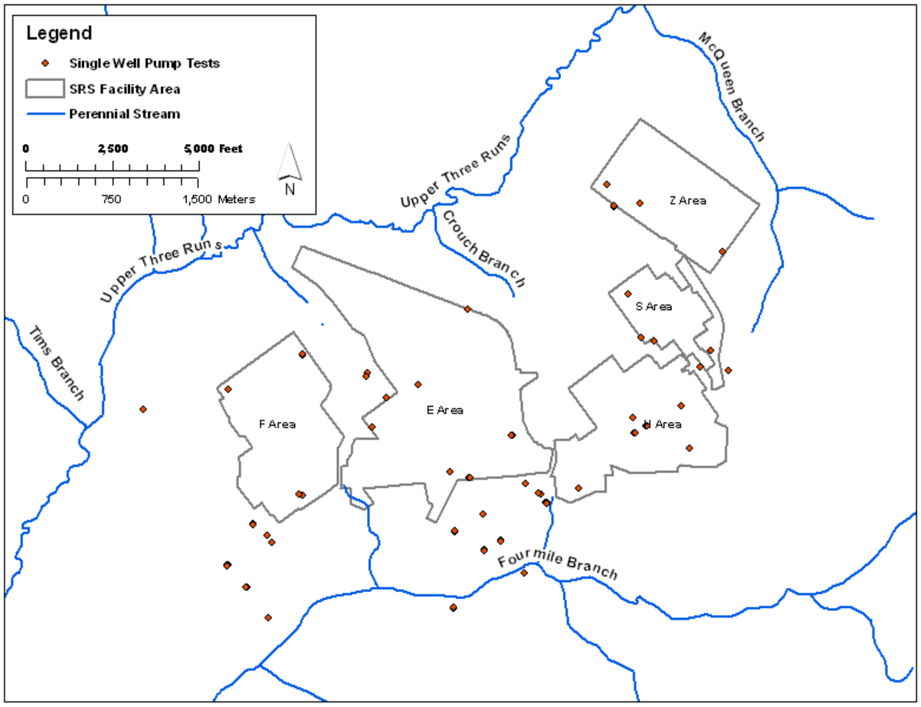
[SRNL-ESB-2007-00035]

Figure 2.1-20: Multiple Well Pump Test Data in GSAD Database



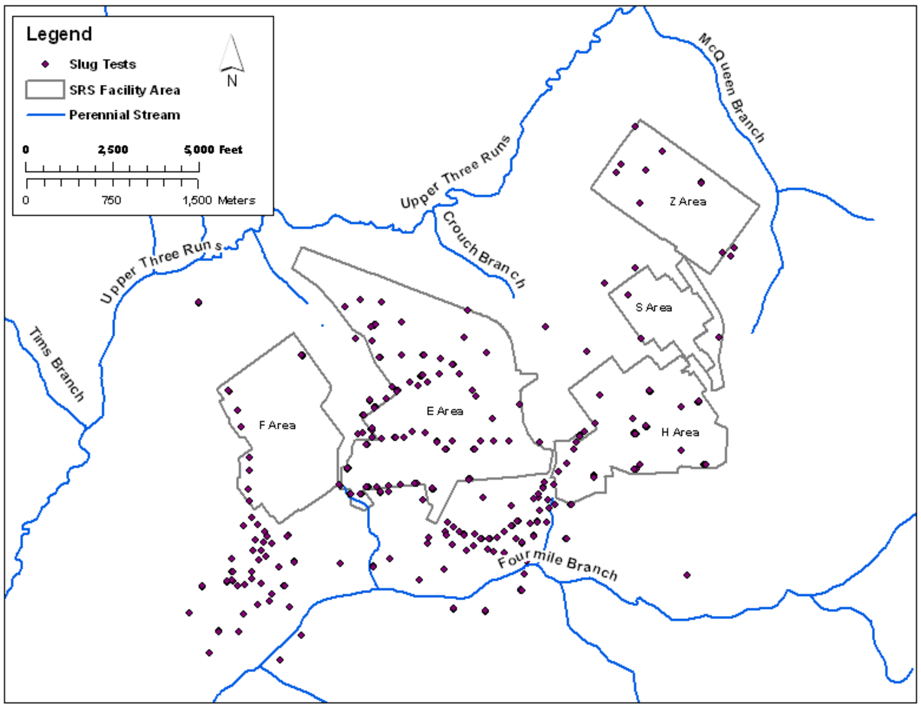
[WSRC-TR-96-0399-Vol. 2]

Figure 2.1-21: Single Well Pump Test Data in GSAD Database



[WSRC-TR-96-0399-Vol. 2]

Figure 2.1-22: Slug Test Data in GSAD Database



[WSRC-TR-96-0399-Vol. 2]

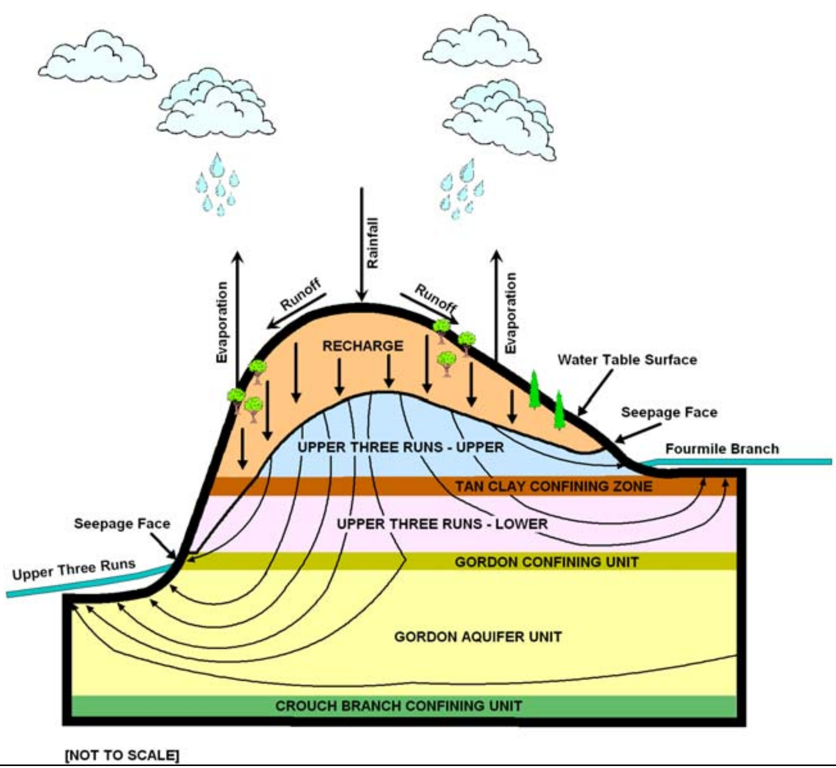
Table 2.1-2: Characterization and Monitoring Data in the GSAD Database

Data Type	Quantity	Reference
Sediment Core Descriptions	204 Locations; ~37,500 feet	WSRC-TR-96-0399-Vol. 1, App. B
Tops of Hydrostratigraphic Units/Zones		
Gordon Aquitard Gordon Aquifer Green Clay Aquitard Upper Three Runs – Lower Zone (UTR-LZ) Tan Clay Aquitard	52 Locations 146 Locations 161 Locations 222 Locations 225 Locations	WSRC-TR-96-0399-Vol. 1, App. A
Permeability Measurements		
Pumping Tests Slug Tests Laboratory Permeability	85 Values 481 Values 258 Values	WSRC-TR-96-0399-Vol. 2, App. B
Water Levels		
Gordon Aquifer UTR-LZ Upper Three Runs – Upper Zone	79 Locations 173 Locations 387 Locations	WSRC-TR-96-0399-Vol. 2, App. C

2.1.5.3 Groundwater Flow in the GSA

The aquifers of primary interest for FTF modeling are the UTR and Gordon Aquifers. Plate 17 of the *Hydrogeological Framework of West-Central South Carolina Report* (PIT-MISC-0112) gives the leakance of the Crouch Branch confining unit (of the Meyers Branch confining system) as roughly 3E-06 day^{-1}, which corresponds to 0.13 inch/year for every 10 feet of head difference. The head difference across the Crouch Branch confining unit ranges from 0 to 20 feet, causing an upward flow averaging 0.13 inch/year. [PIT-MISC-0112, Figure 30] Flow across the unit is therefore a small fraction of total recharge, and is negligible in the FTF modeling. Potential contamination from the FTF is not expected to enter the deeper Crouch Branch Aquifer because an upward gradient exists between the Crouch Branch and Gordon Aquifers near UTR. Figure 2.1-23 is a cross-sectional schematic representation of groundwater flow patterns in the UTR and Gordon Aquifers along a north-south cross-section running through the center of FTF, shown with significant vertical exaggeration. Section 4.0 of the FTF PA provides the modeling inputs associated with groundwater flow characteristics obtained from the GSAD. [SRS-REG-2007-00002]

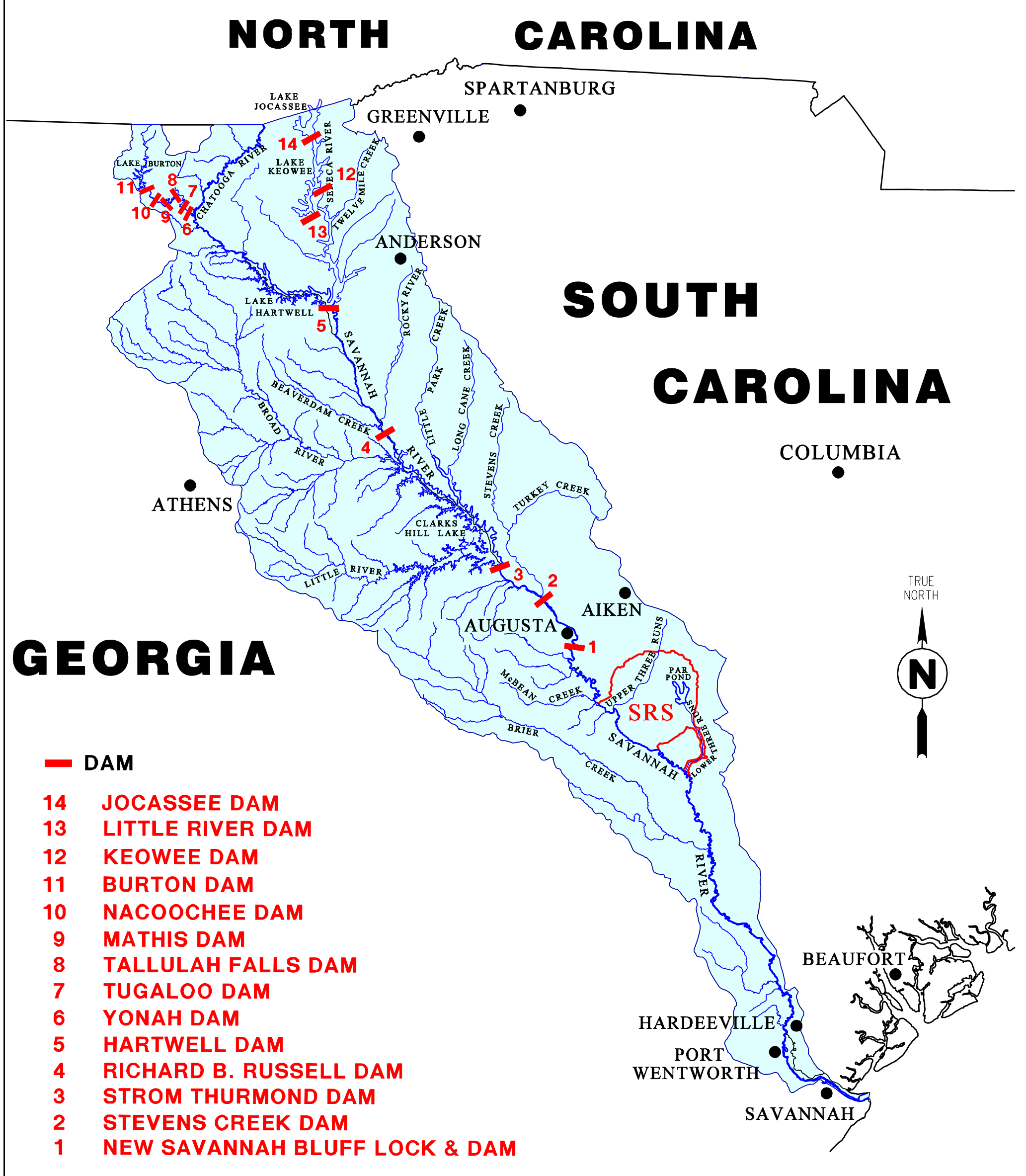
Figure 2.1-23: Conceptual Diagram of Groundwater Flow beneath the GSA



2.1.5.4 Surface-Water Flow in the GSA

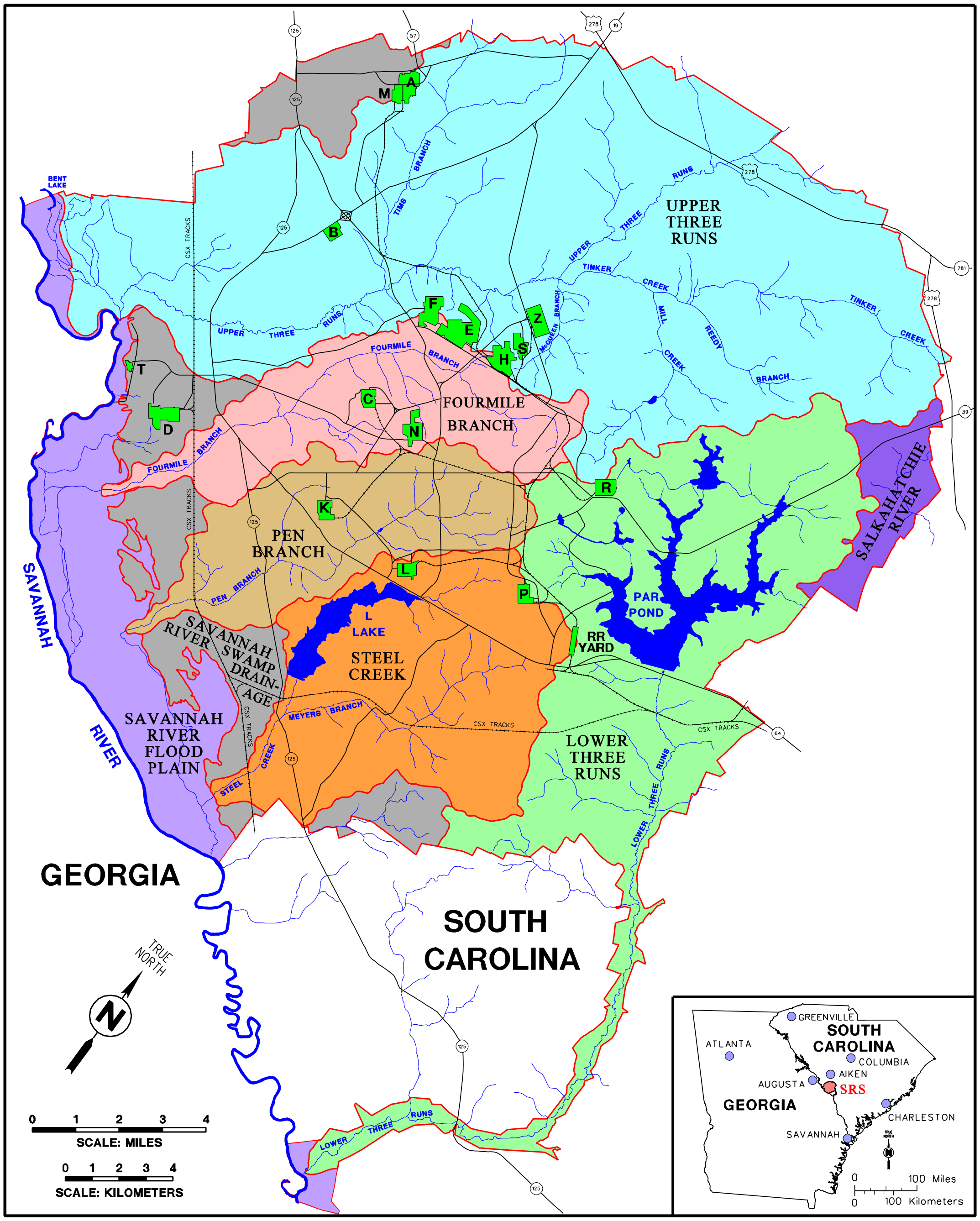
The Savannah River, which forms the boundary between Georgia and South Carolina, is the principal surface-water system near SRS. The river adjoins the site along its southwestern boundary for a distance of approximately 20 miles, and the site is 160 river-miles from the Atlantic Ocean. Five upstream reservoirs – Jocassee, Keowee, Hartwell, Richard B. Russell and Clarks Hill Lake (also known as Thurmond Lake), minimize the effects from droughts and the impacts of low flow on downstream water quality and fish and wildlife resources in the river. Figure 2.1-24 shows the Savannah River Basin dams. The long-term yearly Savannah River flow averages approximately 10,400 cubic feet per second at SRS. For 2007, the annual average measured flow rate was 6,090 cubic feet per second. [WSRC-TR-2005-00201, Table 4-24, WSRC-STI-2008-00057]

Figure 2.1-24: Savannah River Basin Dams



The major tributaries that occur on SRS are UTR, Fourmile Branch, Pen Branch, Steel Creek and Lower Three Runs (Figure 2.1-25). These tributaries drain all of SRS with the exception of a small area on the northeast side, which drains to a tributary of the Salkehatchie River. Each of these streams originates on the Aiken Plateau in the Coastal Plain and descends 50 to 200 feet before discharging into the river. The source of most of the surface water on SRS is either natural rainfall (Section 2.1.2), water pumped from the Savannah River and used for cooling site facilities, or groundwater discharging to surface streams. The streams, which historically have received varying amounts of effluent from SRS operations, are not commercial sources of water. Downstream of SRS, Savannah River supplies domestic water and is used for commercial and sport fishing, boating and other recreational activities. [DOE-EIS-0303, page 3-7]

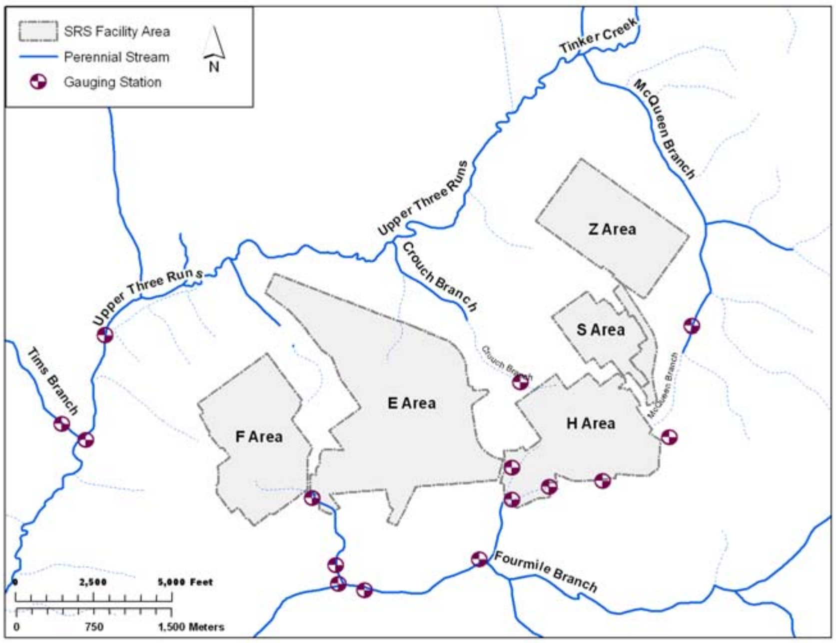
Figure 2.1-25: SRS Watershed Boundaries and Major Tributaries



[WSRC-STI-2008-00057]

The natural flow of SRS streams range from 8 cubic feet per second in smaller streams to 245 cubic feet per second in UTR. [WSRC-IM-2004-00008, Table 1.4-14] Gauging stations located in the GSA (Figure 2.1-26) monitor flows for UTR and Fourmile Branch. Both Fourmile Branch and UTR are measured monthly for water flow, temperature and quality. The annual *Savannah River Site Environmental Report* contains detailed information on flow rates and water quality of the Savannah River and SRS streams. [WSRC-STI-2008-00057]

Figure 2.1-26: GSA Gauging Stations



The SCDHEC regulates the physical properties and concentrations of chemicals and metals in SRS effluents under the National Pollutant Discharge Elimination System (NPDES) program. SCDHEC, which also regulates biological Water Quality Standards for SRS waters, have classified the Savannah River and SRS streams as "Freshwaters." "Freshwaters" are described as suitable for primary and secondary contact recreation and as a source for drinking water supply after treatment in accordance with SCDHEC requirements. "Freshwaters" are suitable for fishing, for the survival and propagation of a balanced indigenous aquatic community of fauna and flora and for industrial and agricultural uses. [DOE-EIS-0303]

The UTR, longest of the SRS streams, is a large blackwater stream in the northern part of SRS that discharges to the Savannah River. It drains an area of over 195 square miles and is approximately 25 miles long, with its lower 17 miles within SRS boundaries. This stream receives more water from underground sources than other SRS streams and is the only stream with headwaters arising outside the site. It is the only major tributary on SRS that has not received thermal discharges. The UTR valley has meandering channels, especially in the lower reaches, and its floodplain ranges in width from 0.25 to 1 mile. It has a steep southeastern side and gently sloping northwestern sides. [DOE-EIS-0303]

Fourmile Branch is a blackwater stream that originates near the center of SRS and flows southwest for 15 miles before emptying into the Savannah River. It drains an area of approximately 22 square miles inside SRS including much of F, H, and C Areas. Fourmile Branch flows parallel to the Savannah River behind natural levees and enters the river through a breach downstream from Beaver Dam Creek. In its lower reaches, Fourmile Branch broadens and flows via braided channels through a delta formed by the deposition of sediments eroded from upstream during high flows. Downstream from the delta, the channels rejoin into one main channel. Most of the flow discharges into the Savannah River while a small portion flows west and enters Beaver Dam Creek. The valley is v-shaped, with sides varying from fairly steep to gently sloping. The floodplain is up to 1,000 feet wide. [DOE-EIS-0303, pages 3-8 and 3-9]

Flood hazard recurrence frequencies have been calculated for the various SRS site areas. The 100-year, 1,000-year and 10,000-year flood water levels for UTR basin near F Area is approximately 138, 140 and 143 feet above mean sea level (MSL), respectively. [WSRC-TR-99-00369] As shown in Table 2.1-6, the lowest elevation of a waste tank within FTF is approximately 227 feet above MSL; thus the highest period flood water level of approximately 143 feet above MSL is approximately 84 feet below the lowest elevation of an FTF waste tank. In addition, the lowest elevation of the lower foundation layer, at the bottom of the side slope of the conceptual closure cap is approximately 270 feet above MSL, which is approximately 127 feet above the highest flood water level of 143 feet. [WSRC-STI-2007-00184] Therefore, flooding is not a concern for the FTF PA.

2.1.6 Geochemistry

The migration of radionuclides in the subsurface environment is dependent on physical and chemical parameters or properties of cementitious materials, soils and groundwater. Studies and analyses have been conducted to determine appropriate Distribution Coefficients (K_ds) and are identified below. The data used in the radionuclide transport model is presented in Section 4.0 of the FTF PA specific to the GSA and is not reproduced in this section. [SRS-REG-2007-00002] The following studies detail the information:

- WSRC-TR-2006-00004, *Geochemical Data Package for Performance Assessment Calculations Related to the Savannah River Site*
- SRNL-ESB-2007-00008, F-*Area Tank Farm Vadose Zone Material Property Recommendations*
- WSRC-STI-2007-00544, *Conceptual Model of Waste Release from the Contaminated Zone of Closed Radioactive Waste Tanks*
- SRNL-SCS-2007-00011, *Preliminary Guidance for the Distribution of Cs, SR, and U, Geochemical Input Terms to Stochastic Transport Models*

2.1.7 Natural Resources

Natural resources at SRS are managed under the *Natural Resources Management Plan* (NRMP) prepared for the DOE by the United States Department of Agriculture. [NRMP-2005] The NRMP was recently updated in May 2005, and fosters the following principles which govern SRS natural resource management:

- All work will be done in accordance with Integrated Safety Management Procedures found in DOE P 450.4, *Safety Management System Policy.*
- Environmental stewardship activities will be compatible with future SRS missions.
- The SRS will continue to protect and manage SRS natural resources.
- Sustainable resource management will be applied to SRS natural resources.
- Close cooperation will be maintained among organizations when managing and protecting SRS natural resources.
- The results of research, monitoring and operational findings will be used in the management of SRS natural resources.
- Restoration of native communities and species will continue.
- Employees, customers, stakeholders, state natural resource officials and regulators will be invited to participate in the natural resource planning process.
- The SRS will maintain the area as a National Environmental Research Park.

2.1.7.1 Water Resources

The SRS monitors non-radioactive liquid discharges to surface waters through the state-administered NPDES permit program and requirements, under the Federal Water Pollution Control Act (commonly called the Clean Water Act). [*http://epw.senate.gov/water.pdf*] As required by EPA and SCDHEC, SRS has NPDES permits in place for discharges to the waters of the United States. These permits establish the specific sites to be monitored, parameters to be tested, and monitoring frequency—as well as analytical, reporting, and collection methods. [WSRC-STI-2008-00057, page 4-7] Continuous surveillance monitoring of site streams occurs downstream of several process areas to detect and quantify levels of radioactivity in effluents transported to the Savannah River. [WSRC–STI–2008–00057, page 5-3]

2.1.7.1.1 Surface Water

The SRS streams and the Savannah River are classified by SCDHEC as “Freshwaters,” which are defined as surface water suitable for:

- primary and secondary contact recreation and as a drinking water source after treatment in accordance with SCDHEC requirements and

- fishing and survival and propagation of a balanced indigenous aquatic community of fauna and flora, and Industrial and agricultural uses. [WSRC–STI–2008–00057]

Table 2.1-3 characterizes Savannah River water quality both upstream and downstream of SRS. Table 2.1-4 characterizes water quality in the UTR upstream and downstream of the GSA.

Table 2.1-3: Water Quality in the Savannah River Upstream and Downstream from SRS (Calendar Year 2007)

		Upstream		Downstream	
Parameter	Unit of Measure	Minimum	Maximum[a]	Minimum	Maximum[a]
Aluminum	mg/L	0.062	0.97	0.0777	0.622
Cadmium	mg/L	ND	ND	ND	ND
Chromium	mg/L	ND	ND	ND	ND
Copper	mg/L	ND	0.0388	ND	0.0083
Dissolved Oxygen	mg/L	6.1	11.1	6.2	10.1
Gross Alpha Radioactivity	pCi/L	ND	1.26	ND	1.55
Lead	mg/L	ND	ND	ND	0.0023
Mercury	mg/L	ND	0.000023	ND	0.000024
Nickel	mg/L	ND	0.003	ND	0.0066
Nitrate (as N)	mg/L	0.18	0.57	0.19	0.41
pH	pH units	5.9	7.1	6.6	16.1[b]
Phosphate	mg/L	0.032	0.18	0.084	0.17
Suspended solids	mg/L	1	47	4	15
Temperature	°F	52.5	78.6	53.2	82.4
Tritium	pCi/L	ND	229	140	1,060
Zinc	mg/L	ND	0.0087	ND	0.0173

a The maximum listed concentration is the highest single result found during one sampling event.
b Highest one month sample, next highest value reported was 7.2 pH unit.
Note: Information extracted from WSRC-STI-2008-00057 accompanying data files. Parameters are those DOE routinely measures as a regulatory requirement, or as part of ongoing monitoring programs.
ND - Non Detectable

Table 2.1-4: Water Quality in Upper Three Runs

	Temperature (°F)	pH	Dissolved Oxygen (mg/L)	Total Suspended Solids (mg/L)
Sampling Location: Upper Three Runs (Upstream of GSA)				
Mean	61	5.9	9.1	4.6
Range	51 - 70	5.4 - 6.7	4.4 - 14	3 - 9
Sampling Location: Upper Three Runs (Downstream of GSA)				
Mean	65	6.5	9.6	6.2
Range	52 - 74	5.2 - 7.6	6.4 - 16	4 - 11

Note: All data extracted from WSRC-STI-2008-00057 accompanying data files

2.1.7.1.2 Groundwater

The SDWA was enacted in 1974 to protect public drinking water supplies. [*www.epa.gov/safewater/sdwa/index.html*] SRS domestic water is supplied by 17 separate systems, all of which utilize groundwater sources. The A-Area, D-Area, and K-Area systems are actively regulated by SCDHEC, while the remaining 14 site water systems receive a reduced level of regulatory oversight. [WSRC-STI-2008-00057]

Table 2.1-5 provides the summary of maximum groundwater monitoring results for those areas that most likely outcrop to Fourmile Branch obtained from the *2007 Environmental Report*. [WSRC-STI-2008-00057] The groundwater in these areas is not being consumed and active remediation projects are in progress to address the groundwater conditions.

Table 2.1-5: Summary of Maximum Groundwater Monitoring Results for Major Areas that Outcrop to Fourmile Branch, 2006–2007

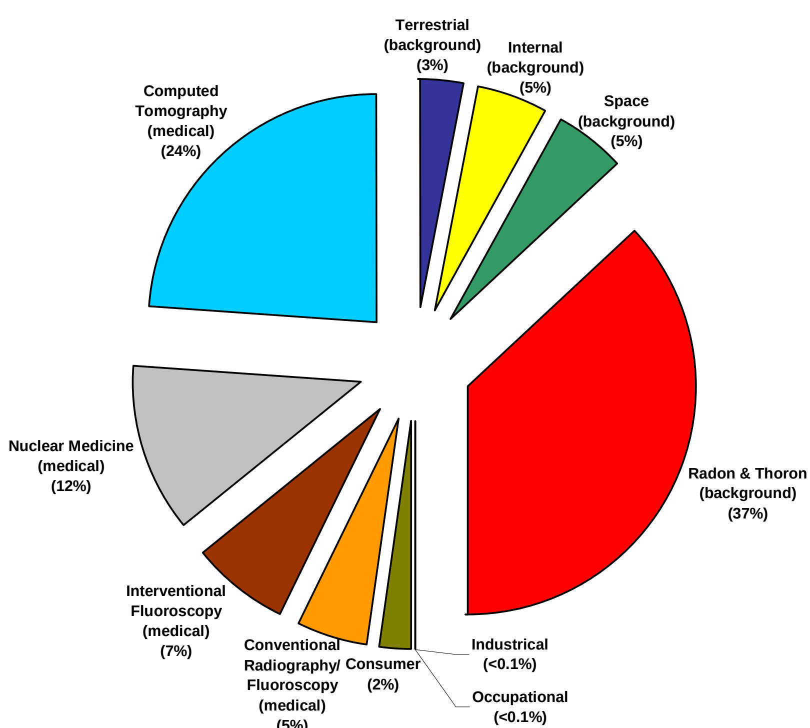
Location	Major Contaminants	Units	2006 Maximum	MCL	2007 Maximum	Likely Outcrop Point
E-Area	Tritium TCE	pCi/L ppb	33,600,000 750	20,000 5	30,800,000 370	UTR/Crouch Branch in North; Fourmile Branch in South
F-Area	TCE Tritium Gross alpha Beta	ppb pCi/L pCi/L pCi/L	78.9 91,500 2,030 1,620	5 20,000 15 4 mrem/yr[a]	52.2 73,000 2,120 380	UTR/Crouch Branch in North; Fourmile Branch in South
F-Area Seepage Basin	Tritium Gross alpha Beta	pCi/L pCi/L pCi/L	7,140,000 627 2,360	20,000 15 4 mrem/hr[a]	5,710,000 523 1,870	Fourmile Branch
H-Area	Tritium Gross alpha Beta	pCi/L pCi/L pCi/L	80,400 98 116	20,000 15 4 mrem/yr[a]	67,200 25.5 55.6	UTR/Crouch Branch in North; Fourmile Branch in South
H-Area Seepage Basins	Tritium Gross alpha Beta	pCi/L pCi/L pCi/L	3,690,000 103 2,840	20,000 15 4 mrem/yr[a]	3,020,000 88.4 2,970	Fourmile Branch

[a] The activity (pCi/L) equivalent to 4 mrem/yr varies according to which specific beta emitters are present in the sample. [WSRC-STI-2008-00057, Page 7-9]

2.1.8 Natural and Background Radiation

All human-beings are exposed to sources of ionizing radiation that include naturally occurring and man-made sources. An individual's average dose contribution estimates from various sources were obtained from the review information presented in NCRP Report 160 and are shown in Figure 2.1-27. [NCRP-160] On average, a person living in the United States or the Central Savannah River Area (CSRA) receives approximately the same annual radiation dose of 620 mrem/year. The dose from SRS operations to the maximally exposed offsite individual during calendar year 2007 was estimated to be 0.1 mrem. [WSRC-STI-2008-00057, page 6-7]

Figure 2.1-27: Major Sources of Radiation Exposure in the Vicinity of SRS



[NCRP-160]

The major sources of radiation exposure to an average member of the public in the CSRA is attributed to naturally occurring radiation (311 mrem/yr) and medical exposure (300 mrem/yr). This naturally occurring radiation is often referred to as natural background radiation and includes dose from background radon and its decay products (228 mrem/yr), cosmic radiation (33 mrem/yr), internal radionuclides occurring naturally in the body (29 mrem/yr) and natural radioactive material in the ground (21 mrem/yr). The dominant medical sources include dose from computed tomography (147 mrem/yr), nuclear medicine (77 mrem/yr) and radiography/fluoroscopy (77 mrem/yr). The remainder of the dose is from consumer products (13 mrem/yr), industrial/educational/research activities (<1 mrem/yr) and occupational exposure (<1 mrem/yr). [NCRP-160]

2.1.9 Tank Farm Operations

Since initiation of operations at SRS, the tank farms combined have received over 140,000,000 gallons of liquid waste. The vast majority of this liquid waste originated from the chemical separation processes in F and H Canyons associated with three major missions: 1) reprocessing of spent nuclear fuel; 2) production of nuclear materials for weapons; and 3) production of material for NASA space missions. [SRR-LWP-2009-00001] In addition to the Canyon waste streams, the DWPF returns a liquid waste stream (referred to as "DWPF recycle") to the FTF and HTF. This "DWPF recycle" liquid waste stream is a by-product of the production of waste canisters in DWPF. These canisters contain SRS high-level waste stabilized in borosilicate glass. [DOE-EIS-0303]

The F- and H-Canyon chemical separation processes used acids to dissolve irradiated reactor target or fuel assemblies to prepare the desired products for extraction. The DWPF process also uses an acid-based process. The resultant waste stream from all of these processes is acidic. Before transferring the waste material from the F and H Canyons and DWPF to the tank farms, sodium hydroxide is added adjusting the waste to a high 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, in small part, uranium, strontium and plutonium hydroxides. These solids are insoluble due to the chemical nature 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 waste tank. To maximize the space available in the waste tanks for storing additional waste, DOE's practice at SRS has been to use the tank farm evaporator systems to reduce the volume of the decanted supernate by concentrating the waste. [HLW-2002-00025]

During the evaporation process, the liquid salt waste is concentrated. After the concentrated salt waste is returned to the waste tank, the concentrated salt waste forms two distinct phases (collectively called salt waste): 1) concentrated supernate solution and 2) solid saltcake. 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 insoluble radioactive solids such as C-14, Sr-90 and Tc-99. When saltcake is dissolved and removed from the tank, these entrained insoluble solids eventually settle on the waste tank bottom adding to the sludge inventory. [SRR-LWP-2010-00040]

As of March 2010, as the result of the evaporation process, the combined total of more than 140,000,000 gallons of liquid waste originally received in FTF and HTF had been reduced to approximately 37,400,000 gallons. [SRR-LWP-2009-00001, SRR-LWP-2010-00040] 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 is not practical. The SRS no longer conducts weapons or NASA-related material production activities or the weapons-related spent nuclear fuel reprocessing that generated the original waste. The DOE has deactivated the Plutonium Uranium Extraction (PUREX) process in F Canyon (one of the two chemical separations canyon facilities) and is no longer generating radioactive liquid waste for storage in the FTF. H Canyon continues to generate radioactive liquid waste when performing stabilization missions such as dissolving non-irradiated fuel and recovering and blending HEU for non-defense related use.

Ongoing SRS operations continue to require the need for waste tank space. Most of the SRS Type III/IIIA waste storage tanks, tanks that meet current secondary containment and leak detection standards and have no prior leak sites, are already at or near full capacity. Projected available waste tank space is carefully tracked to ensure the tank farms do not become "water logged," a term meaning that so much of the useable compliant waste tank space has been filled that normal operations, waste removal and waste processing, cannot effectively continue. Substantial amounts of waste tank space are required to safely and effectively remove tank waste and prepare it for disposal. This includes the preparation of high-activity sludge waste for vitrification in DWPF.

The preparation of saltcake for disposition also requires significant waste 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 to properly dissolve the saltcake into liquid salt solution. [CBU-PIT-2005-00031]

Waste 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 closure activities. A portion of the available waste tank space must also be reserved as contingency space in the event a new waste tank leak occurs. The tank farms also receive new waste from the H Canyon and wash water from sludge washing.

2.1.10 F-Tank Farm

The F Area occupies approximately 364 acres. It includes FTF and the F-Canyon chemical separations facility[8], where nuclear materials were recovered using the PUREX process and the FTF. The FTF is located in the southern region of F Area and occupies approximately 22 acres. The FTF is an active waste storage facility used to store liquid radioactive waste generated primarily during prior operations of F Canyon. Figure 2.1-28 shows the FTF layout.

Figure 2.1-28: General Layout of FTF

[8] The final transfer of waste from F Canyon to the FTF occurred in August 2005.

The FTF consists of:

- 22 carbon steel waste tanks[9] (i.e., Tanks 1 through 8, Tanks 17 through 20, Tanks 25 through 28, Tank 33, Tank 34 and Tanks 44 through 47),
- ancillary structures:
  - the FTF transfer line system including approximately 45,000 linear feet of underground waste transfer lines,
  - three pump pits (FPP-1, FPP-2 and FPP-3),
  - three pump tanks (FPT-1, FPT-2 and FPT-3),
  - one 11,700 gallon catch tank,
  - two evaporator systems used to reduce waste volume (242-F and 242-16F),
  - the 242-F concentrate transfer system,
  - six diversion boxes (FDB-1, FDB-2, FDB-3, FDB-4, FDB-5 and FDB-6), and
  - eight valve boxes (valve boxes 1 through 5, valve boxes 28A and 28B and LDB-17 valve box).

This equipment is discussed separately as follows:

- waste tanks (Type I, III/IIIA and IV)
- ancillary structures

2.1.11 Waste Tanks

All 22 of the FTF waste tanks were built of carbon steel and reinforced concrete in three principal designs, Types I, III/IIIA and IV. The FTF does not contain any Type II waste tanks[10]. The waste tanks were numbered sequentially based on time of design and siting, and are not tank farm specific. For example, the original twelve tanks constructed at SRS were of Type I design and were numbered Tanks 1 through 8 in FTF and Tanks 9 through 12 in HTF. Table 2.1-6 summarizes the important design features.

Table 2.1-6: Waste Tank Nominal Capacities, Nominal Dimensions and Other Features

Type	Waste Tank Numbers	Diameter (ft)	Height (ft)	Capacity (gal)	Cooling Coils	Secondary Containment	Elevation (ft)[a]
I	1, 2, 3, 4, 5, 6, 7, 8	75.0	24.5	750,000	Yes	Yes	237 - 241
III/IIIA[b]	25, 26, 27, 28, 33, 34, 44, 45, 46, 47	85.0	33.0	1,300,000	Yes	Yes	244 - 246
IV	17, 18, 19, 20	85.0	34.5	1,300,000	No	No	227 - 228

a Approximate feet above MSL for the waste tank basemat.
b Tanks 33 and 34 are Type III waste tanks. The remaining tanks are Type IIIA.

The general design features of the waste tanks are summarized below, followed by brief descriptions and illustrations of the different waste tanks types.

2.1.11.1 General Tank Design Features

Each Type I, III and IIIA waste tank has a primary tank and a carbon steel secondary containment liner. The secondary liner for Type III and IIIA tanks extends the full height of the primary tank. The Type I tank secondary liner extends only 5 feet above the bottom of primary tank floor and is sometimes referred to as an “annulus pan.” Because the secondary liner is larger in diameter than the primary tank, an annular space exists between them. This waste tank annulus, which differs in size and capacity for each waste tank type, provides a collection point for any potential leakage from the primary tank, as well as a method for heating or cooling the primary tank wall in conjunction with the annulus ventilation system. Reinforced concrete vaults surround the secondary liner.

[9] Two tanks, Tank 17 and Tank 20, were operationally closed in 1997 and are not part of this Draft FTF 3116 Basis Document.
[10] Four Type II tanks were constructed in HTF; no Type II tanks were constructed in FTF.

The Type IV waste tank consists of a reinforced concrete vault surrounding a single carbon steel liner. The Type IV tanks do not have an annulus. The reinforced concrete vault provides both structural support and radiation shielding for all the waste tanks. The bottom part of the vault is called the basemat. Underneath the basemat of the Type I, III and IIIA tanks lies a concrete working slab. The Type IV waste tanks do not have a concrete working slab below the basemat.

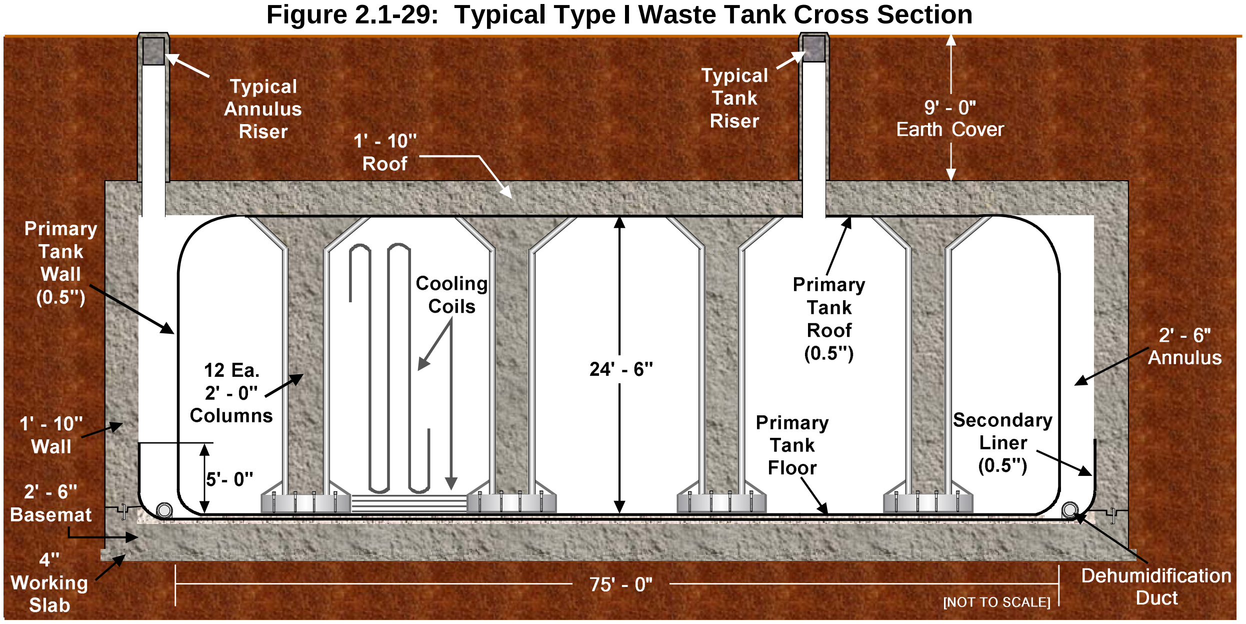
Chromate cooling water provides the primary cooling for the waste stored in the waste tanks, which flows through cooling coils located inside the primary waste tank. The cooling coils are installed in Type I and Type III/IIIA tanks and vary in design for each waste tank type. Type IV tanks do not contain cooling coils.

Risers provide access to the primary waste tank and annulus interiors. Risers are used primarily for inspections, level detection, dip samples and the installation of equipment such as annulus jets, dip tubes, thermocouples, conductivity probes, ventilation inlet and outlets, reel tapes, hydrogen monitors and waste removal equipment. Lead or concrete plugs are inserted in the riser opening if no equipment is installed. The riser structures are made of concrete and lined with carbon steel. Riser layout is dependent on the specific waste tank.

2.1.11.2 Type I Waste Tanks

Type I waste tanks (Tanks 1 through 8) were constructed in the early 1950s. These waste tanks are 75 feet in diameter and 24.5 feet in height with a nominal operating capacity of 750,000 gallons. The tank tops are approximately 9 feet below grade. All Type I tanks have a secondary carbon steel liner 80 feet in diameter and 5 feet high (2.5 feet annulus space). All Type I waste tanks have similarly configured vertical and horizontal cooling coils. A typical[11] Type I waste tank cross section is shown in Figure 2.1-29, waste tank concrete basemat construction is shown in Figure 2.1-30 and a waste tank primary tank and secondary liner construction is shown in Figure 2.1-31. Additional Type I tank details are provided in Section 3.0 of the FTF PA. [SRS-REG-2007-00002]

Figure 2.1-29: Typical Type I Waste Tank Cross Section



[11] The word "typical" as used throughout this section refers to representative design features of the system being described.

The primary tank is made of 0.5-inch thick carbon steel. The walls are joined to the roof and floor of the primary tank by curved knuckle plates made of the same material and are welded in place. The secondary liner is also made of 0.5-inch thick carbon steel. Transfer line penetrations allow 3-inch diameter inlet waste transfer lines to enter the primary waste tank near the top through the top knuckle. Each transfer line is enclosed in a 4-inch diameter carbon steel jacket pipe where it bridges the waste tank annulus. [SRS-REG-2007-00002]

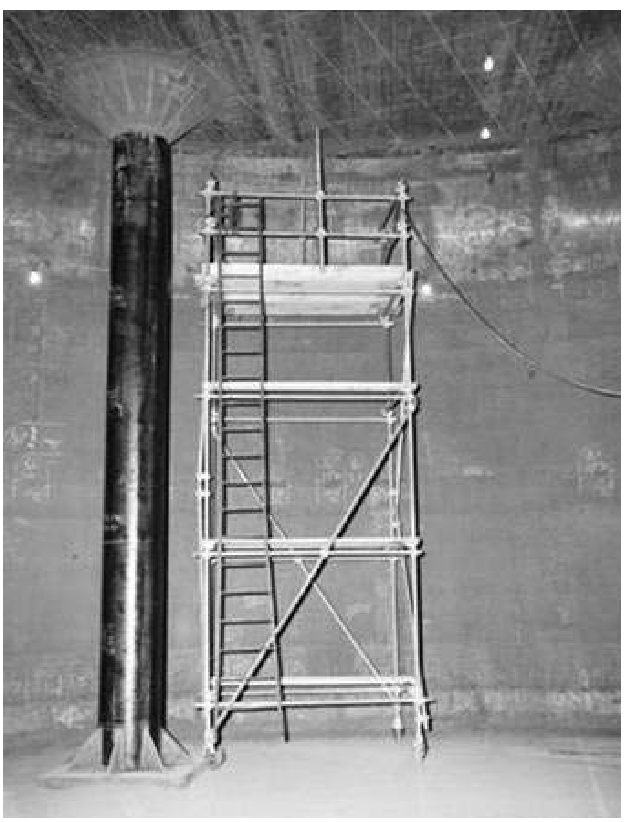
Figure 2.1-30: Type I Waste Tank Basemat Construction

Figure 2.1-31: Type I Waste Tank Primary and Secondary Liner Construction

The waste tank vault is constructed of 22-inch thick reinforced concrete with an inner diameter of 80 feet. Approximately nine feet of soil covers the vault roof as shown in Figure 2.1-29. [SRS-REG-2007-00002]

Each Type I tank has 12 concrete filled steel columns to support the roof. These columns have an outer diameter of 2 feet of 0.5-inch carbon steel pipe filled with concrete and welded to the top and bottom of the primary tank. A waste tank column at the time of construction is shown in Figure 2.1-32. [SRS-REG-2007-00002]

Figure 2.1-32: Typical Type I Waste Tank Column Support

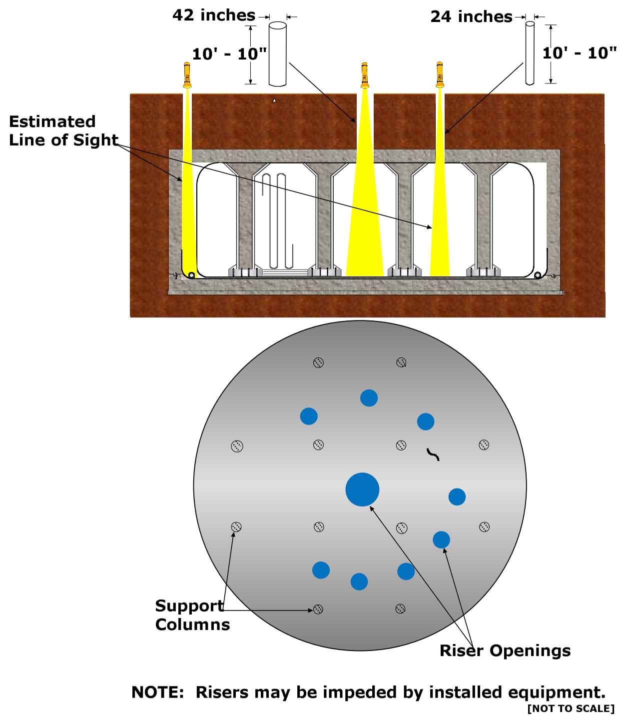


Cooling coils in Type I waste tanks are configured in both a horizontal and a vertical array, which creates obstacles to waste removal and other activities inside the waste tank (Figure 2.1-33). Each Type I waste tank contains 34 vertical cooling coils that are supported from the primary tank roof by hanger and guide rods, which are welded to the primary tank. All combined, the vertical coils consist of 604 vertical section 18.5 feet long with 604 loops (half circle with a 24-inch radius) that connect the vertical sections. Two horizontal cooling coils (upper and lower) extend across the bottom of the waste tanks and are supported by guide rods welded to the primary tank floor. The lower horizontal cooling coil is approximately 1 inch above the tank floor and the upper horizontal cooling coil is approximately 4 inches above the primary tank floor. The horizontal coils consist of 26 horizontal sections and 26 loops (half circle with a 24-inch radius) that connect the horizontal sections. In addition, there are supply pipes that connect the tank top cooling water system to the cooling coils. There are approximately 22,800 linear feet of 2-inch carbon steel pipe cooling coils in a Type I waste tank. [D116048, C-CLC-G-00364, D116001]

Visual and equipment manipulation access within the Type I waste tank is limited by the tank riser design configuration. Riser configuration, above the tank top, limits direct access to equipment and allows a limited view of the primary tank floor as shown in Figure 2.1-34. Additionally, the size of the access ports limits the manipulation of long-handled mechanical tools. Due to access port geometry, choices are limited as to the types of remote equipment that can be successfully deployed. Type I tanks have a 42-inch diameter center riser and eight 24-inch perimeter risers. Each riser is approximately 10-feet-10-inches in length (Figure 2.1-34). [W146625]

Figure 2.1-33: Typical Type I Waste Tank Cooling Coil Obstacles

Figure 2.1-34: Type I Waste Tank Access Area for Waste Removal Equipment Diagram



2.1.11.3 Type III/IIIA Waste Tanks

The Type III waste tanks were constructed in 1969 (Tank 33) and 1972 (Tank 34). The Type IIIA waste tanks were complete in 1978 (Tanks 25 through 28) and 1980 (Tanks 44 through 47). All Type III and IIIA waste tanks have full secondary containment by a secondary steel liner, which is 90 feet in diameter and 33 feet in height (2.5-foot annulus). [SRS-REG-2007-00002]

Each Type III waste tank has a minimum 3.5-foot thick reinforced concrete basemat placed on top of a 4-inch concrete working slab. The Type IIIA waste tanks have a minimum 3-foot-7-inch thick reinforced concrete basemat placed on top of a 4-inch concrete working slab. Both Type III and IIIA waste tanks have a center drop panel that varies the basemat thickness for each tank type. Additional Type III and IIIA waste tank details are available in Section 3.0 of the FTF PA. [SRS-REG-2007-00002]

A typical FTF Type III waste tank cross section is shown in Figure 2.1-35, typical Type IIIA waste tank cross section is shown in Figure 2.1-36 and waste tank basemat construction is shown in Figure 2.1-37.

Figure 2.1-35: Typical FTF Type III Waste Tank Cross Section

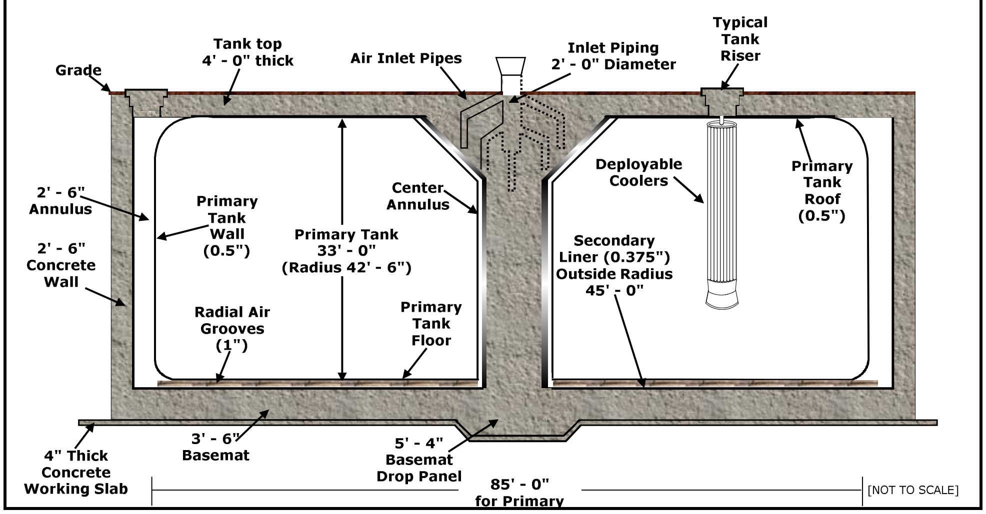


Figure 2.1-36: Typical Type IIIA Waste Tank Cross Section

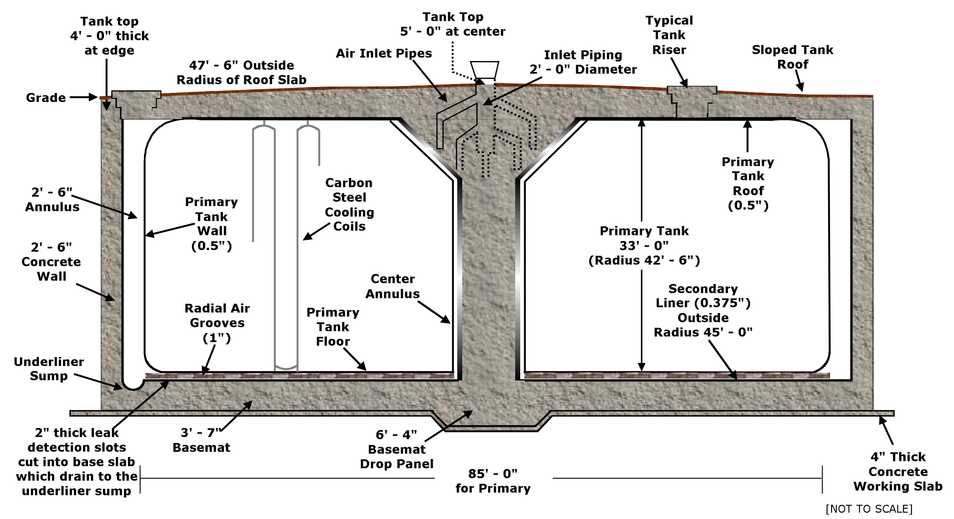
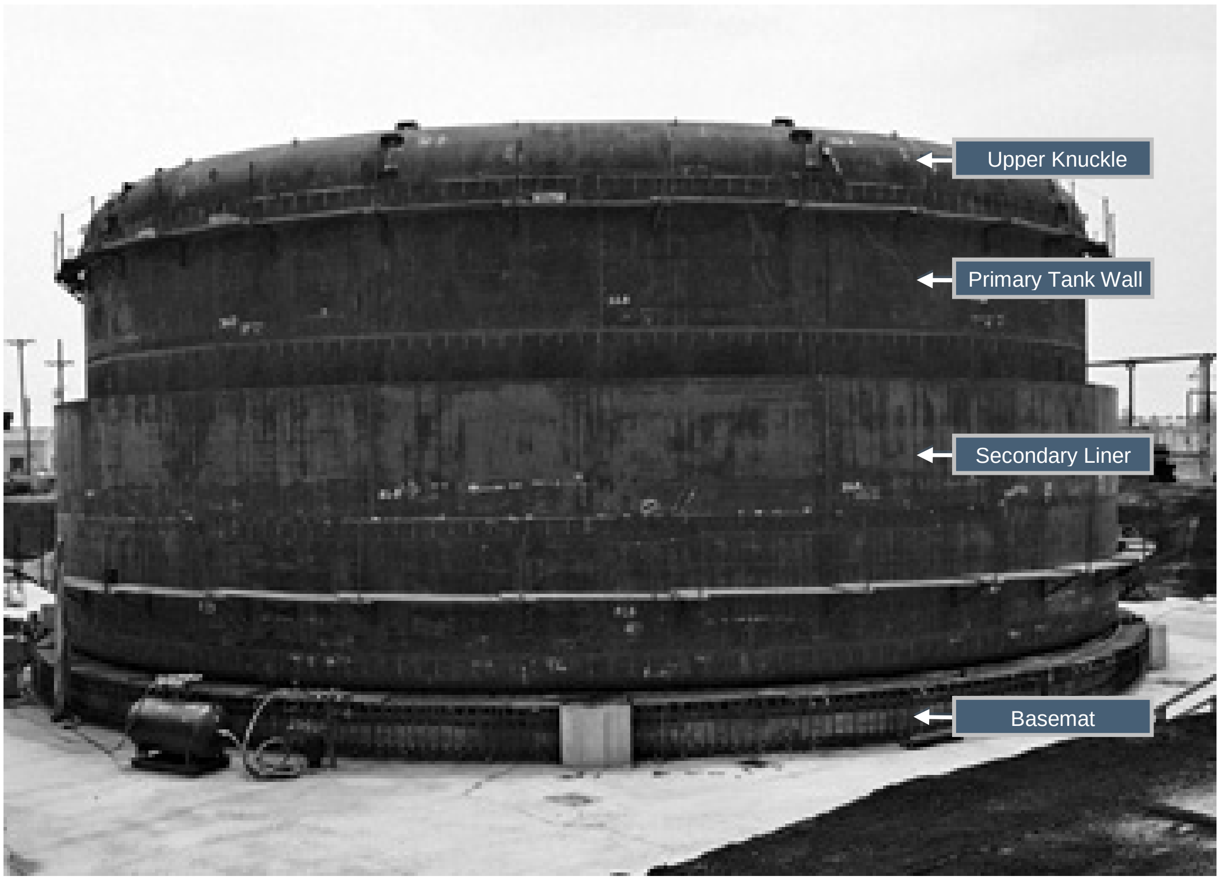


Figure 2.1-37: Typical Type III/IIIA Waste Tank Basemat Construction

The Type III/IIIA waste tank primary tanks are made of carbon steel. The walls are joined to the roof and floor plates by curved knuckle plates made of the same material. The primary tank and secondary liner for a Type IIIA tank late in the construction phase are shown in Figure 2.1-38. The Type III/IIIA waste tank primary tanks were fully stress-relieved by heating after fabrication. The waste tank primary tank and secondary liner plate data are summarized in Table 2.1-7. [SRS-REG-2007-00002]

Figure 2.1-38: Typical Type IIIA Waste Tank Primary Tank and Secondary Liner - Late Construction (Vault Wall Not Constructed)



Both Type III and IIIA waste tanks contain multiple penetrations through a carbon steel primary tank (e.g., 2-inch diameter pipe in 3-inch diameter sleeves) near the top of the waste tank for transfer lines into and out of the tank. [SRS-REG-2007-00002]

The Type III/IIIA primary waste tanks are completely enclosed in a concrete vault. The vault roof is at least 48 inches thick and the walls are 30 inches thick; therefore, there is no earthen cover for shielding on top of these waste tanks. The Type III/IIIA tanks were constructed below grade with native soil backfilling around and between the tanks.

Type III/IIIA waste tanks have both a center and outer annulus. The center annulus is formed between the primary waste tank wall and the roof support column. This design allows for ventilation airflow to the underside of

Table 2.1-7: Type III/IIIA Waste Tanks Primary Tank and Secondary Liner Plate Data

Location	Thickness (in)
Primary tank roof	0.5
Primary tank floor	0.5
Upper knuckle	0.5
Secondary liner - upper band	0.5
Secondary liner - middle band	0.625
Secondary liner - lower band	0.750
Primary tank wall - upper band	0.5
Primary tank wall - lower band	0.625
Lower knuckle - secondary	0.875
Lower knuckle - primary	0.625

the primary tank floor and then out to the outer annulus through the radial air grooves, as shown in Figure 2.1-39.

Type IIIA waste tank tops are 5 feet thick at the waste tank center, 4 feet thick at the edge and sloped to allow rainwater drainage. In the Type III/IIIA designs, the primary waste tank roof is supported by a steel-lined center support column that is integrated into the concrete basemat (Figure 2.1-35 and Figure 2.1-36). Ventilation systems are embedded in the center column. Air flows through the column, into the center annulus, through the radial air grooves and exits through the outer annulus. [SRS-REG-2007-00002]

Type III waste tanks cooling coil piping consists of deployable coolers (conical coolers and cylindrical coolers as shown in Figure 2.1-40). These coolers were inserted through the risers in the closed position and deployed (opened) once inside the tank. The coolers are supported by the tank top. The bottoms of the waste tanks are cooled by the air passing through the annulus and radial air grooves. Type IIIA tanks have permanently installed cooling coils as shown in Figure 2.1-41. These are made of 2-inch diameter carbon steel pipe. [M-CLC-H-02820, SRS-REG-2007-00002]

Figure 2.1-39: Typical Type III/IIIA Waste Tank Radial Air Grooves

Figure 2.1-40: Typical Type III Waste Tank Deployable Cooling Coils

The tank riser design configuration restricts visual and equipment access into Type III/IIIA waste tanks. These waste tanks have 37 to 42 risers, ranging from 8 to 52 inches in diameter, and are approximately 4.5 feet in length as shown in Figure 2.1-42 and Figure 2.1-43. [W704010, W449795]

Figure 2.1-41: Typical Type IIIA Waste Tank Cooling Coils During Construction

Figure 2.1-42: Typical Type III Waste Tank Access Area for Waste Removal Equipment Diagram

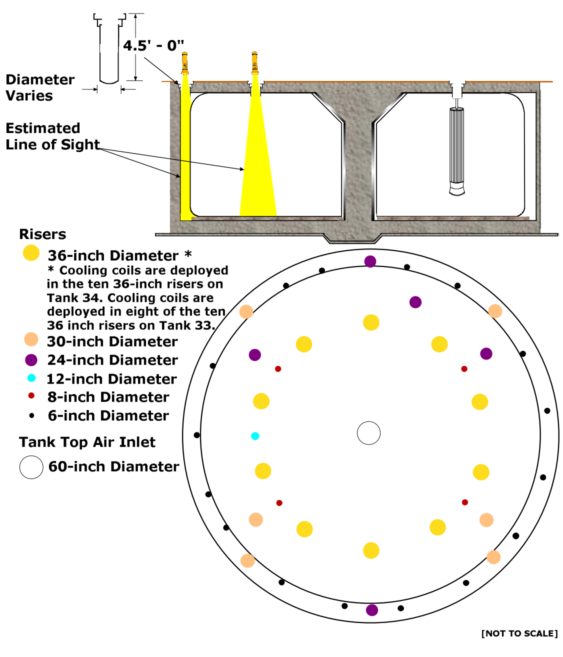
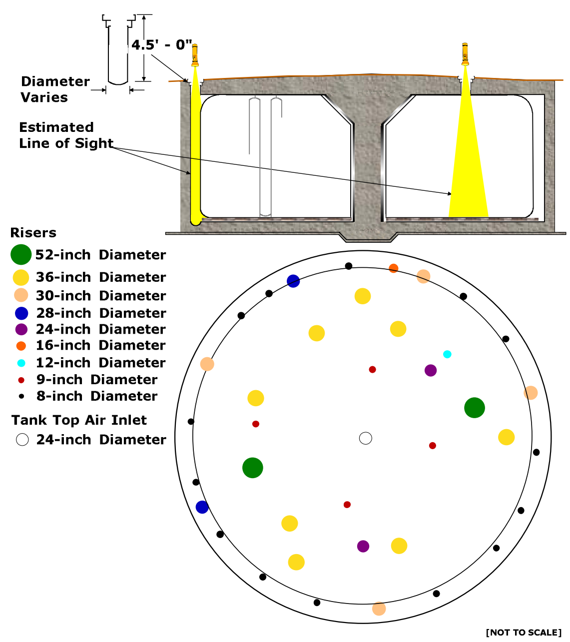


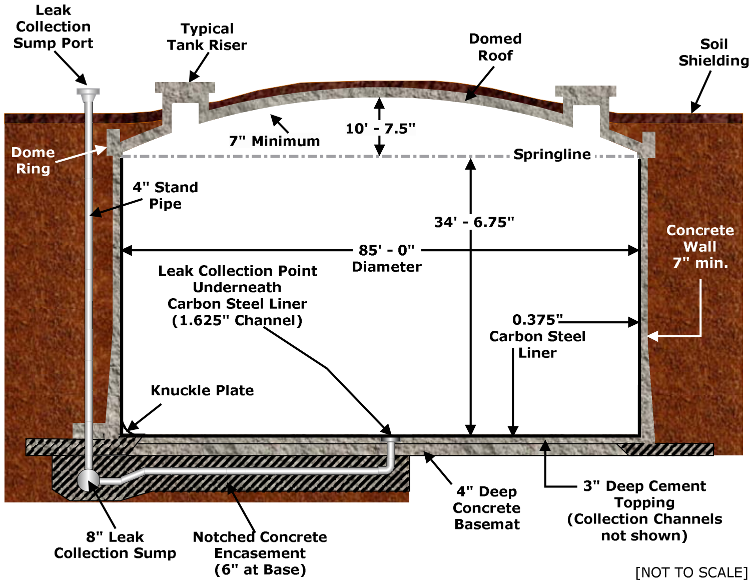
Figure 2.1-43: Typical Type IIIA Waste Tank Access Area for Waste Removal Equipment Diagram



2.1.11.4 Type IV Waste Tanks

The Type IV waste tanks were constructed in the late 1950s. Type IV tanks are 85 feet in diameter and approximately 34 feet in height at the side wall with a nominal operating capacity of 1,300,000 gallons. Type IV tanks do not have a secondary liner or cooling coils. These waste tanks have a single carbon steel liner with a self-supporting, reinforced concrete dome roof. A typical FTF Type IV waste tank cross section is shown in Figure 2.1-44. Additional Type IV waste tank details are available in Section 3.0 of the FTF PA. [SRS-REG-2007-00002]

Figure 2.1-44: Typical FTF Type IV Waste Tank Cross Section



The Type IV waste tank primary liner is an open-top carbon steel cylinder. The sides and bottom are formed of 0.375-inch plates with 0.4375-inch thick knuckle plates, as shown in Figure 2.1-44. The primary waste tank liner is internally reinforced with three, circumferential, 4-inch carbon steel stiffener angles as shown in Figure 2.1-45. The primary liner is anchored to the concrete. Each waste tank has side wall penetrations through the vault and waste tank liner located just below the dome for transfer lines into and out of the tank. [SRS-REG-2007-00002]

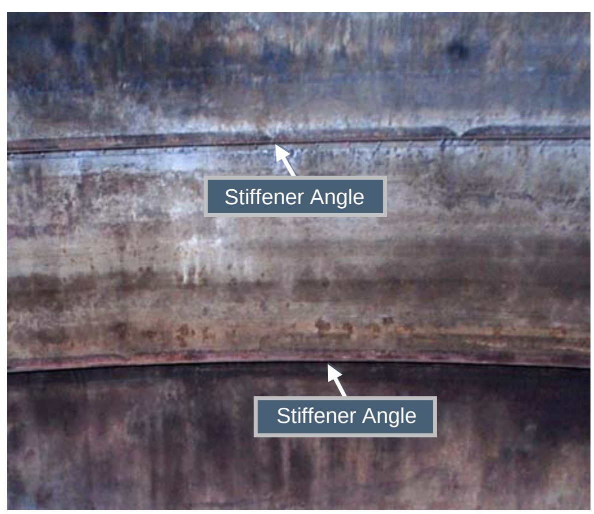
Each Type IV waste tank is enclosed in a concrete vault. Each vault was constructed in layers using a "shotcrete" technique. No secondary containment structure or annulus exists with this design.

The concrete vault walls are cylindrical with an inside diameter of 85 feet and surmounted by a dome ring 33 feet high. The core wall is 7 inches at the top and 11 inches at the bottom. This core wall was prestressed with 163 steel bands that remained in place and were covered with sprayed-on concrete (Figure 2.1-46). The wall thickness, including the bands and concrete cover, is 10 inches at the top and 15 inches at the bottom. [SRS-REG-2007-00002]

Type IV concrete vault sidewalls are surrounded by three layers of backfill. Bags of vermiculite were placed around the vault in brick-like pattern, covered with a special manually-compacted soil and topped with test controlled, compacted soil fill.

The dome roof is 7 inches to 10 inches thick, with the greater thickness near the risers, and is reinforced throughout with steel bars. The dome has an internal curvature radius of 90-foot-4-inches and a rise of 10-foot-7.5-inches above the spring line. The Type IV waste tank dome and risers are shown, near the end of the concrete construction phase, in Figure 2.1-47. [SRS-REG-2007-00002]

Figure 2.1-45: Typical Type IV Waste Tank Stiffener Angles



The concrete roof of a Type IV waste tank is not lined with carbon steel on the inside. The center riser has an inner diameter of 10 feet and is approximately 4 feet in length. The six smaller risers each have a inner diameter of 2 feet and are approximately 5 feet in length. The tank riser design configuration provides limited access to the tank interior. New tank top openings have been constructed in Type IV tanks to support waste removal and sampling activities. [SRS-REG-2007-00002]

Figure 2.1-46: Typical Type IV Waste Tank Wall Steel Bands During Construction

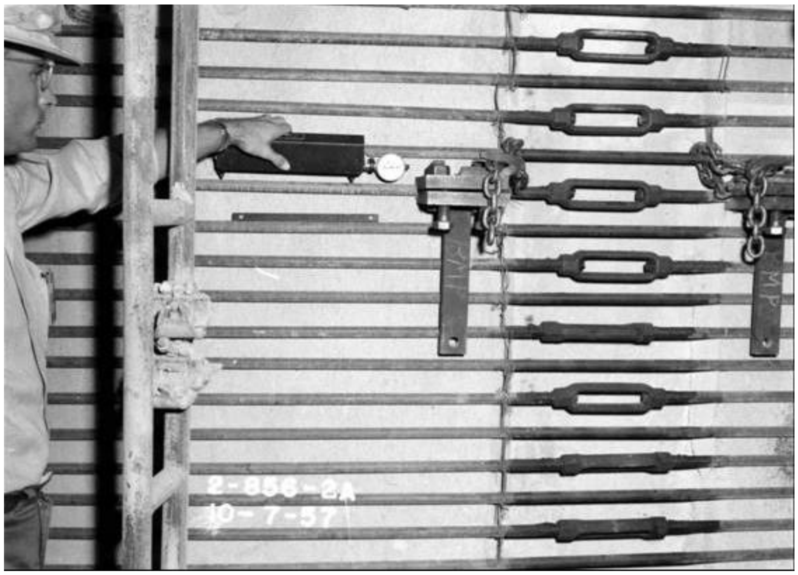


Figure 2.1-47: Typical Type IV Waste Tank Domes and Risers During Construction

2.1.12 Ancillary Structures

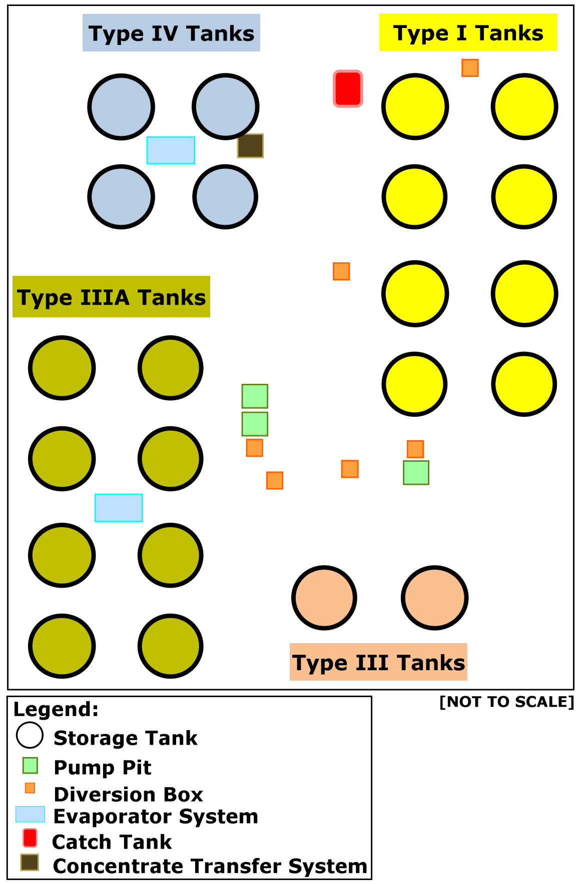
In addition to the waste tanks, FTF contains ancillary structures with a residual inventory that must be accounted for as part of FTF closure. These ancillary structures include buried pipe (transfer lines), pump tanks and evaporators, all of which have been in contact with liquid waste. The ancillary structures are used in the FTF to transfer waste (e.g., transfer lines, pump tanks) and reduce waste volume through evaporation. The amount of contamination on these components depends on such factors as the service life of the component, its materials of construction and the contamination medium in contact with the component. Figure 2.1-48 identifies locations of FTF-specific ancillary structures.

A description of the FTF ancillary structures is provided as follows:

- the FTF transfer line system (approximately 45,000 linear feet of underground waste transfer lines), including transfer line jackets, leak detection boxes, modified leak detection boxes and the Type I tank transfer line encasements,
- the FTF pump tanks (FPT-1, FPT-2 and FPT-3), FTF pump pits (FPP-1, FPP-2 and FPP-3) and the FTF catch tank,
- the 242-F Evaporator System, including the evaporator cell and support tanks (e.g., mercury collection tank, cesium removal column feed tank, concentrate transfer system tank and two overheads tanks),
- the 242-16F Evaporator System, including the evaporator cell and support tanks (e.g., mercury collection tank, cesium removal column feed tank and two overheads tanks),
- the FTF diversion boxes (FDB-1, FDB-2, FDB-3, FDB-4, FDB-5 and FDB-6), and
- the FTF valve boxes (valve boxes 1 through 5, valve boxes 28A and 28B and LDB-17 valve box).

[SRS-REG-2007-00002]

Figure 2.1-48: General Layout of FTF



2.1.12.1 Waste Transfer Lines

There are over 45,000 linear feet of waste transfer line in FTF, with the line segments ranging from a few feet to over 4,000 feet in length. The FTF waste transfer lines are typically constructed of a stainless steel primary core pipe and are normally located below ground. Those lines that are above ground or near the surface are shielded to minimize radiation exposure to personnel. Most primary waste transfer lines have some type of secondary containments. The majority of primary transfer lines are surrounded by another pipe (jacket) constructed of carbon steel, stainless steel or cement-asbestos. These jackets drain to leak detection boxes, modified leak detection boxes or to another primary or secondary containment such as a waste tank. A few primary transfer lines are located inside a covered, concrete encasement. [SRS-REG-2007-00002]

Waste transfer lines are typically sloped to be self-draining and, where a pipe transitions from one size to another, the bottom of the pipe is generally aligned to allow for draining. The line segments are supported using rod or disk type core pipe spacers, core pipe supports, jacket supports, jacket guides or other approved methods. Typically, core pipe spacers and supports are of stainless steel and welded to the core pipe and jacket, while the jacket supports and guides are of stainless steel with a concrete support. Figure 2.1-49 shows waste transfer lines during construction. [SRS-REG-2007-00002]

Figure 2.1-49: Typical Waste Tank Transfer Line During Construction

2.1.12.2 Pump Pits, Pump Tanks and Catch Tanks

The pump pits are shielded reinforced concrete structures located below grade at the low points of transfer lines and are lined with stainless steel. The pump pit walls are approximately 2 to 3 feet thick, sloped floors are approximately 3 feet thick and cell covers are concrete slabs approximately 2 to 3 feet thick. All FTF pump pits house a pump tank with the pump pits providing secondary containment for pump tanks. Figure 2.1-48 shows the location of the three pump pits in FTF. [SRS-REG-2007-00002]

The pump pits are often constructed in conjunction with a diversion box (diversion box details are discussed later in this section). A typical diversion box/pump pit configuration is depicted in Figure 2.1-50. Figure 2.1-51 is a photograph of a pump tank.

Figure 2.1-50: Typical Diversion Box and Pump Pit Design

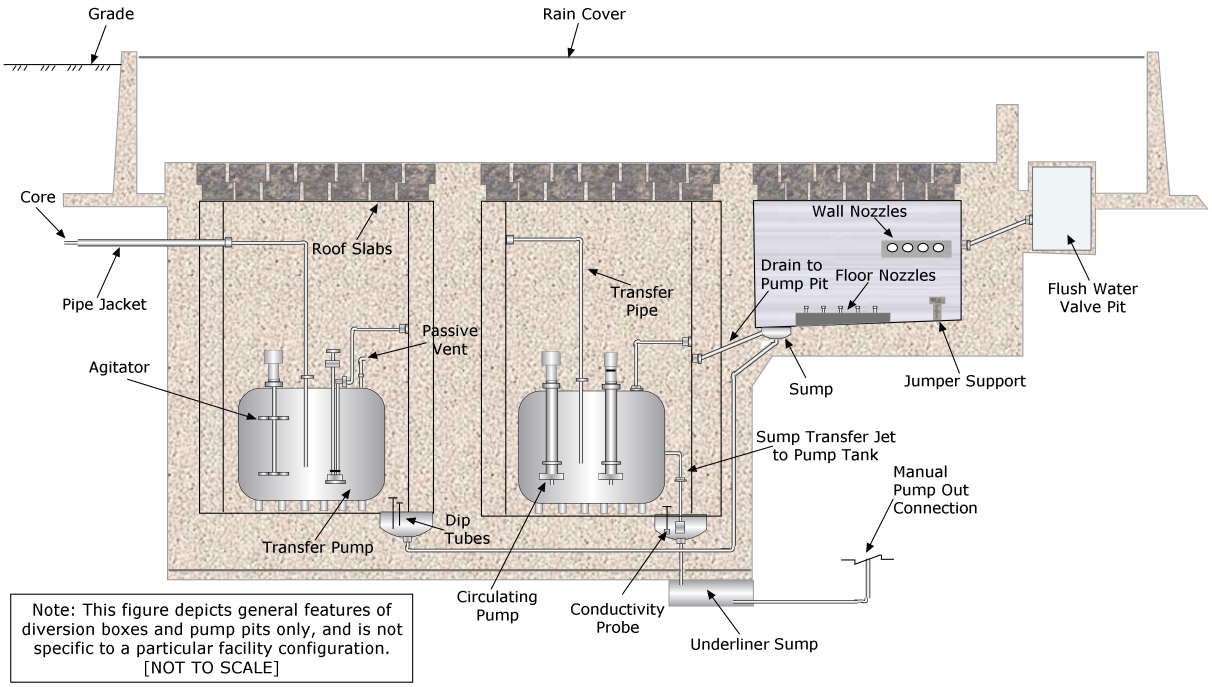


Figure 2.1-51: Typical Pump Tank Top View During Construction

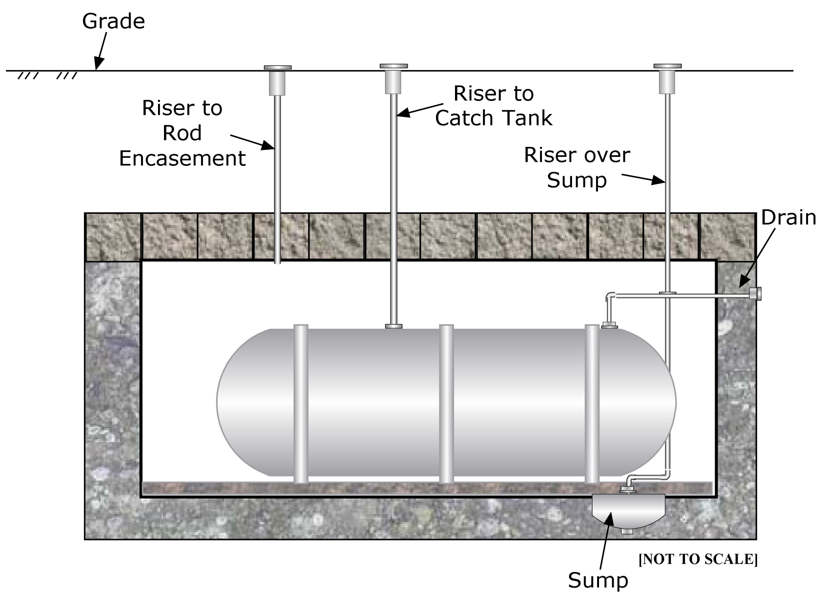
The following is a description of pump pit and pump tank features.

- **FPP-1/FPT-1:** The FPP-1 has a volume of approximately 33,000 gallons. The FPT-1 has a volume of approximately 7,200 gallons and currently serves as the inter-area pump tank.
- **FPP-2/FPT-2 and FPP-3/FPT-3:** These pump pits have volumes of approximately 37,400 gallons. The FPP-3 is connected through a pipe chase to the adjacent FPP-2. The FPP-2 and FPP-3 underliners drain to the FPP-2 underliner sump. The FPT-2 and FPT-3 are 7,200 gallon tanks that previously received waste transfers from the F-Canyon Facility.

[SRS-REG-2007-00002]

There is a single catch tank in FTF designed to collect drainage from FDB-1 and the Type I waste tank transfer line encasements. The stainless steel catch tank capacity is approximately 11,700 gallons and is located in an underground reinforced concrete cell. The catch tank encasement has walls over 3 feet thick and is built on a 4-inch thick concrete pad (Figure 2.1-52). [SRS-REG-2007-00002]

Figure 2.1-52: Typical Catch Tank Cross Section



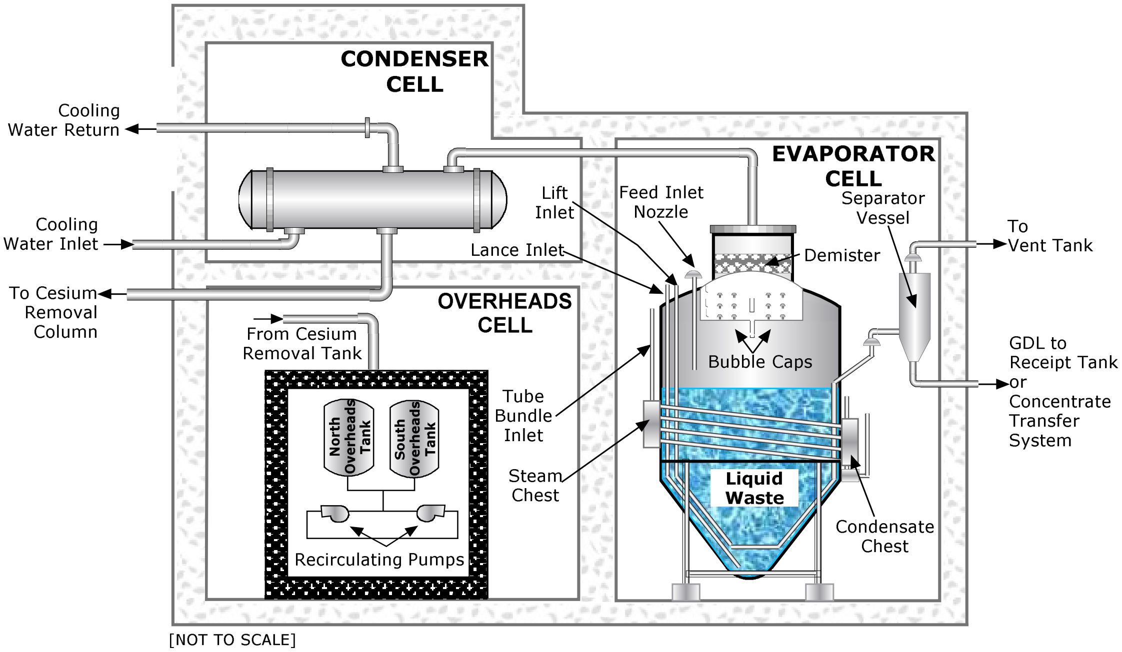
2.1.12.3 Evaporator Systems

There are two evaporator systems in the FTF, the 242-F Evaporator System and the 242-16F Evaporator System. Evaporators are used to reduce the volume of liquid radioactive waste within FTF by driving off a portion of the water in the waste. The evaporator systems are principally comprised of the evaporator, the overheads system and the condenser. The 242-F Evaporator System also included the 242-3F concentrate transfer system, which was used to distribute evaporator concentrated material throughout FTF. [SRS-REG-2007-00002]

2.1.12.3.1 242-F Evaporator System

The 242-F Evaporator Facility was constructed and placed into service in 1960 and was removed from service in 1988. The 242-F evaporator cell is a cuboid with a 16 feet x 15 feet base and a height of 25 feet. The cell includes a floor sump measuring 2 feet x 2 feet x 2.5 inches deep. The cell provided containment for the evaporator and served as shielding for personnel protection. The cell includes a stainless steel liner. [SRS-REG-2007-00002] Figure 2.1-53 provides a schematic of the 242-F Evaporator System.

Figure 2.1-53: 242-F Evaporator System Schematic



- **242-F Evaporator Vessel:** The evaporator vessel, located inside the 242-F evaporator cell, is a stainless steel, cylindrical vessel with a cone bottom. The cylindrical portion is 8 feet in diameter with a height of 8-foot-9.75-inches. The cone has a maximum diameter of 8 feet and a height of 5-feet-11-inches. The evaporator was used to concentrate liquid by evaporating water from the waste to reduce waste volumes.
- **242-F Overheads System:** The receiver cell is cuboid with a 15 feet x 9 feet x 8-inch base and a height of 6-foot-4-inches. The receiver cell includes a floor sump with a 1.5 feet x 1.5 feet base and a depth of 1.5 feet. The North and South overheads tanks, located inside the 242-F receiver cell, are each cylindrical, stainless steel vessels having a diameter of 6 feet and a height of 6 feet. The overheads tanks functioned as receipt tanks for liquids condensed from evaporator vapors. The 242-F condenser is a stainless steel cylindrical vessel with an outer diameter of 18 inches and a height of 9-foot-10.25-inches. The condenser functioned to condense evaporator vapors into liquid, which was drained to the North and South overheads tanks.
- **242-F Concentrate Transfer System:** The concentrate from the 242-F evaporator was steam lifted to the concentrate transfer system. The concentrate transfer system draw-off pump circulated the concentrate continuously through a loop line to the concentrate receipt tanks. Various tanks have served as concentrate receipt tanks over the service life of the 242-F Evaporator System. When concentrate reached a predetermined level in the concentrate transfer system waste tank, a drop valve opened adding the concentrate to a receipt tank.

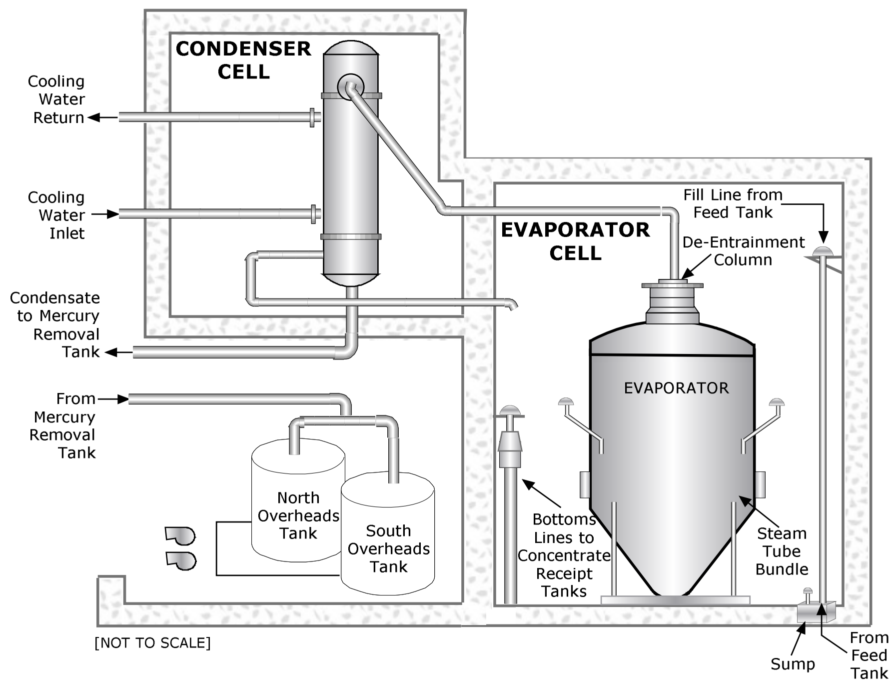
  The concentrate transfer system pit is cuboid with a 12 feet x 12 feet base and a height of 21 feet. The pit includes a floor sump with a 1.5 feet x 1.5 feet base and a depth of 1.5 feet. The concentrate transfer system pit provided containment for the concentrate transfer system waste tank and featured a stainless steel liner. Cell covers provide personnel protection. The stainless steel concentrate transfer system tank, located inside the 242-3F concentrate transfer system pit, is a cylindrical vessel with a diameter of 8 feet and a height of 8-foot-4-inches.

[SRS-REG-2007-00002]

2.1.12.3.2 242-16F Evaporator System

The 242-16F Evaporator Facility was constructed in 1980 and continues to operate. The 242-16F evaporator facilities are arranged into three cells and a gang valve house. The evaporator cell contains the evaporator, the condenser cell contains the condenser and a diked overheads cell contains overheads system components other than the condenser. Figure 2.1-54 provides a schematic of the 242-16F Evaporator System. [SRS-REG-2007-00002]

Figure 2.1-54: 242-16F Evaporator System Schematic



The evaporator cell is a cuboid with a 16 feet x 16 feet base and a height of 25 feet, with walls constructed of stainless steel-lined, grooved concrete that is 3.5 feet thick and a roof 2 feet thick, composed of concrete slab sections and a sloped, galvanized steel rain cover. The concrete slab sections and the rain cover have access ports for valves and viewing. The evaporator cell is stainless steel-lined for collecting leakage from equipment inside the evaporator or condenser cells, leakage from the lift/lance/evaporator cell sump gang valve vent header and liquid from cell spray operations. An evaporator underliner sump collects any leakage through the concrete or stainless steel liner. [SRS-REG-2007-00002]

The condenser cell is 9 feet x 10 feet x 14 feet high with walls constructed of concrete and has a roof composed of concrete slab sections, and a sloped, galvanized steel rain cover. The concrete slab sections and the rain cover have access ports for viewing. The condenser cell contains a stainless steel liner pan on a sloped floor. The condenser cell has an opening to the evaporator cell. The de-entrainment column piping enters the condenser cell through this opening, which also permits airflow to the evaporator cell. [SRS-REG-2007-00002]

The overheads cell (which is open to the environment) contains the mercury removal tank, cesium removal column feed tank, two cesium removal column pumps and two overheads tanks, an overheads tank sample system, filters for removing zeolite from condensate, cesium removal column gamma monitors, cesium removal column charging jet, and two overheads pumps. [SRS-REG-2007-00002]

- **242-16F Evaporator Vessel:** The evaporator vessel has a capacity of approximately 4,400 gallons. The insulated vessel is 8 feet in diameter with a height of 16.5 feet, and a cone-shaped bottom. The vessel is constructed of 0.5-inch stainless steel. There are multiple evaporator vessel service/equipment lines installed in, or penetrating, the vessel, including the feed inlet nozzle, steam tube bundle, warming coil, lift lines, de-entrainment column, lance lines and the seal pot.
- **242-16F Overheads System:** The overheads system includes the condenser, mercury removal tank, cesium removal column feed tank, two cesium removal column pumps, two overheads tanks and two overhead pumps. The condenser is a vertical, single-pass, counter-flow tube and shell type heat exchanger located in the condenser cell. The mercury removal tank receives

condensed overheads from the condenser. When full, the stainless steel tank overflows to the cesium removal column feed tank, permitting the heavier mercury to settle out and remain in the tank. The tank vents to the condenser cell, which vents and drains to the evaporator cell. The path from the evaporator vessel to the overheads tanks travels through a stainless steel cesium removal column feed tank. The overheads tanks are constructed of stainless steel and are 6 feet in diameter with a height of 6 feet. A 2-inch diameter overheads tanks overflow line is routed to the cell sump.

[SRS-REG-2007-00002]

2.1.12.4 Diversion Boxes

The diversion boxes are shielded, reinforced concrete structures containing transfer line nozzles to which jumpers are connected to direct waste transfers to the desired location. The diversion boxes are often constructed in conjunction with a pump pit.

Diversion boxes are located below grade and are either stainless steel-lined or sealed with waterproofing compounds, as described below, to prevent ground contamination. The walls are approximately 2 to 3 feet thick and sloped floors are approximately 3 feet thick. The diversion boxes have concrete slab-type cell covers, approximately 2 to 3 feet thick that must be removed for changing jumper alignment. [SRS-REG-2007-00002]

The following is a description the features associated with the FTF diversion boxes: [SRS-REG-2007-00002]

- **FDB-1** has a total volume of approximately 84,000 gallons. The FDB-1 drains to the F-Area catch tank. The interior of the diversion box is painted with chemical-resistant paint and sealer. FDB-1 has no sump.
- **FDB-2** has a total volume of approximately 8,300 gallons. The FDB-2 is ventilated through a vent duct to the adjacent FTF pump pit (FPP-1). Leakage into the FDB-2 sump drains into FPP-1 through an opening in the sump. The FDB-2 is the connection point from FTF to the Inter-Area Line.
- **FDB-3** has a total volume of approximately 1,900 gallons. The diversion box is stainless steel-lined and has a sump that gravity drains to FDB-2.
- **FDB-4** has a total volume of approximately 21,700 gallons with an underliner. The FDB-4 is ventilated through openings (vent and drain slots and through the pipe chase) to the adjacent FPP-2 and FPP-3. The diversion box is stainless steel-lined and has a sump that gravity drains into FPP-2.
- **FDB-5** has a total volume of approximately 18,600 gallons. The FDB-5 was installed to connect the concentrate transfer system with Tanks 25 through 28 or Tanks 33 and 34. The diversion box is stainless steel-lined and has a sump that gravity drains to FPP-2.
- **FDB-6** has a total volume of approximately 22,000 gallons. The FDB-6 was installed so that the 242-F evaporator could be fed from Tank 26 as well as from Tank 7. The diversion box is stainless steel-lined and has a sump that gravity drains to FPP-3.

[SRS-REG-2007-00002]

2.1.12.5 Transfer Valve Boxes

Transfer valve boxes facilitate specific waste transfers that are conducted frequently. The valves are generally manual ball valves in removable jumpers with flush water connections on the transfer lines. For FTF valve boxes, leakage collects in the valve box and drains back to the associated waste tank, diversion box or leak detection box. Valve boxes are generally located adjacent to the waste tanks they serve. [SRS-REG-2007-00002]

- **Valve boxes 1 through 5** facilitate transfers between FDB-2, FPP-1 and Tanks 4 through 6 and Tank 8. Valve boxes 1 through 4 have approximate volumes of 330 gallons, while valve box 5 is slightly larger at approximately 380 gallons.
- **LDB-17 valve box** has an approximate volume of 140 gallons and facilitates Tank 8 transfers. Leakage in LDB-17 drains back to a riser on Tank 8.

- **Valve boxes 28A and 28B** - Valve box 28A is located on the top of Tank 28 and has an approximate volume of 450 gallons. Valve box 28B is adjacent to Tank 28 and has an approximate volume of 365 gallons. Leakage to valve box 28B drains to leak detection boxes and the leak detection boxes drain header. Valve box 28A drains back to Tank 28. Valve boxes 28A and 28B, together, are commonly referred to as valve box 28.

[SRS-REG-2007-00002]

2.1.12.6 Other Ancillary Structures

The leak detection boxes provide for the collection and detection of leakage from a transfer line. Drain piping is run from a transfer line jacket to a leak detection box. The leak detection boxes typically have conductivity probe leak detection and drain and overflow plugs. Drain piping for the leak detection boxes is provided so that leaks are diverted to a diversion box or pump pit. No leakage into transfer line secondary containment (e.g., transfer line jacket and leak detection box) due to primary line failure has been detected. [SRS-REG-2007-00002]

The modified leak detection boxes serve the same purpose as the leak detection boxes but are larger and are installed at low points that cannot be gravity drained to a collection point. In addition to a conductivity probe, modified leak detection boxes also include a vent line to a diversion box or pump pit, an above-ground pressure gage to monitor for potential over-pressurization and a smear/cleanout pipe for measuring levels and manual pump-out of leakage into the box. [SRS-REG-2007-00002]

2.2 F-Tank Farm Wastes

Most of the waste managed in FTF originated in the F-Canyon Separations Facility[12], which made use of the PUREX separations process. Typically in this process, uranium and plutonium were extracted from aluminum-clad, depleted uranium slugs (known as targets) that were irradiated in the site's nuclear production reactors.

Before transfer of the waste from the F Canyon to the tank farms, sodium hydroxide was added to adjust the waste to a high alkaline state to prevent corrosion of the carbon steel waste tanks. This chemical adjustment resulted in the precipitation of solids. These solids settled 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, uranium and plutonium hydroxides. These solids are insoluble due to the chemical conditions of the solution. After settling of the solids occurred, the liquid salt waste solution (supernate) above this sludge layer was transferred out of the waste tanks. To maximize the space available in the waste tanks for storing additional waste, DOE's 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 waste tanks allows additional receipt of waste from the processing facilities. [HLW-2002-00025]

During the evaporation process, the liquid salt waste is concentrated. After the concentrated salt waste is returned to the waste tank, the concentrated salt waste forms two distinct phases (collectively called salt waste): 1) concentrated supernate solution and 2) solid saltcake. 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 and Tc-99. When saltcake is dissolved and removed from the tank, these entrained sludge particles eventually settle on the waste tank bottom adding to the sludge inventory. [SRR-LWP-2010-00040]

The solids on the waste tank bottom 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 waste tank.

The waste stored in FTF waste tanks is comprised of a combination of sludge, supernate and saltcake.

[12] The final transfer of waste from F Canyon to the FTF occurred in August 2005.

2.2.1 Sources of the Waste

This section describes the origin of the wastes managed in FTF and explains how these wastes are managed.

2.2.1.1 The PUREX Process

This subsection briefly describes the PUREX process that generated most of the FTF waste. [DPSPU 77-11-1]

As discussed previously, most of the waste managed in FTF originated in the F-Canyon chemical separations facility, which made use of the PUREX separations process. The first major step in the F-Canyon PUREX process involved dissolving the aluminum cladding. After this was done, the targets were dissolved in nitric acid. The resulting solution contained uranium (principally U-238) and plutonium, along with fission products such as Cs-137 and Sr-90.

This dissolver solution was processed to remove silica solids and the clarified solution was then fed to the first solvent extraction cycle. In this cycle, uranium and plutonium were separated from the fission products by being extracted into the solvent. Approximately 95% of the fission products remained in the aqueous phase, with the balance of the fission products being contained in the solvent along with the uranium and plutonium.

The solvent product stream underwent additional processing cycles to separate the uranium and plutonium and to purify these materials. The aqueous waste stream containing 95% of the fission products was evaporated to reduce the waste volume and recover the nitric acid. It was then processed through a primary recovery column where plutonium and neptunium were recovered. The waste stream was then neutralized and transferred to one of the waste tanks.

This neutralized waste stream is known as high-heat waste. Another waste stream from the PUREX process resulted from stripping and recovery of nitric acid from first and second uranium purification cycles. This waste stream is known as low-heat waste. These two waste streams made up the bulk of the waste managed in FTF. [DPSPU 77-11-1]

2.2.1.2 Evaporator Products

Figure 2.2-1: Saltcake at Bottom of Waste Tank

The two FTF evaporators concentrated the liquid waste generated in the PUREX process to remove excess water, a process which generated concentrated supernate and evaporator overheads. As discussed previously, the saltcake is precipitated salt waste from the concentrated supernate and is comprised principally of inert salts such as nitrites and nitrates. Figure 2.2-1 shows the saltcake at the bottom of a waste tank. The concentrated supernate is in an alkaline solution that is relatively high in radioactivity. Overheads are the excess water removed from the waste stream by the evaporation process. Overheads are sampled and, depending on the sample results, are sent to the Effluent Treatment Facility (ETF) or, if cesium levels are high, returned to a waste tank to repeat the evaporation process. Additional cesium removal is also part of the process at ETF.

2.2.1.3 Zeolite Resin

Some waste tanks contain zeolite resin from ion exchange columns, which are also known as cesium removal columns. Tanks 19, 25 and 27 were equipped with these columns to remove cesium from the 242-F and 242-16F evaporator overheads waste streams, with the zeolite containing captured Cs-137 being discharged into the waste tanks (zeolite is a natural mineral known for its ability to capture and retain cesium). The cesium removal columns were removed from service. As discussed in the individual waste tank histories in Section 2.2.2.2, Tanks 7, 18, 19, 25 and 27 contain zeolite resin. [CBU-PIT-2005-00099]

2.2.1.4 Miscellaneous Waste Streams

As described below, some FTF waste tanks have also received limited amounts of radioactive waste from other sources such as HTF, which receives waste from the H-Modified (HM) process in H Canyon, and SRNL, which analyzes process samples and tritiated water originating from the K-Area Reactor.

2.2.2 Waste Management

This section addresses processes used for waste storage and volume reduction and provides a capsule history of each FTF waste tank.

2.2.2.1 Waste Storage and Volume Reduction

As discussed previously, the primary function of FTF has been to support the F-Canyon operation by storing waste produced from the PUREX process. Management of waste stored in FTF has been complex due to the large volumes of waste produced by F-Canyon operations, efforts to reduce the volume of waste and waste removal from waste tanks. Because the PUREX process was based on a nitric acid flowsheet and because the waste tanks in FTF were constructed of carbon steel, the pH of the waste originating from F Canyon was chemically adjusted from an acidic solution to an alkaline solution through the addition of sodium hydroxide prior to transfer to FTF to protect the integrity of carbon steel waste tanks.

The volume of waste managed in FTF has been significantly reduced using the FTF evaporator systems. The 242-F Evaporator System operated from 1960 through 1988. The 242-16F Evaporator Facility entered service in 1980 and continues to operate. Operation of an evaporator involves the use of a waste tank to feed the waste to the evaporator and other waste tanks to receive the concentrated salt waste from the evaporation process.

Waste removal from the waste tanks as a precursor to future closure activities results in waste being transferred to other waste tanks. The waste removal process includes adding large volumes of water to aid in dissolving saltcake and suspending sludge into slurry for transfer. These factors make it necessary to carefully manage waste tank space to prevent the tank farm from becoming “water logged.”

The waste tank histories in Section 2.2.2.2 describe how individual waste tanks have been used to receive waste directly from F Canyon and in waste volume reduction and waste tank cleaning efforts.

2.2.2.2 Waste Tank History

The following waste tank histories are based on daily and monthly data report summaries, personal logs and tank-to-tank transfer data. Sludge and saltcake volumes include the interstitial liquid (i.e., liquid within the sludge and saltcake matrix) associated with those phases. Leak site information is included for the Type I and Type IV waste tanks known to have a history of leakage. No primary waste tank leakage has been detected in Type III/IIIA secondary liners or concrete vaults.

Type I Tanks

Tank 1 was constructed during 1952 and 1953 and entered service in 1954 as an F-Canyon waste receipt tank. This waste tank remained active and operational until 1973. Since then, Tank 1 has not received a waste transfer. The largest volume of waste stored in Tank 1 has been approximately 748,000 gallons. On March 31, 2010, Tank 1 contained approximately 480,000 gallons of saltcake, 2,700 gallons of supernate and 7,000 gallons of sludge. [DPSPU 78-11-8, SRR-LWP-2010-00040] Tank 1 received only high-heat waste initially; however, during 1962, it received low-heat waste. After 1962, the remainder of total waste was high-heat waste, which was last received in 1967. To prepare the waste tank for salt waste storage service, sludge removal was performed in 1969. Tank 1 became an evaporator concentrate receipt tank receiving concentrated supernate from the 242-F evaporator in 1969. This service concluded in 1973. Since then, Tank 1 has served as a salt waste storage tank. [DPSPU 78-11-8, WSRC-TR-93-425] Tank 1 is known to have leaked and a small quantity of waste was observed on the annulus floor. However, the exact location of the leak site has not been identified. The secondary containment system has worked as designed; the waste has been contained within the annular pan and has not reached the surrounding soils. [SRR-STI-2010-00283]

Tank 2 was constructed during 1952 and 1953 and entered service in 1955 as an F-Canyon waste receipt tank. This waste tank remained active and operational until 1973. Since then, Tank 2 has not received a waste transfer. The largest volume of waste stored in Tank 2 has been approximately 730,000 gallons. On March 31, 2010, Tank 2 contained approximately 536,000 gallons of saltcake and 4,000 gallons of sludge. [DPSPU 83-11-5, SRR-LWP-2010-00040] Tank 2 received high-heat waste for less than a year with the final receipt in 1956. To prepare the waste tank for salt waste storage service, sludge removal was performed in 1966. In 1967, Tank 2 became an evaporator concentrate receipt tank, receiving concentrated supernate from the 242-F evaporator. This service concluded in 1973. Since then, Tank 2 has served as a salt waste storage tank. [DPSPU 83-11-5, WSRC-TR-93-425] Tank 2 has no known leak sites. [SRR-STI-2010-00283]

Tank 3 was constructed during 1952 and 1953 and entered service in 1956 as an F-Canyon waste receipt tank. This waste tank remained active and operational until 1974. Since then, Tank 3 has not received additional waste. The largest volume of waste stored in Tank 3 at any time has been approximately 718,000 gallons. On March 31, 2010, Tank 3 contained approximately 536,000 gallons of saltcake and 4,000 gallons of sludge. [DPSPU 83-11-9, SRR-LWP-2010-00040] Tank 3 served as an F-Canyon waste receipt tank for high-heat waste through 1961 when it received its last direct receipt from F Canyon. To prepare the waste tank for salt waste storage service, sludge removal was performed in 1968. In 1968, Tank 3 became an evaporator concentrate receipt tank, receiving concentrated supernate from the 242-F evaporator. This service concluded in 1974. A brief interstitial liquid removal campaign was performed in 2003. [CBU-SPT-2004-00099] Since then, Tank 3 has served as a salt waste storage tank. [DPSPU 83-11-9, WSRC-TR-93-425] Tank 3 has no known leak sites. [SRR-STI-2010-00283]

Tank 4 was constructed during 1952 and 1953 and entered service in 1961 as an F-Canyon waste receipt tank. This waste tank remained active and operational until 1981. Since then, Tank 4 has not received waste. The largest volume of waste stored in Tank 4 has been approximately 728,000 gallons. [DPSPU 80-11-9] On March 31, 2010, Tank 4 contained approximately 32,000 gallons of sludge and 353,000 gallons of supernate. [SRR-LWP-2010-00040] Tank 4 received high-heat waste during two periods. The first began in 1961 and concluded in 1967. The second began in 1974 and concluded in 1980. Between the periods, Tank 4 was used as an evaporator concentrate receipt tank, receiving concentrated supernate from the 242-F evaporator. This service concluded in 1981. In 2007, a hard mineral layer called burkeite that existed above the sludge in the waste tank was dissolved and transferred out of the waste tank to prepare for sludge removal activities. [DPSPU 80-11-9, LWO-LWE-2007-00190, WSRC-TR-93-425] In 2009, approximately 136,000 gallons of sludge waste was transferred to HTF Tank 51. [SRR-LWP-2010-00007] Tank 4 has no known leak sites. [SRR-STI-2010-00283]

Tank 5 was constructed during 1952 and 1953 and entered service in 1959 as an F-Canyon waste receipt tank. This waste tank remained active and operational until 1973. The largest volume of waste stored in Tank 5 has been approximately 730,000 gallons. [DPSPU 80-11-9] On March 31, 2010, Tank 5 contained approximately 3,300 gallons of sludge and approximately 2,100 gallons of supernate. [SRR-LWP-2010-00040] Tank 5 received high-heat waste through 1969 when it last received direct receipt from F Canyon. Tank 5 also received approximately 106,000 gallons of waste described as “high chloride” from SRNL in 1968. From 1971 to 1973, Tank 5 supported 242-F evaporator operations by receiving and transferring concentrated supernate. In 2001, approximately 266,000 gallons of liquid was transferred from H Area to Tank 5 to create space in support of H Canyon and DWPF operations. Leak sites were identified and waste was removed below the lowest known leak site after the external primary waste tank wall was inspected in 2001. [DPSPU 81-11-10, WSRC-TR-93-425, HLW-2002-00025, ESH-EPG-2006-00005] Bulk waste removal and mechanical cleaning using submersible mixer pumps (SMPs) performed from 2005 to 2008 and left approximately 3,500 gallons of residual solids. [M-ESR-F-00147] A chemical cleaning campaign using oxalic acid, completed in 2008, reduced the residual solids volume to approximately 3,300 gallons. [M-ESR-F-00160] Additional cleaning in Tank 5 is planned for 2010. Inspections of Tank 5 through 2009 identified 34 leak sites and a small quantity of waste (less than 10 gallons) has been observed on the annulus floor. The secondary containment system has worked as designed; the waste has been contained within the annular pan and has not reached the surrounding soils. [SRR-STI-2010-00283]

Tank 6 was constructed in 1952 and entered service in 1964 as an F-Canyon waste receipt tank. This waste tank remained active and operational until 1974. The largest volume of waste stored in Tank 6 at any one time has been approximately 732,000 gallons. [DPSPU 81-11-4] On March 31, 2010, Tank 6 contained approximately 3,500 gallons of sludge and approximately 400,000 gallons of supernate. [SRR-LWP-2010-00040] Tank 6 received high-heat waste through 1972 when it received its last direct receipt from F Canyon. Tank 6 also received approximately 194,000 gallons of high-heat waste from SRNL in 1970. From 1972 through 1973, Tank 6 received and transferred over 1,000,000 gallons of high-heat waste supernate to support 242-F evaporator operations. In 1997, Tank 6 received approximately 280,000 gallons of material to support the Tank 17 closure activities. In 2001, approximately 315,000 gallons of liquid were transferred from H Area to Tank 6 to create space in support of DWPF operations. [DPSPU 81-11-4, WSRC-TR-93-425, HLW-2002-00025, ESH-EPG-2007-00016] Bulk waste removal and mechanical cleaning using SMPs, performed from 2006 to 2007, left approximately 6,000 gallons of residual solids. [M-ESR-F-00132] A chemical cleaning campaign using oxalic acid, completed in 2009, reduced the residual solids volume to approximately 3,500 gallons. [M-ESR-F-00165] Additional cleaning is planned for 2010. Tank 6 is known to have leaked at 11 identified leak sites and approximately 90 gallons of waste has been observed on the annulus floor. The secondary containment system has worked as designed; the waste has been contained within the annular pan and has not reached the surrounding soils. [SRR-STI-2010-00283]

Tank 7 was constructed during 1952 and 1953 and entered service in 1954 as an F-Canyon waste receipt tank until 1968. This waste tank remains active and operational. The largest volume of waste stored in Tank 7 has been approximately 733,000 gallons. On March 31, 2010, Tank 7 contained approximately 133,000 gallons of sludge and 263,000 gallons of supernate. [WSRC-TR-93-425, SRR-LWP-2010-00040] From 1960 and intermittently through 1973, Tank 7 also received approximately 402,000 gallons of non-canyon waste, which consisted of mostly heat exchanger and sand filter flushes from other SRS areas. Because the transfer jets for Tanks 1 through 6 and Tank 8 discharged into Tank 7, many transfers were made into and out of this waste tank. From 1960 to 1976, Tank 7 supported 242-F evaporator operations by receiving and transferring concentrated supernate from other Type I tanks to the Tank 18 evaporator feed tank. In 1974, the first inter-area waste transfer took place in which high-heat waste from HTF was transferred to FTF Tanks 7 and 18. Tank 7 directly fed the 242-F evaporator between 1977 and 1984. Tank 7 material was transferred to Tank 26, from 1985 to 1987, to support 242-16F evaporator operations. In 2003, Tank 7 supported Tank 18 waste removal and, in 2004, sludge waste was transferred from Tank 7 to HTF Tank 51. In 2003, a transfer of Tank 18 material introduced zeolite into Tank 7. The transfer system design places Tank 7 along the conventional transfer route to exit the Tank 1 through Tank 8 area. From 2005 to 2010, Tank 7 supported the waste removal operations for Tanks 4, 5, 6, 8, 18 and 19. [WSRC-TR-93-425, HLW-2002-00025, LWO-PIT-2007-00083, LWO-LWP-2008-00007, SRR-LWP-2010-00007] Tank 7 has no known leak sites. [SRR-STI-2010-00283]

Tank 8 was constructed during 1952 and 1953 and entered service in 1956 as an F-Canyon waste receipt tank. This waste tank remained active and operational until 1981. The largest volume of waste stored in Tank 8 has been approximately 732,000 gallons. On March 31, 2010, Tank 8 contained approximately 20,000 gallons of sludge and 431,000 gallons of supernate. [WSRC-TR-93-425, SRR-LWP-2010-00040] Tank 8 received only low-heat waste from 1956 through 1959. In 1960, Tank 8 began receiving high-heat waste, which was last received in 1974. In 1974, the remainder of the total waste received was low-heat waste, which was last received in 1981. During 1968 to 1970, Tank 8 also received approximately 436,000 gallons of waste from SRNL. Bulk waste removal operations were performed in 2000 through 2001. Sludge from Tank 8 went into a sludge batch for feed to DWPF [U-ESR-F-00009] In 2004, Tank 8 underwent sludge heel removal operations. [WSRC-TR-93-426, CBU-ENG-2005-00005] Supernate was transferred into and out of Tank 8 to support waste removal in Tank 4, 5, 6, 18 and 19 from 2005 to 2009. Tank 8 stored liquid from Tank 4 in 2007 to support burkeite removal from Tank 4. Aluminum-rich leachate was transferred to Tank 8 from Tank 51 in support of low-temperature aluminum dissolution for sludge processing preparation in 2009. [LWO-PIT-2006-00063_Redacted, LWO-PIT-2007-00083, LWO-LWP-2008-00007, SRR-LWP-2010-00007] Tank 8 has no known leak sites. [SRR-STI-2010-00283]

Type IV Tanks

Tank 17[13] was constructed in 1958 and entered service in 1961 as an F-Canyon waste receipt tank. This waste tank remained active and in operational service until 1982; it is now closed and filled with grout. The largest volume of waste stored in Tank 17 was approximately 1,300,000 gallons. [WSRC-TR-2004-00284, ESH-EPG-2008-00012, WSRC-TR-93-425] Tank 17 was an F-Canyon low-heat waste receipt tank during two periods. The first period began in 1961 and concluded in 1962. The second period, as an F-Canyon waste receipt tank, began in 1969 and concluded in 1977. In between the two periods, Tank 17 was used as an evaporator concentrate receipt tank, receiving concentrated supernate from the 242-F evaporator from 1964 to 1966. In 1977, at the end of its operations as a canyon waste receipt tank, Tank 17 received approximately 198,000 gallons of concentrated salt solution from the 242-F evaporator. From 1974 and to 1982, Tank 17 received approximately 177,000 gallons of non-canyon waste from various other SRS areas. In 1983, bulk waste removal operations began and then concluded early in 1986 with a spray washing campaign. In 1992, approximately 100,000 gallons of tritiated water from the K-Reactor Area was placed in Tank 17. Final waste removal in support of waste tank closure was completed in 1997 when Tank 17 was filled with grout and closed. The SCDHEC approved closure of Tank 17 on December 15, 1997. [PIT-MISC-0004, WSRC-TR-93-425, PIT-MISC-0016] Tank 17 had no history of leaks.

Tank 18 was constructed in 1958 and entered service in 1959 as an F-Canyon waste receipt tank. This waste tank remained active and operational until 1986 when waste removal activities were initiated. The largest volume of waste stored in Tank 18 has been approximately 1,300,000 gallons. On March 31, 2010, Tank 18 contained approximately 4,000 gallons of supernate and 4,000 gallons of sludge. [WSRC-TR-2004-00284, CBU-PIT-2004-00024, SRR-LWP-2010-00040] Between 1959 and 1977, Tank 18 received low-heat waste directly from F Canyon during multiple periods. Tank 18 also supported the 242-F evaporator operations, as both a receiver of concentrated supernate and overheads and as a feed tank for the evaporator. From 1962 to 1981, Tank 18 received concentrated supernate and, from 1966 to early 1983, overheads from the 242-F evaporator. From 1960 through 1976, Tank 18 was used as a feed tank for the 242-F evaporator. In 1973, Tank 18 also received approximately 12,000 gallons of waste from HTF evaporator overheads and, in 1974, approximately 719,000 gallons of high-heat waste from HTF were received. Tank 18 was designed as the sole conventional transfer route to exit the Tank 17 through Tank 20 area. In 1980 and 1981, Tank 18 received salt and/or sludge removal waste from Tanks 17, 19 and 20 waste removal activities. Throughout this period, Tank 18 served as a hub for Tanks 17 through 20 activities. During this operation, some of the zeolite resins, which were confined to Tank 19, were transferred to, and settled in, Tank 18. [CBU-PIT-2005-00124] Mechanical heel removal occurred in Tank 18 in 2003 using a large mixer pump, the Advanced Design Mixer Pump (ADMP). Additional heel removal was performed in 2009 using a mechanical crawler-based ultra-high-pressure eductor retrieval system (Mantis) that reduced the volume of residual solids to approximately 4,000 gallons. [U-ESR-F-00041] Tank 18 has no known leak sites. [SRR-STI-2010-00283]

Tank 19 was constructed in 1958 and entered service in 1961 as an F-Canyon cascade waste receipt tank. This waste tank remained active and in operational service until 1980 when waste removal activities were initiated. The largest volume of waste stored in Tank 19 has been approximately 1,300,000 gallons. On March 31, 2010, Tank 19 contained approximately 2,000 gallons of sludge and less than 830 gallons of supernate. [WSRC-TR-93-425, CBU-PIT-2005-00124, SRR-LWP-2010-00040] Tank 19 initially received fresh canyon waste from Tank 17, the primary F-Canyon waste receipt tank, to create space for Tank 17 to directly receive additional canyon waste. Therefore, Tank 19 was considered to be a cascade waste receipt tank. Tank 19 served this function by receiving low-heat waste in the first six months of operation and again from 1969 to 1972. In 1974, Tank 19 received a relatively small volume of low-heat waste directly from F Canyon. From 1962 to 1976, Tank 19 served as an evaporator

[13] Tank 17 and Tank 20 were grouted and removed from service in 1997 and are not part of this Draft FTF 3116 Basis Document.

concentrate receipt tank for the 242-F evaporator. Tank 19 also received approximately 13,000 gallons of spent zeolite resin from a cesium removal column that was mounted on a Tank 19 riser to remove cesium from the evaporator overheads. In 1980, Tank 19 salt removal began and concluded in 1981. In 2000 and 2001, mechanical heel removal was performed using SMPs. [CBU-PIT-2005-00124] Subsequently, additional heel removal was performed in 2008 and 2009 using a Mantis that reduced the volume of residual solids to approximately 2,000 gallons. [U-ESR-F-00042] Tank 19 has a history of in-leakage. Visual inspections have revealed two sites where in-leakage occurred prior to 1994. Two additional in-leakage sites where documented in 2009. [SRR-STI-2010-00283]

Tank 20 was constructed in 1958 and entered service in 1960 as a concentrate receipt tank for the 242-F evaporator. This waste tank remained active and operational until 1977; it is now closed and filled with grout. The largest volume of waste stored in Tank 20 was approximately 1,300,000 gallons. [WSRC-TR-93-425, ESH-EPG-2008-00012, DPSPU 82-11-10] Tank 20 began its operational life as the 242-F evaporator concentrate receipt tank from 1960 and to 1976. During this time, Tank 20 received a combination of evaporator concentrate from both low-heat and high-heat waste processing. Tank 20 also received approximately 264,000 gallons of condensation water from the concentrate transfer system and flushes of the concentrate transfer system pump pit. The first saltcake removal campaign began in 1980 and concluded in 1982. In 1986, the second salt removal campaign occurred. In 1988, Tank 20 was spray washed. In 1996, Tank 20 was filled with grout and closure operations concluded. The SCDHEC approved closure of Tank 20 on July 31, 1997. [DPSPU 82-11-10, PIT-MISC-0002, PIT-MISC-0018] Tank 20 had a history of in-leakage only. [SRR-STI-2010-00283, SRR-LWP-2010-00007]

Type III Tanks

Tank 33 was constructed in 1969 and entered service in 1974 as a concentrate receipt tank used, primarily for the 242-F evaporator, but also with limited use for the 242-16F evaporator. The largest volume of waste stored in Tank 33 has been approximately 1,270,000 gallons. As of March 31, 2010, Tank 33 contained approximately 294,000 gallons of saltcake, 659,000 gallons of supernate and 80,000 gallons of sludge. [SRR-LWP-2010-00040] Through 1984, Tank 33 received mainly concentrated salt solution from the 242-F evaporator and a small volume from the 242-16F evaporator. From 1983 to 2003, Tank 33 received both high-heat and low-heat waste directly from F Canyon. In 2006 and 2007, supernate was removed from Tank 33 to create space to receive dissolved burkeite salt solution from Tank 4. In 2007, Tank 33 received approximately 200,000 gallons of dissolved burkeite salt solution. In 2008 and 2009, approximately 400,000 gallons of supernate were transferred from Tank 33 to Tank 7 to neutralize oxalic acid used during chemical cleaning of Tanks 5 and 6. Tank 33 was fitted with deployable cooling coils in 1976. [WSRC-TR-93-425, LWO-PIT-2007-00083, SRR-LWP-2010-00007]

Tank 34 was constructed in 1972 and entered service in 1973 as a concentrate receipt tank for the 242-F evaporator. The largest volume of waste stored in Tank 34 has been approximately 1,298,000 gallons. As of March 31, 2010, Tank 34 contained approximately 191,000 gallons of saltcake, 902,000 gallons of supernate and 12,600 gallons of sludge. [WSRC-TR-93-425, SRR-LWP-2010-00040] Through 1985, Tank 34 received concentrated salt solution from the 242-F evaporator. From 1980 to 1995, Tank 34 received high-heat and low-heat waste directly from F Canyon. Tank 34 also received waste solution from the HTF in 2001 and 2006. In 2006 and 2007, Tank 34 received interstitial liquid from Tank 25. In 2008, approximately 270,000 gallons of supernate were transferred from Tank 34 to Tank 7 to neutralize oxalic acid used during chemical cleaning of Tanks 5 and 6. Tank 34 was fitted with deployable cooling coils in 1976. [LWO-PIT-2007-00088, WSRC-TR-93-425, SRR-LWP-2010-00007]

Type IIIA Tanks

Tank 25 was constructed in 1978 and entered service in 1980 as a concentrate receipt tank for the 242-F evaporator. This tank continues to operate. The largest volume of waste stored in Tank 25 has been approximately 1,280,000 gallons. On March 31, 2010, Tank 25 contained approximately 506,000 gallons of saltcake and 306,000 gallons of supernate. [CBU-SPT-2004-00081, SRR-LWP-2010-00040] Tank 25 began its operational life of as the 242-F evaporator concentrate receipt tank from 1980 to 1988. The tank was equipped with a cesium removal column used to remove cesium from the evaporator overheads. Through 1988 approximately 1,050 gallons of zeolite resin used in the column were discharged into the tank. From 1988 through 1991, only transfers out of Tank 25 occurred. From 1991 to 2005, Tank 25 was inactive. [CBU-SPT-2004-00081, CBU-PIT-2005-00099, WSRC-TR-93-425] In 2005, saltcake removal operations were initiated to allow Tank 25 to serve as a concentrate receipt tank for the 242-16F evaporator and the supernate above the salt layer was removed. In 2006 and 2007, interstitial liquid was removed and transferred to Tank 34. Salt dissolution activities were performed in 2008 and 2009, and Tank 25 was converted to the 242-16F evaporator concentrate receipt tank in 2010. [LWO-PIT-2007-00039, CBU-SPT-2004-00081, SRR-LWE-2010-00027]

Tank 26 was constructed in 1978 and entered service in 1980 as a feed tank for the 242-16F evaporator. The largest volume of waste stored in Tank 26 has been approximately 1,274,000 gallons. On March 31, 2010, Tank 26 contained approximately 876,000 gallons of supernate and 267,000 gallons of sludge. [SRR-LWP-2010-00040, WSRC-TR-93-425] Tank 26 has served as a feed tank for the 242-F and 242-16F. Because the 242-F evaporator is no longer in service, this waste tank currently feeds only the 242-16F evaporator. Tank 26 received concentrated supernate, low-heat waste and overheads from the evaporators. Tank 26 also received supernate from the HTF, some of which originated from the Receiving Basin for Off-site Fuel, which was fed to the 242-16F evaporator. [WSRC-TR-93-425, SRR-LWP-2010-00007]

Tank 27 was constructed in 1978 and entered service in 1980 as a receipt tank for salt wastes and concentrated supernate. The largest volume of waste stored in Tank 27 has been approximately 1,274,000 gallons. As of March 31, 2010, Tank 27 contained approximately 1,160,000 gallons of saltcake, 87,000 gallons of supernate and 3,900 gallons of sludge. [SRR-LWP-2010-00040, WSRC-TR-93-425] From 1980 to 1984, Tank 27 received low-heat waste supernate, salt waste and dissolved salt solution from Tanks 7 and 18. From 1983 through 1988, Tank 27 served as an evaporator concentrate receipt tank for the 242-F evaporator. It was equipped with a cesium removal column and served as an evaporator concentrate receipt tank for the 242-16F evaporator from 2004 to 2009. Since then, Tank 27 has served as a salt waste storage tank. Approximately 5,000 gallons of zeolite resin used in the cesium removal column was discharged into the tank through 1989. During 2004 and 2006, Tank 27 also received waste solutions from the HTF to feed the 242-16F evaporator. [WSRC-TR-93-425, CBU-PIT-2005-00099, SRR-LWP-2010-00007]

Tank 28 was constructed in 1978 and entered service in 1980 as an evaporator concentrate receipt tank for the 242-16F evaporator. The largest volume of waste stored in Tank 28 has been approximately 1,266,000 gallons. As of March 31, 2010, Tank 28 contained approximately 1,030,000 gallons of saltcake and 195,000 gallons of supernate. [CBU-PIT-2005-00051, SRR-LWP-2010-00040] From 1980 until 1985, Tank 28 served as an evaporator concentrate receipt tank for the 242-16F evaporator and, in 1980, also received concentrated supernate from Tank 18. Transfers out of Tank 28 were made until 1988. Since then, Tank 28 has not received a waste transfer. [CBU-PIT-2005-00051, WSRC-TR-93-425]

Tank 44 was constructed in 1980 and entered service in 1982 as an evaporator concentrate receipt tank for the 242-16F evaporator. The largest volume of waste stored in Tank 44 has been approximately 1,273,000 gallons. As of March 31, 2010, Tank 44 contained approximately 1,010,000 gallons of saltcake and 244,000 gallons of supernate. [CBU-PIT-2006-00101, SRR-LWP-2010-00040] From 1982 through 1992, Tank 44 received concentrated salt solution from the 242-16F evaporator. Supernate was transferred from Tank 44 to Tank 26 from 1982 to 1992 and again in 2004. Since then, Tank 44 has not received a waste transfer. [CBU-PIT-2006-00101, SRR-LWP-2010-00007]

Tank 45 was constructed in 1980 and entered service in 1982 as an evaporator concentrate receipt tank for the 242-16F evaporator. The largest volume of waste stored in Tank 45 has been approximately 1,269,000 gallons. As of March 31, 2010, Tank 45 contained approximately 1,100,000 gallons of saltcake and 144,000 gallons of supernate. [CBU-PIT-2006-00044, SRR-LWP-2010-00040] From 1982 to 1993, Tank 45 received concentrated salt solution from the 242-16F evaporator and supernate was transferred from Tank 45 to Tank 26 from 1983 to 1993. Since then, Tank 45 has not received a waste transfer. [WSCR-TR-93-425, CBU-PIT-2006-00044, SRR-LWP-2010-00007]

Tank 46 was constructed in 1980 as a spare tank and entered service in 1994 to support the 242-16F evaporator. The largest volume of waste stored in Tank 46 has been approximately 1,249,000 gallons. As of March 31, 2010, Tank 46 contained approximately 1,160,000 gallons of saltcake and 96,700 gallons of supernate. [SRR-LWP-2010-00040] Tank 46 received concentrated salt solution from the 242-16F feed tank in 1994 and supernate was transferred from Tank 46 to Tank 26 from 1994 to 1995, and from

2003 to 2005. Since then, Tank 46 has not received a waste transfer. [WSRC-TR-93-425, SRR-LWP-2010-00007]

Tank 47 was constructed in 1980 and entered service in 1981 as a waste receipt tank for fresh F-Canyon waste and concentrated salt solution from evaporator operations. The largest volume of waste stored in Tank 47 has been approximately 1,272,000 gallons. As of March 31, 2010, Tank 47 contained approximately 773,000 gallons of saltcake, 209,000 gallons of supernate and 248,000 gallons of sludge. [WSRC-TR-93-425, SRR-LWP-2010-00040] From 1981 to 1986, Tank 47 received low-heat waste from F Canyon. From 1983 to 1988, the tank also received concentrated salt solution from the 242-16F evaporator. In 1981, Tank 47 received salt removal waste from Tank 18, in 1987 concentrated supernate from Tank 26, and from 1982 to 1991 Tank 47 received approximately 280,000 gallons of non-canyon low-level waste from other SRS areas. Tank 47 served as the 242-16F evaporator concentrate receipt tank in 2008 and 2009. [SRR-LWP-2010-00007]

2.3 Waste Removal Approach for Waste Tanks, Annuli and Ancillary Structures

The closure process for the tank systems begins with removing the waste using mechanical, chemical, and/or vacuum waste removal techniques, or other methods of comparable or greater effectiveness, discussed below.

Bulk waste removal is the first step toward waste tank closure and typically employs agitation/mixer pumps to suspend solids and potentially dissolve soluble material. If the tank contains saltcake, chemically treated water is added to dissolve the waste which results in a considerable impact to the tank farm waste storage space. In order to make the salt mobile for transferring and to adjust the salt concentrations (molarity) for processing and disposal, significant quantities of chemically treated water must be added. Type III/IIIA tank space must be available for receiving and adjusting these solutions. [DOE-WD-2005-001] Bulk sludge and/or salt waste is transferred from the tank, leaving behind a heel.

Mechanical heel removal employs techniques such as agitation, spraying, lancing, pulse jet mixing, vacuum retrieval, mechanical manipulators, robotic devices or recycle systems to augment existing waste removal equipment to reduce the heel volume. These technologies are described in Section 2.3.2.

Chemical heel removal employs oxalic acid or other specialized chemical treatment of the heel to dissolve solids that could not be removed by mechanical methods and water addition alone. The oxalic acid may be sprayed into the tank to clean contaminants from the internal tank surfaces (e.g., walls, cooling coils, support columns, equipment).

If necessary, at the conclusion of chemical heel removal, the interior of the waste tank may be washed with water to rinse oxalic acid from internal surfaces and dislodge loose contamination. The wash water will then be removed.

For waste tanks that have leaked waste from primary to secondary containment, waste will be removed as appropriate from the annulus as described in Section 2.3.6.

Waste tank cleaning is subject to a variety of operating constraints including:

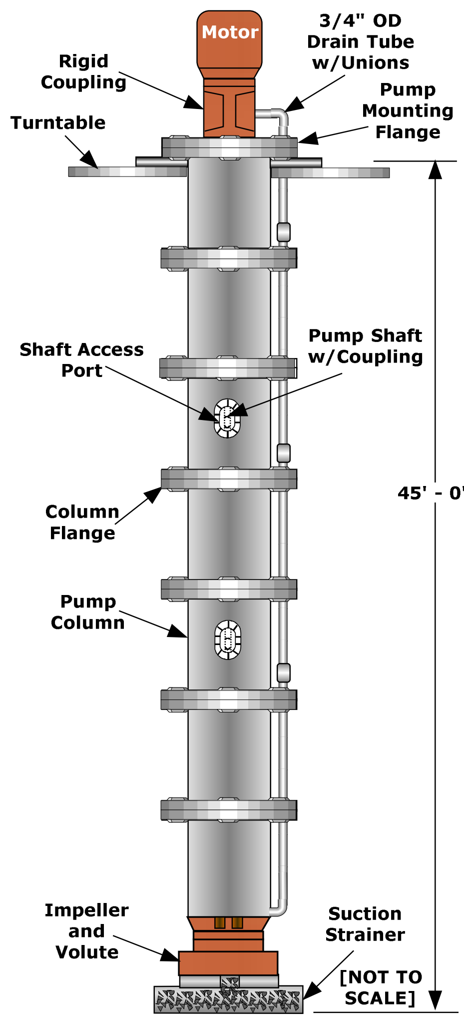
- maintaining emergency tank space,
- controlling tank chemistry and radionuclide inventory,
- requirements to remove waste from tanks with a leakage history and tanks that do not meet secondary containment and leak detection requirements,
- preparing waste for downstream waste treatment facilities, and
- meeting safety basis requirements.

The complex interdependency of safety and process requirements of the various liquid waste facilities drive the sequencing of the tanks undergoing waste removal and tank cleaning. Plans for bulk waste removal, mechanical, chemical and vacuum cleaning and tank closure are summarized in the *Liquid Waste System Plan*. [SRR-LWP-2009-00001] See Section 1.3 for discussion on the closure schedule.

2.3.1 Tank Waste Removal History

Early waste removal efforts at SRS (1960s) employed hydraulic mining and sluicing techniques using once-through water at high pressure. The practice was discontinued because insufficient tank storage space was available to accommodate the large volume of water added to the tank farm system. The technique was modified to use existing waste supernate as the slurry liquid media and slurry pumps for breaking up and suspending the sludge. Using this technique, several slurry pumps are submerged in the tank being cleaned in lieu of the external pumps formerly used. This change allows the slurrying operation to be repeated as often as necessary to suspend the sludge without adding significant new waste volume to the tank farm. Figure 2.3-1 presents a typical standard slurry pump design. A slurry pump is a vertical shafted centrifugal pump with the drive motor mounted topside. A coupled shaft connects the motor and pump. Suction is drawn into the pump and discharged from two nozzles (aimed in opposite directions from each side of the pump). The nozzles are shaped such that high velocity jets are ejected into the liquid. The pump rotates on a turntable, thereby spinning the jets in the horizontal plane. The pumps are typically installed in available risers such that the circular pattern of suspended solids, or effective cleaning radius, of each individual pump overlaps with the adjacent pump so the entire waste tank contents are slurried. The initial elevation of the pump suction is typically positioned just above the sludge layer. Water may be added to the tank if there is not enough supernate to use as the slurry media. The pumps typically suspend sludge that can be suspended (at that slurry pump elevation setting) within a few days. The slurry pumps are then lowered typically in 10- to 17-inch increments, more water is added, if needed, and the next layer of sludge is suspended. This process is repeated until the slurry pumps are at the lowest elevation practical, typically 10 inches above the waste tank floor. The transfer pump is then lowered to the desired elevation, typically 6 inches above the tank floor. The sludge slurry is then transferred out of the tank. To obtain the proper weight percent of suspended solids in the resulting sludge batch, more than one transfer may be required. Examples of FTF tanks in which this technique has been successfully used for bulk sludge removal include Tanks 8, 17 and 18. [HLW-2002-00025]

Figure 2.3-1: Typical Standard Slurry Pump



[WSRC-TR-2001-00313]

Slurry pumps have also been used for bulk waste removal in tanks containing saltcake. To remove saltcake, the pumps are positioned just above the saltcake and water is added to the tank. The water is stirred by the pumps to dissolve the top layer of saltcake. Once the resulting solution becomes nearly saturated with dissolved salt it is transferred out of the tank. The slurry pumps are then lowered, water is added and the process is repeated. In FTF, this technique has been successfully used for bulk salt removal in Tank 19. [HLW-2002-00025]

Saltcake can also be dissolved without agitation. To dissolve the saltcake and create salt solution batches, the waste tank is filled with dissolution water, which is water chemically treated to prevent corrosion of the carbon steel waste tanks, until the saltcake surface is flooded. The water dissolves a portion of the saltcake forming a salt solution. The salt solution is then transferred out of the tank. In FTF, this process has been used for saltcake removal in Tank 20. [HLW-2002-00025]

Processing the salt waste requires waste tank space. To make the salt mobile for transferring and to adjust the salt concentrations (molarity) for processing and disposal, significant quantities of chemically treated water must be added. Type III/IIIA tank space must be available for receiving and adjusting these solutions.

2.3.2 Waste Removal Technologies

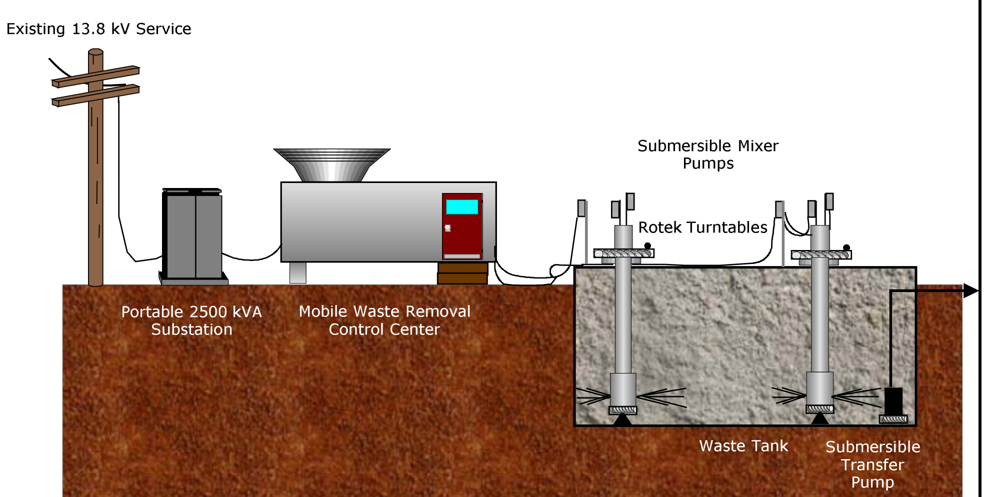
Slurry pumps previously installed in FTF waste tanks may continue to be used in waste removal activities. In addition, a new generation of waste removal equipment has been developed as described in the following subsections.

2.3.2.1 Submersible Mixer Pumps - Type I and III/IIIA Waste Tanks

In 2003, SRS used a systematic process to identify, evaluate and select equipment for waste removal tasks to accelerate clean up. This process is documented in a Systems Engineering Evaluation. [G-ESR-G-00051] The study investigated options for bulk waste mixing, waste transfer and heel removal. The study graded the options on weighted selection criteria such as technical maturity, effectiveness, reliability, reusability, radiological control requirements, integration with the tank farm system and cost. Knowledgeable tank farm operations, engineering, plant support and maintenance personnel identified potential technology candidates based on experience, literature, world wide web research and contacts with other knowledgeable personnel in the DOE complex and commercial industry.

The team recommended using floor-mounted canned SMPs for bulk waste mixing, a mast-mounted Submersible Transfer Pump (STP) for waste transfer, chemical cleaning using oxalic acid for final heel removal and an air-driven submersible pump to enhance final heel removal if centrifugal STPs are insufficient to remove the final residual material. The technology consists of a mobile substation that provides power, a Mobile Waste Removal Control Center that provides local control and monitoring capabilities, SMPs for mixing and suspending waste solids and an STP for waste transfer. These mobile units have the capability of being co-located near any tank or tanks scheduled for waste removal. This concept efficiently performs waste removal using mobile and reusable equipment (Figure 2.3-2)[14]. [G-ESR-G-00051]

Figure 2.3-2: Submersible Mixer Pump Waste Removal Diagram



[14] Figure 2.3-2 depicts two SMPs located in the waste tank, however, during actual cleaning operations DOE may deploy from one to four SMPs within a waste tank based upon the particular tank configuration and waste characteristics.

To date, the SMPs have been used to perform bulk waste removal on Tanks 4, 5 and 6 and support mechanical and chemical cleaning of residual heels in Tanks 5 and 6. The SMPs are variable speed, single-stage centrifugal pumps with a 305 hp motor that can operate up to 1,600 rpm. The SMPs utilize the tank liquid waste to cool the motor and lubricate the upper and lower bearings. Two discharge nozzles give the SMPs the capability to produce an effective cleaning radius of up to 50 feet. The SMPs are rotated by a turntable assembly that provides the motive force for oscillation, or allows for stationary indexing operation. The SMPs have a rotating foot attached to the lower end of the pump, which allows the SMP to rest on the tank floor and oscillate. [M-CLC-G-00344]

Waste transfers during Tanks 4, 5 and 6 bulk waste removal and Tanks 5 and 6 mechanical cleaning were achieved by using a STP. The STP is a 15 hp, 3,450 rpm, 250 gpm centrifugal pump that has the capability of being located at any elevation within the waste tank. The pump is typically located inside a 22-inch diameter sleeve pipe (caisson) that rests on the tank floor. The caisson protects the STP from direct discharge from the SMPs. Pump configurations are shown in Figure 2.3-3 and Figure 2.3-4.

Figure 2.3-3: Submersible Mixing Pump and Submersible Transfer Pump

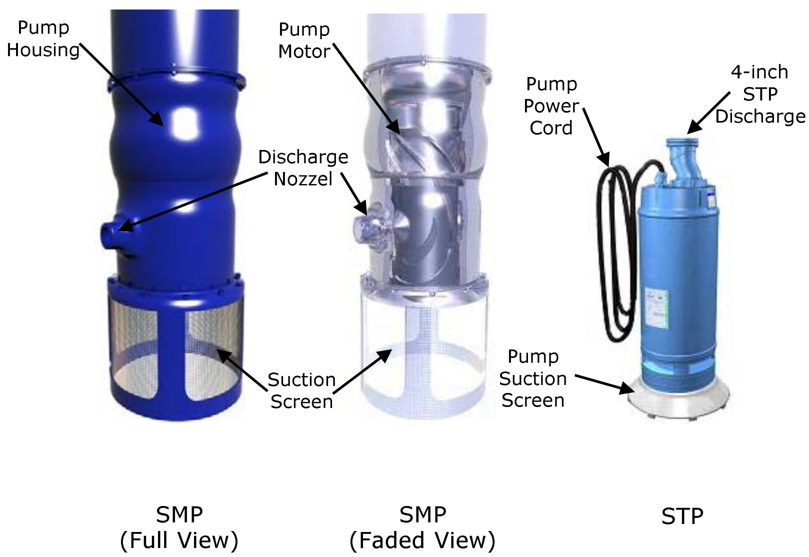
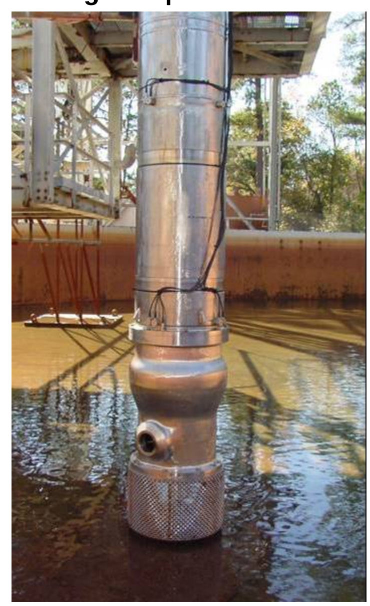


Figure 2.3-4: Submersible Mixing Pump in a Test Tank



During waste removal supported by SMPs, supernate may be used as the slurry media to minimize the amount of water added to the tank farm system. Minimization of water additions to the tank farm is important because water introduced into a waste tank becomes additional waste that occupies the limited available waste storage space. The SMPs are operated using various strategies depending on the configuration of the waste in that particular tank. To keep the solids suspended in the slurry, mixing continues as long as possible while the slurry is transferred out of the tank. The SMPs are required to be shut down as the liquid level approaches the elevation of the discharge nozzles to prevent waste spraying, which may cause filters in the tank ventilation system to become inoperable and cause potentially high exposure to workers. The STP continues to operate until it loses the ability to pump out any additional waste. After the transfer, the residual solids configuration and volume are assessed based on the condition of the waste remaining in the tank. Liquid may be transferred back into the waste tank for additional mixing and transfer cycles.

2.3.3 Chemical Cleaning

Chemically-aided cleaning techniques have been evaluated for additional levels of waste removal following mechanical heel removal. A team of knowledgeable and experienced engineers and scientists assessed the current knowledge base and collected and evaluated information available on chemical-based methods for removing residual solids from the waste tanks. [WSRC-TR-2003-00401] As part of this study, the team developed recommendations for chemical treatments to remove residual solids. The cleaning agents identified included:

- oxalic acid,
- a mixture of oxalic acid and citric acid,
- a combination of oxalic acid with hydrogen peroxide,
- nitric acid,
- formic acid, and
- organics.

The results of the evaluation support oxalic acid as the optimal cleaning agent. Nitric acid, formic acid and oxalic acid with hydrogen peroxide were all closely grouped for the next best choice. The mixture of oxalic acid and citric acid rated poorly (primarily due to the fact that it performed less well than oxalic acid and the presence of citrate could adversely impact downstream operations, such as the Salt Waste Processing Facility [SWPF] and the DWPF). Organics rated even more poorly due to large uncertainties in performance and downstream impacts.

The use of oxalic acid was recommended for a number of reasons. First, oxalic acid has been widely studied and used several times to clean waste tanks at SRS and at other sites within the DOE Complex. Second, the oxalic acid has been shown to be effective for a wide variety of sludge types and out-performed nitric acid and other chemical cleaning agents in head-to-head tests. Lastly, oxalic acid is less corrosive to the carbon steel tank than nitric acid or a combination of oxalic acid and hydrogen peroxide. [WSRC-TR-2003-00401]

Chemical cleaning using bulk oxalic acid is the method planned for heel removal in FTF waste tanks. However, other cleaning solutions or methods may be used if they have comparable or greater effectiveness. The planned chemical cleaning methods are similar to bulk oxalic acid cleaning methods used in Tanks 5, 6 and HTF Tank 16[15], consisting of several oxalic acid strikes, use of agitation to facilitate particle-acid contact and a final clean water spray wash and mixing operation. If needed, oxalic acid will be sprayed into the tank to clean contaminants from internal tank surfaces (e.g., walls, cooling coils, support columns, equipment, etc.).

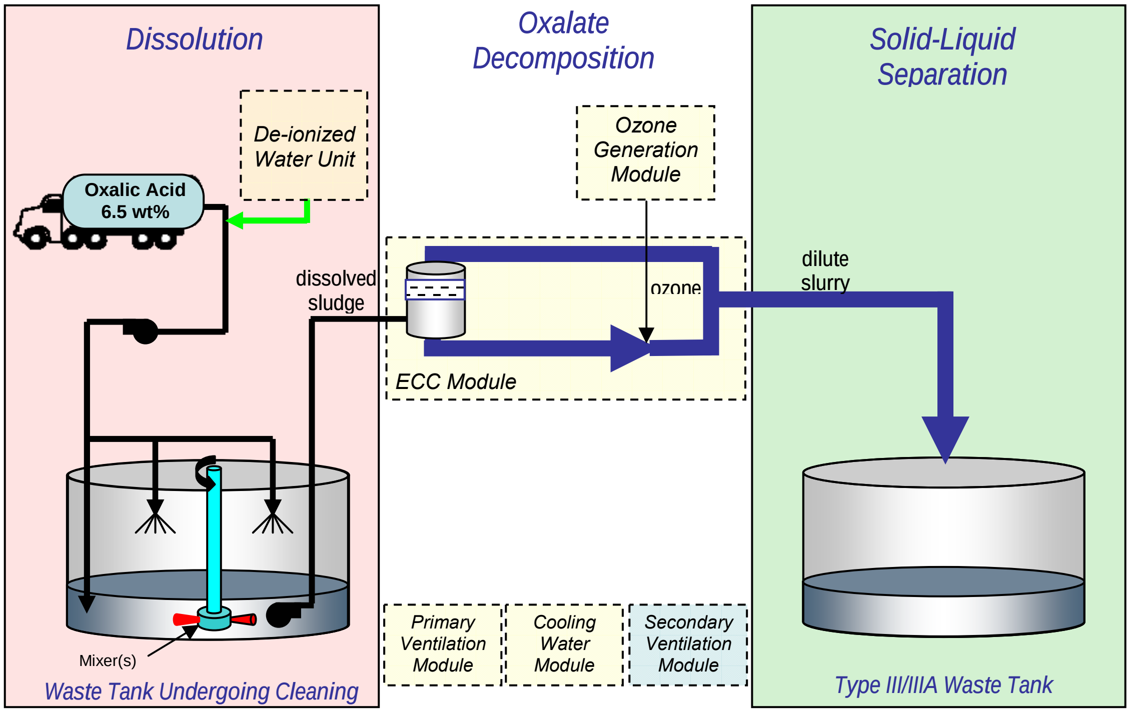
Oxalic acid cleaning produces sodium oxalates in the solids slurry that will be added to sludge batches that feed the DWPF. Because of sodium limits and oxalate restrictions on the DWPF feed, preparation of the feed results in a significant amount of additional material being generated that eventually must be processed through SWPF and disposed of in the Saltstone Disposal Facility[16]. Therefore, the oxalic acid flowsheet evaluations have considered the effects of oxalates on DWPF and salt processing to determine the quantities of oxalic acid that can be tolerated in the Liquid Waste System. [WSRC-TR-2004-00317] Due to the downstream effect of oxalic acid on DWPF and salt processing, the selection of a chemical cleaning method will be considered on each individual application basis. Environmental conditions to which the waste has been exposed also affect its dissolution characteristics; therefore, in future chemical cleaning planning, each waste tank (or groups of tanks with similar waste and similar historical conditioning) will be considered individually.

Enhanced Chemical Cleaning (ECC) will also be used to minimize downstream impacts of bulk oxalic acid cleaning. The ECC system consists of three distinct process steps: 1) dissolution/removal, 2) oxalic acid decomposition and 3) solid/liquid separation (Figure 2.3-5)[17]. In this system, waste is dissolved using dilute oxalic acid and transferred in batches to the oxalate decomposition process. In the decomposition process ozone is used to decompose the oxalate. As the oxalate is decomposed, the solids reform primarily as oxides, creating a dilute slurry. This dilute slurry is oxidized until the oxalate concentration falls below the target. When the oxalate concentration is low enough, the decomposition loop is stopped and the dilute slurry is fed to a settling tank (i.e., FTF waste tank) with the liquid fraction undergoing

[15] Discussion on Tank 16 cleaning is included for information regarding use of the oxalic acid cleaning process; however, Tank 16 is located in the HTF at SRS and is not within the scope of this Draft FTF 3116 Basis Document.
[16] See Appendix A for a brief description of DWPF, SWPF and Saltstone Disposal Facility operations.
[17] The ECC process is still in the stage of being tested and designed by DOE. The figure depicted represents a conceptual diagram of the ECC process and is subject to change as development of the process advances.

subsequent evaporation. This process is repeated until the tank sludge heel has been removed. [LWO-2008-0056]

Figure 2.3-5. ECC Process Diagram



2.3.4 Mantis - Type IV Waste Tanks 18 and 19

Tanks 18 and 19 have each undergone bulk waste removal using slurry pumps and mechanical heel removal. Bulk waste removal was performed in Tank 18 during 1986 and 1987; approximately 515,000 gallons of sludge was removed. [CBU-PIT-2005-00233] Mechanical heel removal occurred in Tank 18 in 2003 using a large mixer pump, the ADMP. The Tank 18 interior tank walls were spray washed following mechanical heel removal. Bulk waste removal was performed in Tank 19 during 1980 and 1981; approximately 1,033,000 gallons of saltcake was removed. [DPSP-84-17-7] Mechanical heel removal occurred during 2000 and 2001 using rotating submersible jet mixer pumps. The principal difficulty of the Tank 19 heel removal efforts was the characteristics of zeolite. Zeolite has proven to be resistant to removal because of its significantly higher settling velocity when compared to that of metal hydroxides and insoluble salts (i.e., traditional sludge waste). Zeolite is also resistant to dissolving with acids. [WSRC-TR-2002-00288]

As a result of the March 2006 DOE-sponsored Tank Cleaning Technical Exchange, a new waste tank tethered mechanical crawler-based cleaning technology was identified. The DOE has adapted and successfully used this new technology in the unobstructed Type IV tanks, Tanks 18 and 19. [CBU-PIT-2006-00067] The cooling coils in Type I, Type III and Type IIIA tanks severely restrict the mobility of tethered mechanical crawlers and make deployment of this device impractical in obstructed tanks. This new cleaning device allowed additional removal of waste from Tanks 18 and 19 to a greater extent than the technologies available when waste removal was initially discontinued in 2003 and 2001, respectively.

The cleaning device, called a Mantis, consisted of a mechanical crawler and an eductor assembly that made up a retrieval system utilizing an ultra-high-pressure water eductor (Figure 2.3-6). The process system consisted of a remotely controlled in-tank Mantis, an umbilical hose containing hydraulic supply lines and the high-pressure water hoses, in-tank waste retrieval hose, a diesel-driven ultra-high-pressure water pump, a motor-driven high pressure water pump, hydraulic pump skid, a diesel generator, above-ground hose-in-hose transfer lines, Waste Mixing Chamber (WMC) and support equipment. The device entered the tank through a 24-inch riser in a folded position. Once inside the tank, the device unfolded into its operational configuration.

Figure 2.3-6: Mantis

The Mantis was remotely driven around the waste tank floor by an operator located in the Control Center. A high pressure (maximum 10 thousand pounds per square inch) hydro-lance at its front was used to break up waste mounds and an eductor was used to aspirate waste from the floor of the waste tank. Operators monitored the operation of the Mantis using in-tank lighting, cameras and video surveillance equipment. The waste traveled through the eductor in-tank waste retrieval flexible hose and up into a tee spool piece located on top of the riser.

The waste then traveled through above-ground transfer lines (one from Tank 18 and one from Tank 19) routed to the receipt tank. The above-ground transfer lines consisted of a “hose within a hose” design with appropriate shielding. The above-ground waste transfer lines terminated inside a WMC installed inside a riser on the receipt tank. The WMC contained particle size reduction equipment (an immersion mill). The immersion mill, located near the bottom of the WMC, size-reduced solid waste particles so that the particles could be more easily re-suspended in future waste removal activities. [WSRC-TR-2007-00327] Heel removal in 2008 and 2009 using the Mantis in Tank 19 reduced the volume of residual solids to approximately 2,000 gallons. [U-ESR-F-00042] Heel removal operations with the Mantis in Tank 18 in 2009 reduced the volume of residual solids to approximately 4,000 gallons. [U-ESR-F-00041]

2.3.5 Removal of Residual Waste from Failed Cooling Coils

DOE recognizes that some waste tank cooling coils have failed and that additional failures during waste removal are likely. Potential failure mechanisms include:

- pitting and cracking due to corrosion,
- cracking due to thermal expansion and contraction, and
- breaks due to forces imposed on the coils from mixer operation during bulk waste removal and heel removal.

Once a coil has failed, the coil may become internally contaminated with waste. Cooling coils with the potential for residual waste holdup will be flushed, as needed.

2.3.6 Annulus Cleaning

Currently, three waste tanks have leak sites on the primary tank walls. None of the leaks from the primary tanks have resulted in waste escaping to the surrounding soils. Tank 1 is known to have leaked and a small quantity of waste has been observed on the annulus floor. The location of the leak site has not been identified. Inspections performed through 2009 have identified 34 leak sites in Tank 5 with a small quantity of waste (less than 10 gallons) on the annulus floor (Figure 2.3-7). Dried salt nodules and deposits have formed on the tank exterior wall. Tank 6 has 11 known leak sites with approximately 92 gallons of dried waste on the annulus floor (Figure 2.3-8). Waste material in the annulus is expected to be readily dissolved and removed. In 2009, 75% of the Tank 5 exterior wall and 100% of the Tank 6 exterior wall were inspected and dried deposits were cleaned using a magnetically mounted wall crawler with washing capabilities. [SRR-STI-2010-00283]

Figure 2.3-7: Waste Deposits on Exterior Primary Tank Wall Before Cleaning (Tank 5)

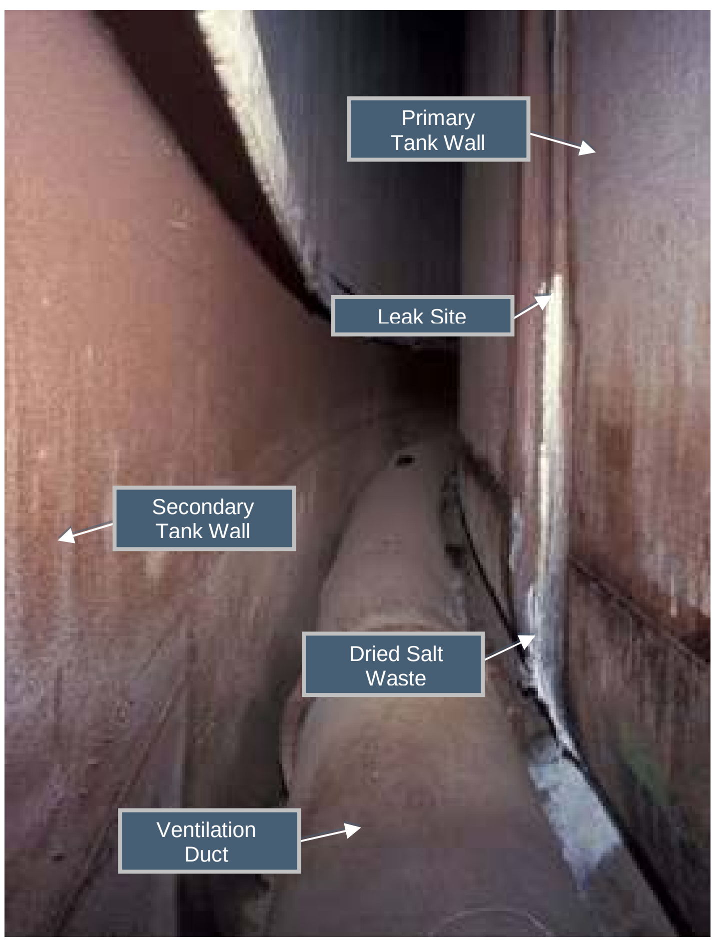
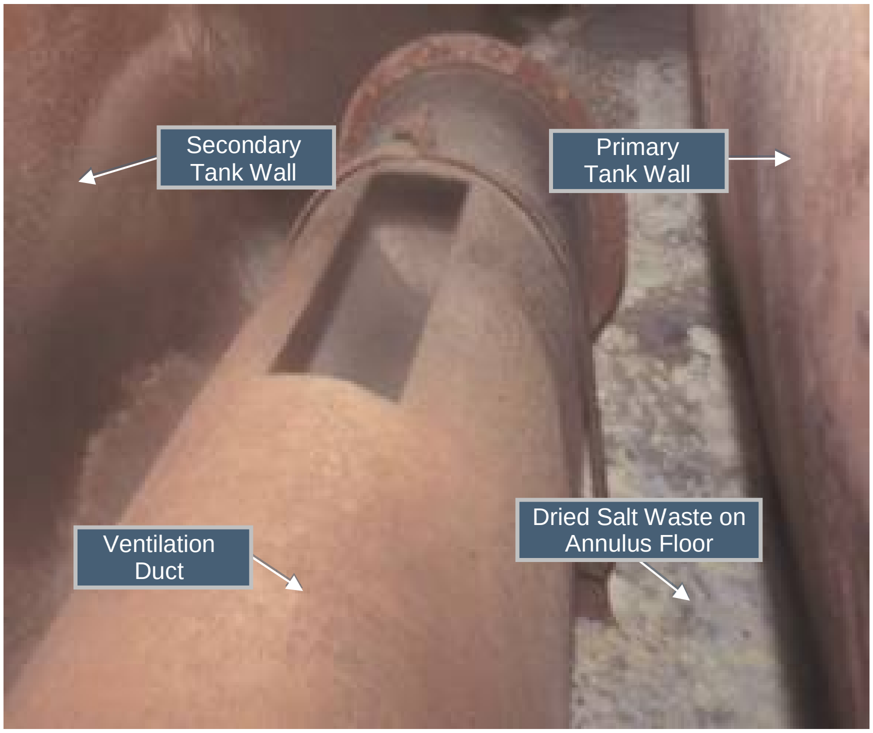


Figure 2.3-8: Waste Deposits on Annulus Floor Before Cleaning (Tank 6)



2.3.7 Potential Future Waste Tank Cleaning Technology

DOE will continue to review and consider technological developments relevant to waste tank cleaning and will evaluate technologies of comparable, or greater, effectiveness than those discussed above. A range of potential technologies for evaluation will potentially include proven technologies developed and/or used at other DOE sites, in domestic commercial industry and in international applications. Waste tank cleaning technologies that will potentially be evaluated include, but are not necessarily limited to, sluicing, mixing, chemical cleaning, vacuum retrieval techniques, mechanical manipulators, robotic devices and processes that remove (chemically extract) the radionuclides from the residual material that may remain in the tank.

2.3.8 Ancillary Structures Cleaning

Ancillary structures are described in Section 2.1.12.

Flushing the transfer lines after use has long been practiced for waste transfers to prevent material build up within the systems. Transfer line core pipe flushing has been part of operations of the tank farms from at least the mid-1970s, and there is also indication that some level of flushing has always been a part of transfer system operations. The rigor to which flushing has been applied has increased over the years. [WSRC-TR-2002-00403]

In addition to operational practices of flushing, specific design practices have contributed to removing the waste from waste transfer line piping systems. The installation of stainless steel for the waste transfer core piping, the transfer piping sloping toward a waste tank with minimal valves and the layout of turn radii are specific design features that prevent waste accumulation in the piping systems.

Secondary containment (e.g., transfer line jackets, leak detection box encasements) is provided for transfer line core pipes. No leakage of waste from primary core pipes into secondary containment has been identified. Transfer lines currently in service, which make up approximately 57% of the total linear footage of lines, are tested as part of the Structural Integrity Program. [S-TSR-G-00001] Most of the remaining lines that are no longer in service are part of the 242-F Evaporator System and the Tank 17 through 20 system. The structural integrity air or helium testing of transfer lines procedurally requires radiological surveys to check for indications of contamination. [SW10.6-SVP-5] This testing has not identified any significant contamination in FTF transfer line secondary containment. Therefore, no residual waste is expected in these structures at the time of closure.

As described in Section 2.1.12, there are two evaporator systems in FTF, the 242-F Evaporator System and the 242-16F Evaporator System. The 242-F evaporator and associated concentrate transfer system were shut down in 1988. Various waste removal campaigns were completed in 1991, 1992 and again in 2004. During the 2004 waste removal campaign, various mixing and transfer cycles were performed in the evaporator vessel and concentrate transfer system waste tank. Following the 2004 waste removal campaign, the evaporator vessel and the concentrate transfer system tank were inspected and the contents were sampled and characterized. [CBU-LTS-2004-00078]

The 242-16F evaporator cell and vessel remain operational. The final residual levels in the 242-16F Evaporator System are expected to be comparable to the residual levels achieved in the 242-F Evaporator System due to similar design, operational histories, and comparable cleaning techniques.

There are three stainless steel pump tanks in FTF: The FPT-1 serves as the inter-area pump tank, and both FPT-2 and FPT-3 previously received waste transfers from F Canyon. There is a single catch tank in FTF designed to collect drainage from FDB-1 and the Type I tank transfer line encasements. These stainless steel tanks are accessible for waste removal with existing technologies.

Prior to closure, secondary containment structures (e.g., pump pits, diversion boxes, leak detection boxes, modified leak detection boxes and valve boxes) will be inspected and exposure to waste over their service life determined. The pump pits are shielded reinforced concrete structures located below grade at the low points of transfer lines and are lined with stainless steel. These structures are secondary containments that house the pump tanks. The diversion boxes are shielded reinforced concrete structures containing transfer line nozzles to which jumpers are connected to direct waste transfers to the desired location. The majority of the diversion boxes are located below ground and are either stainless steel-lined or sealed with waterproofing compounds. Valve boxes provide secondary containment for valves and transfer line jumpers to facilitate specific waste transfers. The valves are generally manual ball valves in removable jumpers with flush water connections on the transfer piping. Since these structures do not provide primary containment, they are expected to contain only minimal radionuclide contamination at the time of closure.

2.4 Radionuclide Inventory in F-Tank Farm Facility Systems, Structures and Components

Minimal quantities of residual material are expected to remain in FTF at the time of closure following waste tank and ancillary structures cleaning activities. Data developed for the FTF PA, Revision 1[18], provided the estimated FTF residual radionuclide inventory used in this Draft FTF 3116 Basis Document. [SRS-REG-2007-00002]

Estimated radionuclide concentrations for residual material in FTF waste tanks at closure are determined by three methods:

- sample analysis
- process knowledge data maintained in a controlled database (WCS)
- special analysis (SA)

The inventory of each waste tank was used to establish the characterization of the residual material in the ancillary structures and was decayed to year 2020[19]. The following subsections describe the process for estimating residual material inventory at closure for the FTF waste tanks and ancillary structures, as provided in the FTF PA, Revision 1. [SRS-REG-2007-00002]

[18] Unless otherwise noted, the information in Section 2.4 and subsequent Sections 2.5 and 2.6 is based on information developed in the FTF PA, Revision 1. As required by DOE Manual 435.1-1, maintenance of the FTF PA will include future updates to incorporate new information, update model codes, analysis of actual residual inventories, etc., as appropriate. [SRS-REG-2007-00002]

[19] The year 2020 corresponds to the year the FTF PA assumes, for the purposes of analysis, that a 100-year period of institutional control will begin.

A methodical approach was used to construct reasonably conservative estimates of FTF waste tank closure inventories to be used in PA modeling. Independent steps were developed to systematically construct the FTF tank inventories, with each step adjusting inventory either by tank or by radionuclide. The steps used in inventory development were as follows: [SRS-REG-2007-00002]

1. The initial list of radionuclides to be included in the FTF tank inventories was established. This list was developed beginning with an initial listing of 849 radionuclides compiled from a variety of published resources including those from Tables 1 and 2 of 10 CFR 61.55. [CBU-PIT-2005-00228]
2. The contaminant screening process consisted of several steps to arrive at an appropriate list of 54 isotopes to be included in the FTF waste tank closure inventory estimates to be used in the FTF PA modeling. This initial screening process considered the following information:
   - Physical properties of each radionuclide such as half-life and decay mechanism,
   - Waste source and handling based on radionuclide production mechanisms and time since the radionuclide was produced, and
   - Screening factors for radionuclide ground disposal developed in NCRP-123 which convert a quantity of each radionuclide to a dose.

   This initial screening reduced the radionuclide list from 849 down to 159 radionuclides. [CBU-PIT-2005-00228]

   Additional screening of the 159 radionuclides was performed to identify the radionuclides, from this list of 159, to be considered in the initial FTF inventory for the PA. The screening criteria included the following:

   - Radionuclides were screened out if there were no ancestors present from the specific decay chain or no decay source for the radionuclide.
   - Evaluation of FTF waste production history information for the potential for a specific radionuclide. This criterion screened out radionuclides not present within the FTF waste.
   - In general, radionuclides present due to ingrowth from a decay series were screened out, however, production history was used to retain those radionuclides present at a greater proportion than from the decay series. This criterion eliminated radionuclides that are present only due to the decay of their parent radionuclide. The inventory of these radionuclides can be controlled by removing the parent radionuclide(s).
   - Radionuclides with less than a five year half-life were screened out. This criterion reflects that active institutional control will be maintained over the site for 100 years after facility closure. The inventories of these radionuclides will be significantly diminished due to the amount of radioactive decay that will occur during the 100-year institutional control period.

   Based on these screening criteria, an additional 105 radionuclides were screened out, thus reducing the FTF PA modeling initial inventory radionuclide number to 54. The screening process is described in detail in Section 3.3 of the FTF PA. [SRS-REG-2007-00002]

3. Additional radionuclides were added to the list of radionuclides of concern based on the potential for activation products being present. These additions were based on the possibility that these radionuclides may have been created at various points of production in SRS history. These radionuclides are Cl-36, K-40, Nb-93m, Pd-107, Pt-193 and Zr-93. These six radionuclides were added to the list of radionuclides of concern at inventories equivalent to the sample analysis detection capability. Because minimal amounts of these radionuclides were created, setting the inventory for this radionuclides at the detection capability was reasonably conservative (approximated at 1.0E-03 Ci).
4. The list of radionuclides to be included was then reduced based on individual isotope half-life. Radionuclides with a half-life of less than five years were removed from inventory estimates. This was based on the consideration that active institutional control over the closed waste tanks will be maintained for 100 years. Therefore, any radionuclides with less than a five-year half-life would decrease to insignificant levels due to radioactive decay during the institutional control period. This does not apply to radionuclides that are part of a decay chain with a parent with a greater than five-year half-life. The twelve radionuclides removed from further inventory estimates based

on the five-year half-life criteria are as follows: Bk-249, Ce-144, Cm-242, Cs-134, Eu-155, Na-22, Pm-147, Pr-144, Rh-106, Ru-106, Sb-126 and Te-125m.

5. The radionuclides inventories developed for Revision 0 of FTF PA [SRS-REG-2007-00002] were used as a starting point for the inventory of each individual waste tank. The WCS database [DOE-WD-2005-001] was utilized in developing the FTF PA Revision 0 inventories. The WCS is an electronic information system that tracks waste tank data, including projected radionuclide inventories based on sample analyses, process histories, composition studies and theoretical relationships. The WCS, initially developed in 1995, and populated with historical information, tracks, among other things, the dry sludge concentrations of radionuclides in each of the SRS waste tanks. The primary purpose for developing WCS was to provide reasonable estimates on which to base safety analysis evaluations such as criticality issues in the tanks farms. Safety analysis evaluations, in general, build in a degree of conservatism, consequently, a level of conservatism for some materials is reflected in the data. The radionuclides tracked in the WCS were selected primarily based on their impact on waste tank safety basis source term, inhalation dose potential or on the E-Area Vault waste acceptance criteria (WAC).
6. The waste tanks were binned according to waste tank type. The tank type generally had an effect on the type of waste received and therefore guided the group selection. In general, each waste tank was built at approximately the same time as others of the same type. In addition, the tanks built within a specific time frame were built in close proximity to each other.
7. Within each bin, the inventories were adjusted as applicable within that bin.

Type I: An adjustment to the inventory estimates was made by multiplying by a factor of 10 to provide for greater uncertainties in the residual inventories from the planned waste removal operation. Based on recent waste removal activities in Tanks 5 and 6, there is uncertainty around the projected inventories for the waste tanks remaining to be cleaned. To account for this uncertainty and ensure that the PA would provide a reasonably conservative inventory projection, the existing inventories were multiplied by a factor of 10.

For a majority of the radionuclides with an adjusted inventory less than one curie, the inventories were adjusted to either one curie or the analytical detection limit (1.0E-03 Ci) to allow for more efficient and cost effective means of confirming concentrations within residual materials for radionuclides with a limited potential impact to dose. For those radionuclides that have been observed (through previous analyses or scoping studies) to have greater potential impact on the overall dose, the inventory was adjusted to the analytical detection limit. The following radionuclides fit into this category: Ac-227, Cm-247, Cm-248, I-129, Nb-94, Pa-231 and Pu-244. Those radionuclides with adjusted inventories greater than one curie were not adjusted in this step.

The inventory estimates for both Ra-226 and Th-230 were revised from the overly conservative estimates included in Revision 0 of the FTF PA. The initial Revision 0 estimates assumed that these radionuclides were in full transient equilibrium with the parent U-234. This assumption is extremely conservative; therefore, a more reasonable estimate was made assuming a decay ingrowth period of 70 years. Although this estimate was more reasonable, the resulting inventory value fell below the detection level capability. Therefore, the inventory estimate for both Ra-226 and Th-230 were adjusted to the minimum detection level.

Within the Type I waste tank group, the maximum waste tank inventory for any one tank was used to estimate inventory for the other waste tanks within the grouping. As each waste tank is cleaned, the waste removed from within the tank during waste removal will be transferred to other tanks. Due to transfer line configuration, the material in one tank will typically pass through other tanks of the same type prior to exiting the FTF. Due to the uncertain order of tank waste removal and closure activities, the maximum inventory concentrations associated with an individual tank within a tank group were applied to the other tanks within the bin. The maximum adjusted inventory, to this point, was used for all tanks within the Type I waste tank group.

Type IV: Tanks 17 and 20 have been removed from service and grouted. Following waste removal activities, the residual material within Tanks 17 and 20 was sampled and analyzed.

This analysis provided the basis for estimates of the residual material. The Tanks 17 and 20 residual estimates were used exclusively; no adjustments were made. The Tanks 18 and 19 inventory estimates are based on concentrations documented in residual characterization reports with the total final inventories adjusted to reflect the bounding waste residual estimates (i.e., 5,000 gallon for Tank 18 and 3,000 gallon for Tank 19) at the time. The final inventory estimates for Tanks 18 and 19 apply the bounding waste residual floor estimates while allowing the corrosion product estimates to remain unchanged from Revision 0 of the FTF PA.

Calculating the inventory contained within the corrosion products (excluding contaminant diffusion into the steel) required an estimate of the K_d value used for estimating the uranium content in the tank walls' corrosion products. This value was 600 L/kg. An evaluation was performed to determine the appropriate value to use for the SRS tanks not cleaned with oxalic acid. Due to the high pH environment present in the waste tanks, high ionic strength and moderately high carbonate concentrations, the adsorption of the uranium ion to the rust surfaces was expected to be two or three orders of magnitude less than the 6,000 L/kg value previously used. In the absence of additional data, a decrease of only one order of magnitude was used to provide a conservative uranium inventory in the corrosion products.

Diffusion of contaminates into the steel was considered negligible. A comparison of the amount of material diffused into the transfer lines to the amount of material estimated in the corrosion products was performed. This comparison demonstrated that the estimated quantity of material in the corrosion products exceeded that which diffused into the steel. Therefore there was no estimate of material diffused into the steel.

For a majority of the radionuclides with an adjusted inventory less than one curie, their inventories were adjusted to either one curie or the analytical detection limit (1.0E-03 Ci). This allows for more efficient and cost effective means of confirming concentrations within residual materials for radionuclides with a limited potential impact to dose. For those radionuclides that have been observed (through previous analyses or scoping studies) to have greater potential impact on the overall dose, the inventory was adjusted to the analytical detection limit. The following radionuclides fit into this category: Ac-227, Cm-247, Cm-248, I-129, Nb-94, Pa-231 and Pu-244. Those radionuclides with adjusted inventories greater than one curie were not adjusted in this step.

Based on recent preliminary sample results, the estimates for Am-243, Cs-135 and Np-237 (Tank 18 only) were further adjusted higher to ensure the estimates used are reasonably conservative.

The inventory estimates for both Ra-226 and Th-230 were revised from the overly conservative estimates included in Revision 0 of the FTF PA. The initial Revision 0 estimates assumed that these radionuclides were in full transient equilibrium with the parent U-234. This assumption is extremely conservative; therefore, a more reasonable estimate was made assuming a decay ingrowth period of 70 years. Although this estimate was more reasonable, the resulting inventory value fell below the detection level capability. Therefore, the inventory estimate for both Ra-226 and Th-230 were adjusted to the minimum detection level.

Type IIIA: The Type IIIA tank grouping consists of Tanks 25 through 28 and Tanks 44 through 47. An adjustment to the inventory estimates was made by multiplying by a factor of 10 to provide for greater uncertainties in the residual inventories from the planned waste removal operation. Based on recent waste removal activities in Tanks 5 and 6, there is uncertainty around the projected inventories for the waste tanks remaining to be cleaned. To account for this uncertainty and ensure that the PA would provide a reasonable bounding inventory projection, the existing inventories were multiplied by a factor of 10.

For a majority of the radionuclides with an adjusted inventory less than one curie, the inventories were adjusted to either one curie or the analytical detection limit (1.0E-03 Ci). This allows more efficient and cost effective means of confirming concentrations within residual materials for radionuclides with a limited potential impact to dose. For those radionuclides that have been observed (through previous analyses or scoping studies) to have greater potential impact on the overall dose, the inventory was adjusted to the analytical detection limit. The following radionuclides fit into this category: Ac-227, Cm-247, Cm-248, I-129, Nb-94, Pa-231 and Pu-244. Those radionuclides with adjusted inventories greater than one curie were not adjusted in this step.

The inventory estimates for both Ra-226 and Th-230 were revised from the overly conservative estimates included in Revision 0 of the FTF PA. The initial Revision 0 estimates assumed that these radionuclides were in full transient equilibrium with the parent U-234. This assumption is extremely conservative; therefore a more reasonable estimate was made assuming a decay ingrowth period of 70 years. Although this estimate was more reasonable, the resulting inventory value fell below the detection level capability. Therefore, the inventory estimate for both Ra-226 and Th-230 were adjusted to the minimum detection level.

Within the Type IIIA waste tank group, the maximum waste tank inventory for any one tank was used to estimate inventory for the other waste tanks within the grouping. As each waste tank is cleaned, the waste removed from within the tank during waste removal will be transferred to other tanks. In most case, this transfer will be to another tank of the same type. So the material in one tank will typically pass through other tanks of the same type prior to exiting the FTF. Due to the uncertain order of tank waste removal and closure activities, the maximum inventory concentrations associated with an individual tank within a tank group were applied to the other tanks within the bin. The maximum adjusted inventory, to this point, was used for all tanks within the Type IIIA waste tank group.

Type III: The Type III tank grouping consists of Tanks 33 and 34. For a majority of the radionuclides with an adjusted inventory less than one curie, their inventories were adjusted to either one curie or the analytical detection limit (1.0E-03 Ci). This allows more efficient and cost effective means of confirming concentrations within residual materials for radionuclides with a limited potential impact to dose. For those radionuclides that have been observed (through previous analyses or scoping studies) to have greater potential impact on the overall dose, the inventory was adjusted to the analytical detection limit. The following radionuclides fit into this category: Ac-227, Cm-247, Cm-248, I-129, Nb-94, Pa-231 and Pu-244. Those radionuclides with adjusted inventories greater than one curie were not adjusted in this step.

The inventory estimates for both Ra-226 and Th-230 were revised from the overly conservative estimates included in Revision 0 of the FTF PA. The initial Revision 0 estimates assumed that these radionuclides were in full transient equilibrium with the parent U-234. This assumption is extremely conservative; therefore a more reasonable estimate was made assuming a decay ingrowth period of 70 years. Although this estimate was more reasonable, the resulting inventory value fell below the detection level capability. Therefore, the inventory estimate for both Ra-226 and Th-230 were adjusted to the minimum detection. [SRS-REG-2007-00002]

At the completion of waste removal for each of the tanks, the estimated inventory identified for that tank in the FTF PA will be compared and evaluated against the actual inventory measured after the tank has been cleaned. [SRS-REG-2007-00002] The inventory will be developed from a determination of the residual material volume combined with analytical data from a statistically based sampling program of the residual material.

2.4.1 Residual Inventory for Tank Annuli, Inside Failed Cooling Coils and Internal Tank Surfaces

The inventories in the annulus, inside failed cooling coils and on the surface of the waste tank walls, cooling coils and columns are assumed to be covered by the estimated total tank inventories. The material in the annulus is expected to be dried salt that can be readily dissolved and effectively removed when the annular regions are cleaned during closure preparation. At the time of closure, radioactive contamination will remain on the wall and floor of the annulus. Cooling coils with the potential for residual waste holdup will be evaluated and flushed as appropriate. Flushing is expected to remove residual waste that may have entered damaged coils. The volume of cooling coils represents less than 1% of the entire waste tank volume. [SRS-REG-2007-00002]

When a solution is in contact with a solid phase, the constituents may partition between the liquid and solid. In the waste tank, liquid waste is in contact with corrosion products on the tank interior wall, so small quantities of radioactive material could be held on the corrosion products. [SRT-WPT-2005-00049] The surfaces of waste tank internal walls, cooling coils and support columns are not expected to contain significant deposits based on inspections performed to date to support mechanical heel removal activities.

2.4.2 Residual Waste Inventory for Transfer Lines, Pump Tanks and Catch Tanks and Evaporator Systems

Ancillary structures include transfer lines, transfer line secondary containment, pump tanks, pump pits, the FTF catch tank, diversion boxes, valve boxes and the evaporator systems. Over the operating life of the facility, radioactive waste comes into physical contact with some of these components, contaminating them and hence, leaves contamination on the components. The degree of contamination depends on many factors, which include, but are not limited to, the service life of the component, the material of construction and the type of waste in contact with the component. Some of the listed equipment serves only as secondary containment and may not have contacted the waste. [SRS-REG-2007-00002]

Ancillary structures inventories are estimated for the following three categories:

- transfer lines,
- pump tanks and catch tank, and
- evaporators and concentrate transfer system tank.

The inventory of each waste tank is used to establish the characterization of the residual material in the ancillary structures, as described in the FTF PA. The results of a review of waste transfers within FTF and between FTF and HTF have been sorted to determine the percentage of the volume of all waste transfers that can be attributed to each FTF waste tank. The representative concentration was then determined by applying a weighted average to each isotopic distribution in the FTF waste tanks. The characterization of dry sludge was used for each waste tank for conservatism. It is important to note that while the sludge concentrations were used, dry sludge is only a small portion of the total waste that passes through the transfer lines that are routinely flushed with a high volume of supernate. Using the dry sludge concentrations provides a conservative representation of the actinides and long-lived isotopes. The short-lived isotopes, which are more concentrated in the supernate than the sludge, will have decayed significantly during the 100 year active institutional control period and are not expected to be as significant to an inadvertent intruder as the long-lived isotopes and actinides. [SRS-REG-2007-00002]

2.5 Residual Waste Stabilization

In May 2002, DOE issued an Environmental Impact Statement (EIS) on waste tank cleaning and stabilization alternatives. DOE studied five alternatives:

- empty, clean and fill tank with grout,
- empty, clean and fill tank with sand,
- empty, clean and fill tank with saltstone,
- clean and remove tanks, and
- no action.

The EIS concluded the Fill with Grout option was preferred. DOE also issued a Record of Decision (ROD) selecting the Fill with Grout alternative for SRS waste tank closure. [DOE-EIS-0303 ROD, SRS-REG-2007-00002]

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 waste tanks. This alternative offers several advantages over the other alternatives evaluated such as the following:

- provides greater long-term stability of the tanks and their stabilized contaminants than the sand-fill approach,
- provides for retaining radionuclide 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, and
- avoids the excessive personnel radiation exposure and greater occupational safety impact that would be associated with the clean and remove alternative.

Cementitious materials are often used to stabilize radioactive wastes. Grout has been one of the most commonly used materials for solidifying and stabilizing radioactive wastes and the technology is at a mature stage of development. Grout is a mixture of primarily cement and water proportioned to produce a pourable consistency. Stabilization is needed to maintain the waste tank structure and minimize water infiltration over an extended period of time, thereby impeding release of stabilized contaminants into the environment.

The waste tank fill grout will be reducing grout, which has low reduction potential (E_h) minimizing the mobility of the radionuclides after closure. All grout formulas are alkaline because grout is a cement-based material that naturally has a high pH, which is compatible with the tank carbon steel. The tank fill grout will have high compressive strength and low permeability, which enhances its ability to limit the migration of contaminants after closure. The grout formulas must be fluid to allow a near level placement. [SRS-REG-2007-00002]

The grout properties studied consisted of two major states, cured and fresh. [WSRC-STI-2007-00369] The major requirements for cured properties of grout include compressive strength, saturated hydraulic conductivity, porosity and dry bulk density. The fresh grout properties include flow, bleed water, set time, air content and wet unit weight (density). [WSRC-STI-2007-00641]

The fluidity of the mixtures is one of the main requirements. Grout requirements consist of both mechanical and chemical properties. The mechanical requirements of the grout consist of adequate compressive strength to withstand the overburden load and provide a physical barrier to discourage intruders. The chemical requirements of grout include a high pH and a low E_h to create an environment that makes contaminants less soluble and less mobile. Grout with a high pH can accept protons to lower the pH of contaminants to make them less soluble. Grout with low E_h has the tendency to donate electrons and thereby reduce contaminants to make them less mobile.

Reducing grout will be used to fill the Type I, III and IIIA waste tanks. In the Type IV tanks, reducing grout will be used to fill the waste tank volume and the tank dome will be filled with a strong grout (i.e., a grout with compressive strength properties in excess of 2,000 psi). However, if the final reducing grout recipe provides equivalent compressive strength (2,000 psi minimum) as the strong grout, then only reducing grout will be used for the Type IV waste tanks.

For waste tank types with cooling coils and annulus, the cooling coils and annulus will be grouted to minimize void spaces, to minimize fast flow pathways and for stability. Annulus risers and ductwork will be filled with grout up to grade level and closed and capped.

Various pieces of equipment will remain in the waste tanks at the time of closure. This equipment consists of items such as transfer jets, thermowells, level instrumentation, a leak detection system, transfer piping out of the waste tank and equipment directly used in tank closure operations (such as submersible mixers and pumps, cables, temporary transfer hoses). These various pieces of equipment, in both the primary tanks and the annulus, will be grouted to the extent practical. In addition, steel tapes and other miscellaneous debris will remain in the waste tank after closure. These components will be entombed in the grout as part of the closure process.

With the exception of the transfer lines, ancillary structures (e.g., diversion boxes, pump tanks, pump pits) associated with FTF will be filled, as necessary, at final closure to prevent subsidence. The waste transfer lines were modeled in the FTF PA with no grout and the result was in compliance with the required performance objectives. Since any residual radioactive waste would be on the interior wall of the transfer lines and the leach rate would not be significantly influenced by grout, there is no environmental benefit to grouting these small diameter transfer lines. [SRS-REG-2007-00002]

2.6 Closure Cap

An engineered closure cap will be installed over the FTF following the closure of the waste tanks and ancillary equipment. The closure cap description is based on a recent SRNL report on the FTF closure cap concept and estimated initial infiltration presented in WSRC-STI-2007-00184. The design information being provided is for planning purposes sufficient to support evaluation of the closure cap as part of the integrated site conceptual model being evaluated in the FTF PA. The closure cap design will be finalized closer to the time of FTF closure, to take advantage of possible advances in materials and closure cap technology that could be used to improve the design.

A detailed discussion on closure cap design is provided in Section 3.0 of the FTF PA. An overview of the general design features and figures from the FTF PA are provided below. [SRS-REG-2007-00002]

Figure 2.6-1 presents the general design of the closure cap above a closed waste tank. Figure 2.6-2 presents the closure cap footprint. Figure 2.6-3 and Figure 2.6-4 present cross sections of the closure cap conceptual design. Figure 2.6-5 depicts the finished closure cap. Figure 2.6-6 presents the elevations of the closure cap surfaces and the grading plan.

Figure 2.6-1: FTF Closure Cap General Concept

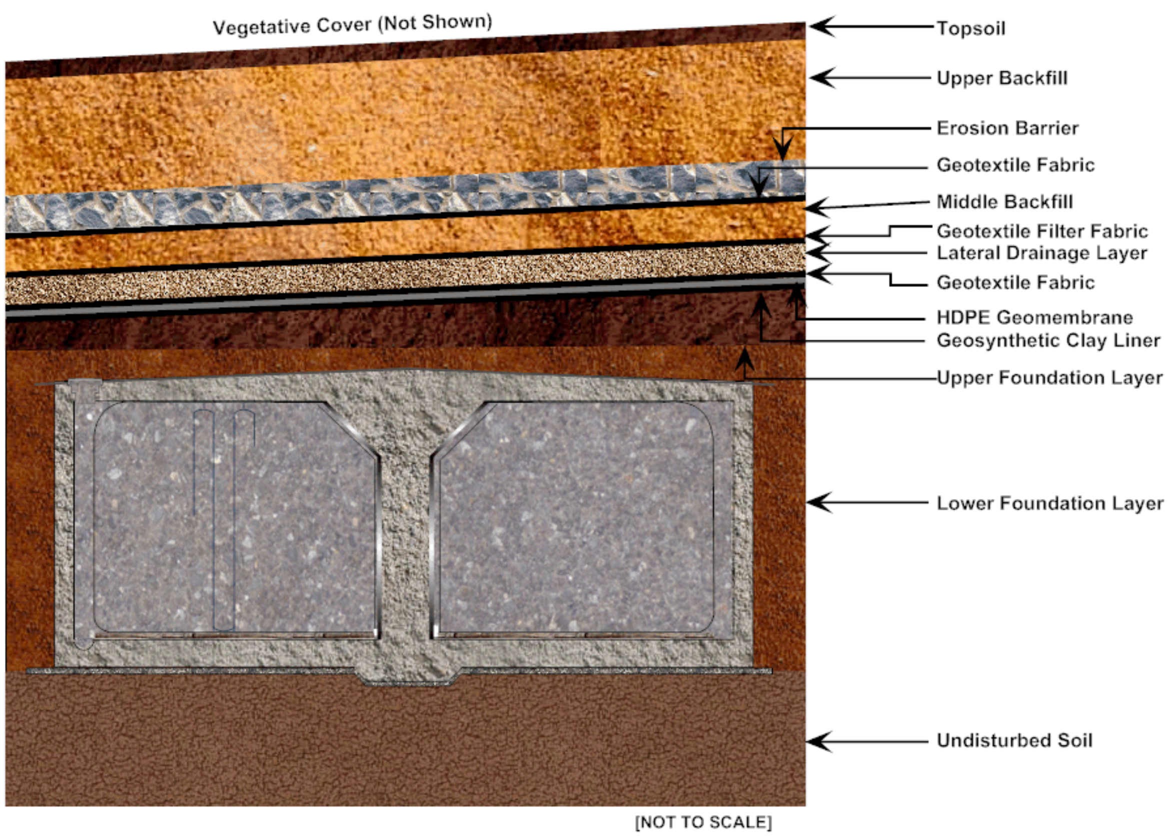
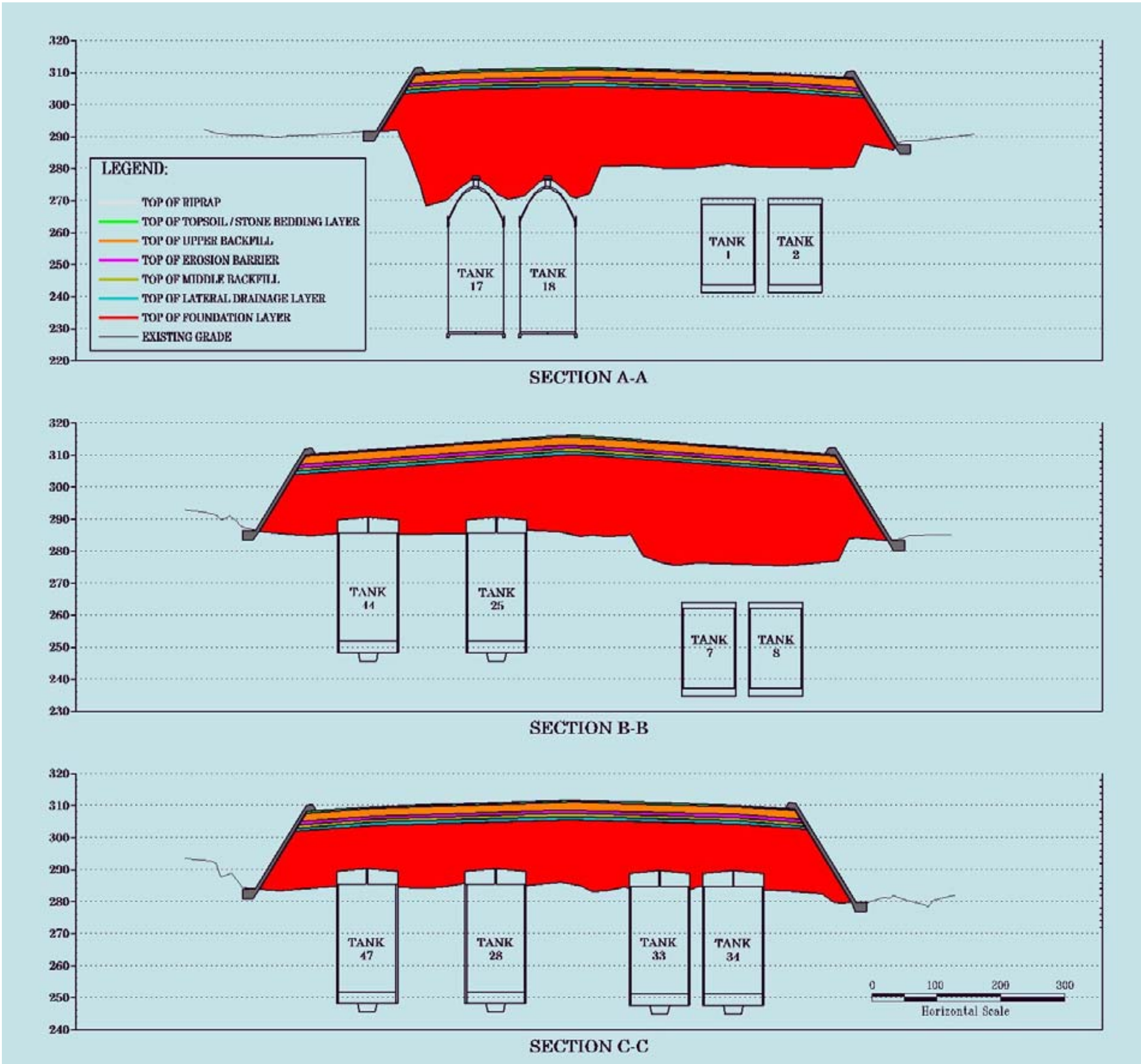


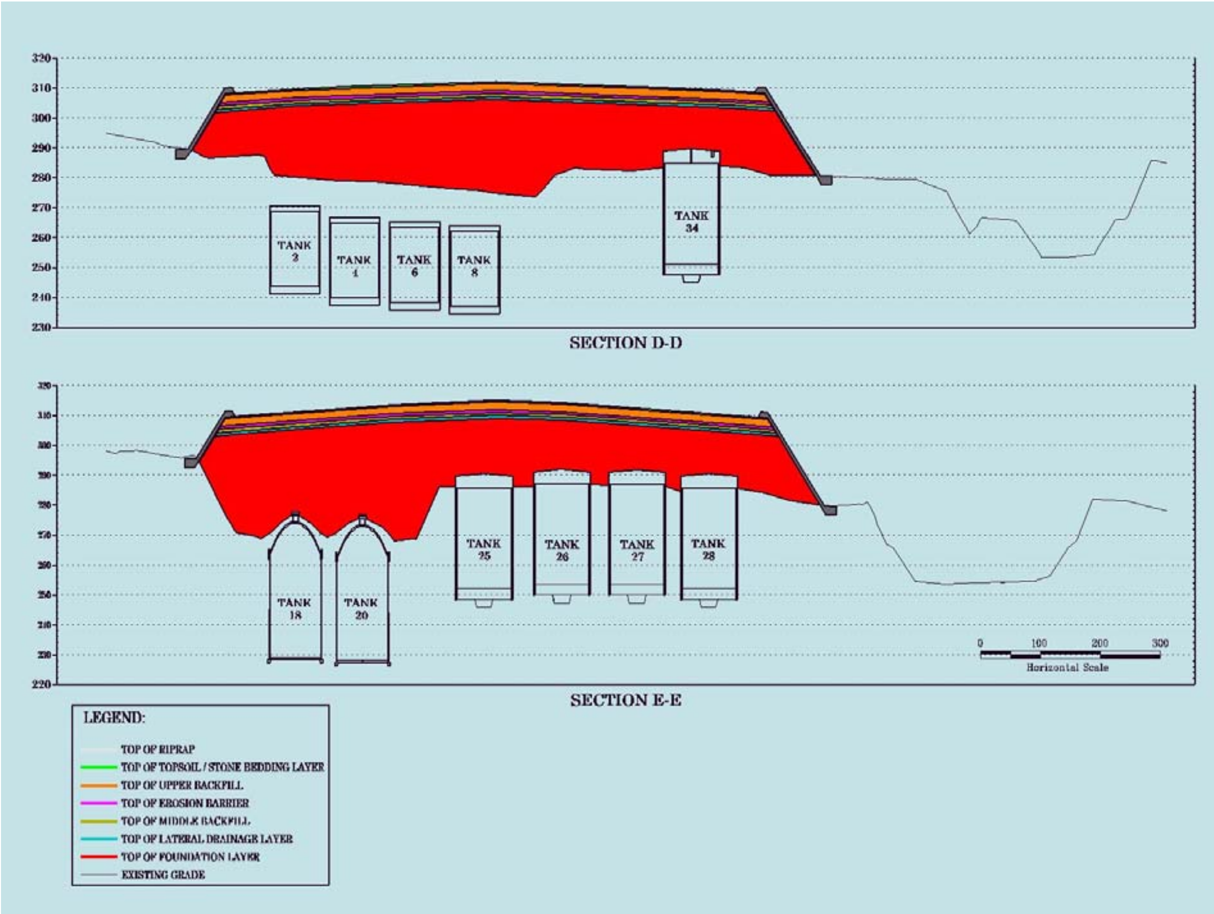
Figure 2.6-2: FTF Closure Cap Conceptual Design, Cap Footprint

Figure 2.6-3: FTF Closure Cap Conceptual Design, Sections A-A, B-B, and C-C



NOTE: Vertical scale of sections has been exaggerated 5 times in order to show all closure cap layers.

Figure 2.6-4: FTF Closure Cap Conceptual Design, Sections D-D and E-E



NOTE: Vertical scale of sections has been exaggerated 5 times in order to show all closure cap layers.

Figure 2.6-5: FTF Closure Cap Conceptual Design, Finished Cap

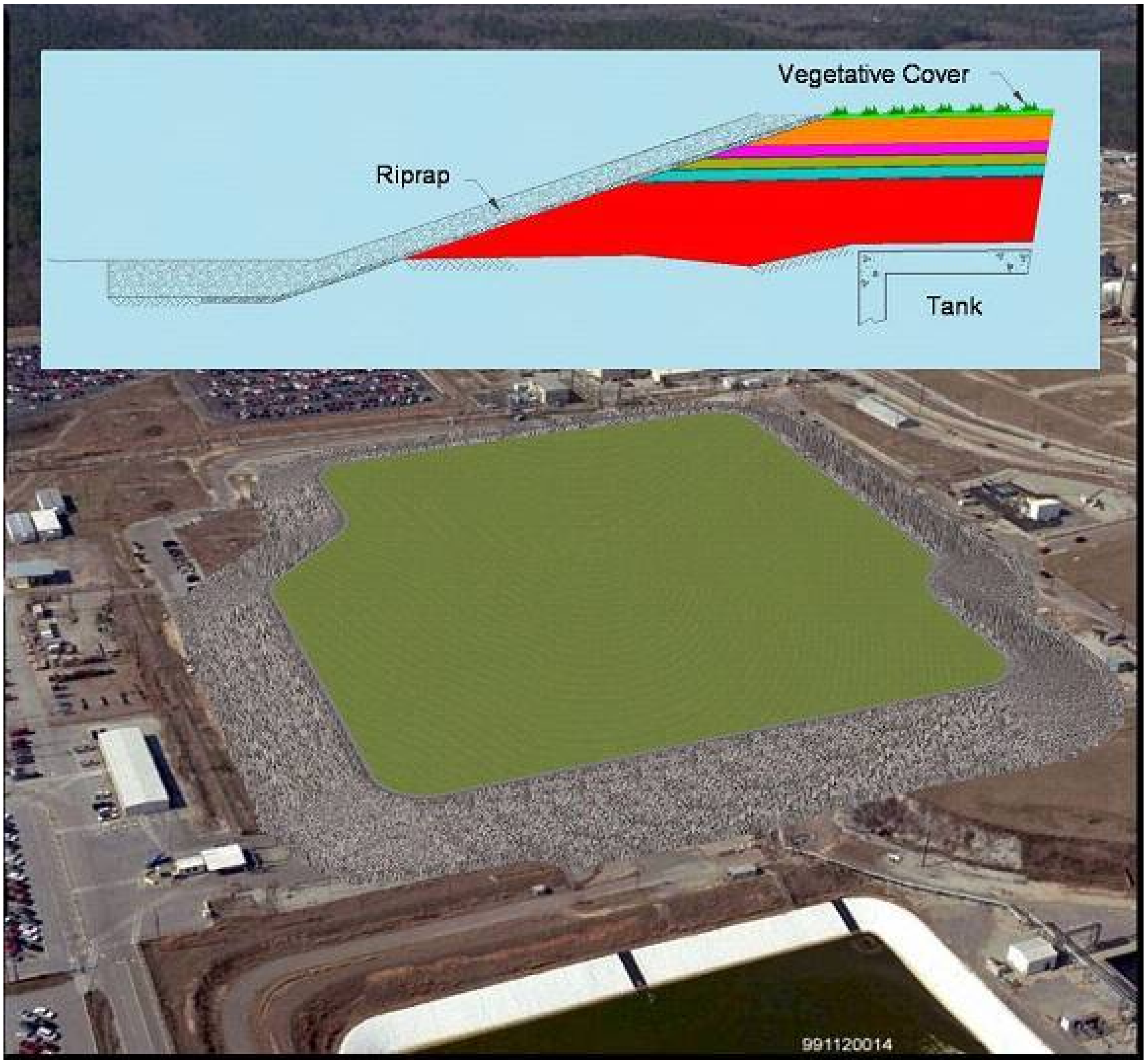
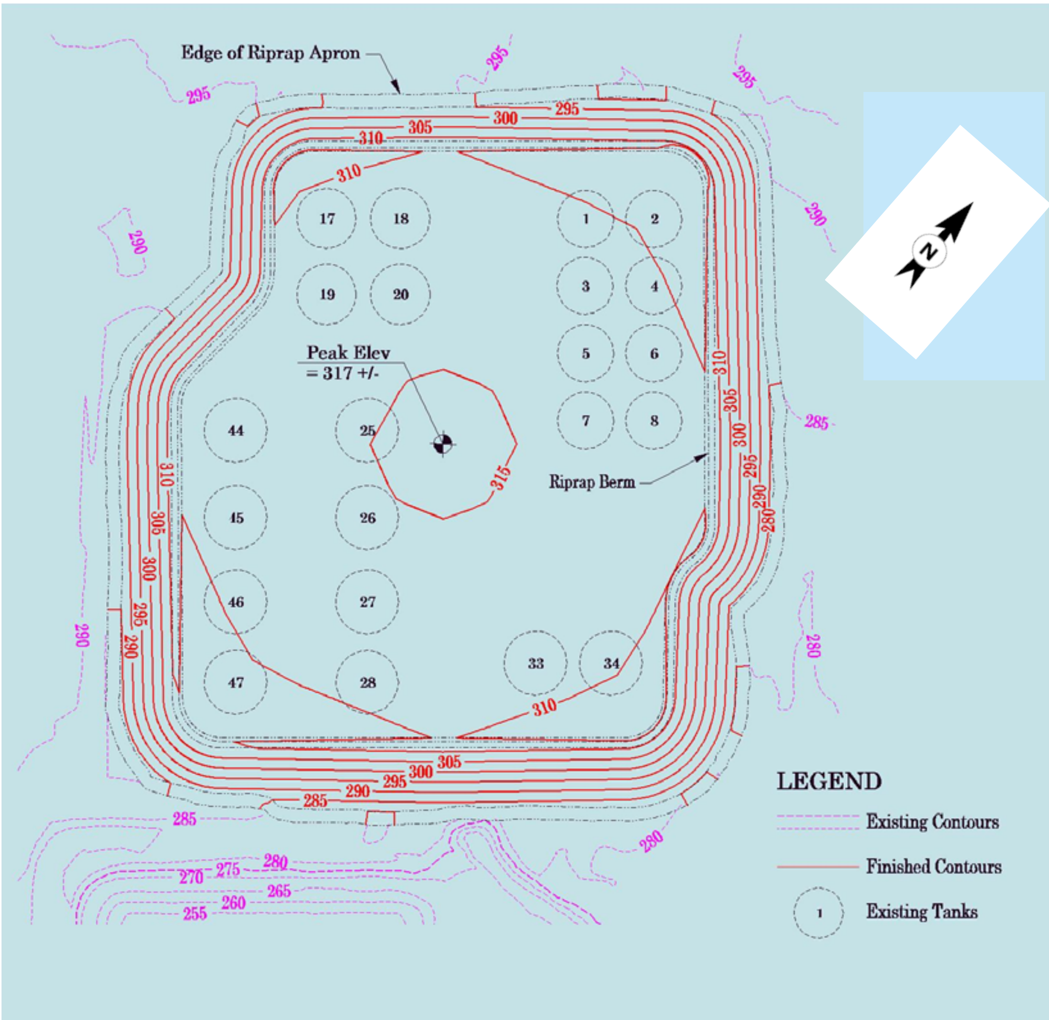


Figure 2.6-6: FTF Closure Cap Conceptual Design, Grading Cap



2.6.1 Function of Closure Cap Layers

It is anticipated that the FTF closure cap will consist of the layers illustrated in Figure 2.6-1 and identified in Table 2.6-1 from top to bottom. Table 2.6-2 summarizes the function of each of these layers. Detailed discussion of each layer in the FTF closure cap concept is provided in WSRC-STI-2007-00184. The concepts for the side slopes and toes of the closure cap based upon the results of physical stability calculations referred to above are also detailed in WSRC-STI-2007-00184.

Table 2.6-1: FTF Closure Cap Layers

Layer	Layer Thickness (in)
Vegetative Cover	NA
Topsoil	6
Upper Backfill	30
Erosion Barrier	12
Geotextile Fabric	-
Middle Backfill	12
Geotextile Filter Fabric	-
Lateral Drainage Layer	12
Geotextile Fabric	-
HDPE Geomembrane	0.06 (60 mil)
Geosynthetic Clay Liner (GCL)	0.2
Upper Foundation Layer	12
Lower Foundation Layer	72 (minimum)

\- Negligible

NA - Not Applicable
[WSRC-STI-2007-00184, Table 11]

Table 2.6-2: Function of the FTF Closure Cap Layers

Layer	Function
Vegetative Cover	The vegetative cover will promote runoff, minimize erosion, and promote evapotranspiration. The initial vegetative cover will be a persistent grass such as Bahia. If it is determined that bamboo is a climax species that prevents or greatly slows the intrusion of pine trees, bamboo will be planted as the final vegetative cover at the end of the 100-year institutional control period. Bamboo is not assumed in present design calculations and modeling
Topsoil	The topsoil is designed to support a vegetative cover, promote runoff, prevent the initiation of gullying, and provide water storage for the promotion of evapotranspiration.
Upper Backfill	The upper backfill is designed to increase the elevation of the closure cap to that necessary for placement of the topsoil and to provide water storage for the promotion of evapotranspiration.
Erosion Barrier	The erosion barrier is designed to prevent riprap movement during a PMP event and therefore form a barrier to further erosion and gully formation (i.e., provide closure cap physical stability). It is used to maintain a minimum 10 feet of clean material above the tanks and significant ancillary equipment to act as an intruder deterrent. It also provides minimal water storage for the promotion of evapotranspiration.
Geotextile Fabric	This geotextile fabric is designed to prevent the penetration of erosion barrier stone into the underlying middle backfill and to prevent piping of the middle backfill through the erosion barrier voids.
Middle Backfill	The middle backfill provides water storage for the promotion of evapotranspiration in the event that the topsoil and upper backfill are eroded away since the overlying erosion barrier provides only minimal water storage.
Geotextile Filter Fabric	This geotextile fabric is designed to provide filtration between the overlying middle backfill layer and the underlying lateral drainage layer. This filtration allows water to freely flow from the middle backfill to the lateral drainage layer while preventing the migration of soil from the middle backfill to the lateral drainage layer.
Lateral Drainage Layer	The lateral drainage layer is a coarse sand layer designed to: • divert infiltrating water away from the underlying tanks and ancillary equipment and transport the water to the perimeter drainage system, in conjunction with the underlying composite hydraulic barrier (i.e., HDPE geomembrane and GCL) and • provide the necessary confining pressures to allow the underlying GCL to hydrate properly.
Geotextile Fabric	This geotextile fabric is a nonwoven geotextile fabric designed to protect the underlying HDPE geomembrane from puncture or tear during placement of the overlying lateral drainage layer.
HDPE Geomembrane	The HDPE geomembrane forms a composite hydraulic barrier in conjunction with the GCL. The composite hydraulic barrier is designed to promote lateral drainage through the overlying lateral drainage layer and minimize infiltration to the tanks and ancillary equipment.
GCL	The GCL forms a composite hydraulic barrier described above in conjunction with the HDPE geomembrane. As part of the composite hydraulic barrier the GCL is designed to hydraulically plug any holes that may develop in the HDPE geomembrane.
Upper Foundation Layer Lower Foundation Layer	The foundation layers are designed to: • provide structural support for the rest of the overlying closure cap, • produce the required contours and produce a slope of 2% for the overlying layers, • produce the maximum 3:1 side slopes of the closure cap, • provide a suitable surface for installation of the GCL (i.e., a soil with a moderately low permeability and a smooth surface, free from deleterious materials), • promote drainage of infiltrating water away from and around the tanks and ancillary equipment, and • contain utilities, equipment, facilities, etc., that are not removed from above current grade prior to installation of the closure cap.

[WSRC-STI-2007-00184, Table 12]

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

The NDAA Section 3116(a) provides that radioactive waste that results from reprocessing is not "high-level radioactive waste" if the Secretary of Energy determines, in consultation with the NRC, that the waste meets certain specified criteria.

The NDAA Section 3116(a) provides in pertinent part:

> *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 Department of Energy 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;*
>
> *(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

The NDAA Section 3116(a) 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.*

Under NDAA Section 3116(a), certain wastes from reprocessing are not "high-level radioactive waste" if the Secretary of Energy, in consultation with the NRC, determines that certain criteria are met. The NDAA Section 3116(a) sets out three criteria. Criterion (2), which is set forth in NDAA Section 3116(a)(2), requires removal of highly radionuclides to the maximum extent practical. Criterion (3) generally mirrors the regulatory criteria that the NRC has established for determining whether waste qualifies for land disposal as low level waste. That criterion provides that disposal of the waste must meet the NRC performance objectives at 10 CFR Part 61, Subpart C, that the waste must not exceed concentration limits for Class C waste in 10 CFR 61.55 or must be disposed of pursuant to plans developed by the Secretary in consultation with NRC, and that disposal must be pursuant to a State-approved closure plan or permit. Criteria (2) and (3) will be discussed in subsequent sections of this Draft FTF 3116 Basis Document, which demonstrate that those criteria are satisfied.

Criterion (1), quoted above, is a broader criterion that allows the Secretary of Energy, in consultation with the NRC, to consider whether there are other considerations that, in the Secretary of Energy's judgment, warrant permanent isolation of the radioactive waste 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 of Energy has consulted with NRC concerning DOE's disposal plans[20]. Normally, 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 does not "require permanent isolation in a deep geologic repository" because non-repository disposal will be protective of public health and safety.

However, 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) of Section 3116(a) 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 the near- or intermediate-surface, nevertheless presents non-proliferation risks that the Secretary concludes cannot be adequately guarded against absent deep geologic disposal[21]. Criterion (1) allows the Secretary of Energy, in consultation with the NRC, to consider such factors in determining whether waste that meets the other two criteria may need permanent isolation in a deep geologic repository in light of these considerations.

This is not the case here. As demonstrated later in this document, the stabilized residuals within the waste tanks and ancillary structures, the waste tanks, and the ancillary structures (including integral

[20] As the NRC explained in *In the Matter of Louisiana Energy Services, L.P. (National Enrichment Services),* [CLI-05-05], 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."

[21] In NUREG-1854, *NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations*, the NRC similarly explains: "In general, there is reasonable assurance that this criterion can be met if the two other NDAA criteria can be met. In other words, if highly radioactive radionuclides have been removed to the maximum extent practical and the waste will be disposed of in compliance with the performance objectives in 10 CFR Part 61, Subpart C (which are the same performance objectives NRC uses for disposal of low-level waste), then this supports a conclusion that the waste does not require disposal in a deep geologic repository. However, this criterion allows for the consideration that waste may require disposal in a geologic repository even though the two other NDAA criteria may be met. Those circumstances under which geologic disposal is warranted to protect public health and safety and the environment could be considered; for example, unique radiological characteristics of waste or nonproliferation concerns for particular types of material."

equipment) located at FTF at the time of closure 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 the risk to public health and safety. Furthermore, the stabilized residuals within the waste tanks and ancillary structures, the waste tanks, and the ancillary structures (including integral equipment) do not raise any unique considerations that, notwithstanding these demonstrations, require permanent isolation in a deep geologic repository. Accordingly, the stabilized residuals within the waste tanks and ancillary structures, the waste tanks, and the ancillary structures (including integral equipment), meet the criterion of Clause (1) of NDAA Section 3116(a).

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

Section Purpose

The NDAA Section 3116(a) provides that certain waste resulting from reprocessing is not high-level waste if the Secretary of Energy, in consultation with the NRC, determines, among other things, that the waste has had highly radioactive radionuclides (HRRs) removed "to the maximum extent practical" (MEP). This section demonstrates that the FTF residual waste, tanks and ancillary structures, upon completion of waste removal activities at closure, will have had HRRs removed to the MEP and meet this criterion.

Section Contents

Section 5.1 identifies the HRRs for the purpose of this Draft FTF 3116 Basis Document. Section 5.2 describes the removal processes used to remove HRRs to the maximum extent practical. Section 5.3 demonstrates that, at closure, the HRRs will have been removed to the MEP.

Key Points

- The list of HRRs for FTF describes the radionuclides that could reasonably be expected to exist in the FTF waste tanks and ancillary structures and that, using a risk-informed approach, contribute significantly to the radiological risk to workers, the public and the environment, taking into account scientific principles, knowledge and expertise.
- The list of HRRs for FTF includes all radionuclides important to meeting the performance objectives in 10 CFR Part 61, Subpart C, and all radionuclides in Tables 1 and 2 of 10 CFR 61.55 were considered.
- Greater than 95% of the waste volume and radioactivity will have been removed from the FTF tanks and ancillary structures collectively prior to closure activities using mechanical, chemical or vacuum technologies that have previously been successfully demonstrated in similar waste removal activities at the SRS.

The NDAA Section 3116(a) 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 Methodology

Based on consultation with the 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. Strontium-90, Tc-99, I-129, Cs-137, U-234, Np-237, Pu-238, Pu-239, Pu-240 and Am-241 are the HRRs in the FTF stabilized residuals, FTF waste tanks and FTF ancillary structures at the closure of FTF that DOE has determined contribute significantly to radiological risk to workers, the public and the environment, taking into account scientific principles, knowledge and expertise[22].

[22] Some of the radionuclides listed as HRRs in this Draft FTF 3116 Basis Document may not be listed in other 3116 Basis Documents if such radionuclides are not present in the waste or do not contribute significantly to dose to the worker, the public or the inadvertent intruder.

The list of HRRs, Table 5.1-1, was developed beginning with an initial listing of 849 radionuclides compiled from a variety of published resources (such as National Council on Radiation Protection and Measurements information), and included radionuclides not necessarily present in the projected FTF inventory at closure. [CBU-PIT-2005-00228] The initial listing of 849 radionuclides included those radionuclides from Tables 1 and 2 of 10 CFR 61.55[23]. DOE reviewed this initial list and identified those radionuclides that were present in the waste and may be important in meeting performance objectives in 10 CFR Part 61, Subpart C because they contribute to the dose to the workers, the public and/or the inadvertent intruder (for one or more reasonable intruder scenarios) in the FTF PA base (expected) case and sensitivity and uncertainty analyses. In DOE's view, this approach results in a risk-informed list of HRRs 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.

Table 5.1-1: FTF Highly Radioactive Radionuclides

Radionuclide	Radionuclide Half-Life (yr)	Potential Long-Term Radiological Hazards	Potential Short-Term Radiological Hazards
Sr-90 [b, c, d, f]	2.89E+01		X
Tc-99 [c, e]	2.11E+05	X	
I-129 [a]	1.57E+07	X	
Cs-137 [b, c, d, f]	3.00E+01		X
U-234 [a, c]	2.46E+05	X	
Np-237 [a, b, c, e]	2.14E+06	X	
Pu-238 [a, e]	8.77E+01	X	
Pu-239 [a, c, e]	2.41E+04	X	
Pu-240 [a, c, e]	6.56E+03	X	
Am-241 [a, b, c, e]	4.32E+02	X	

[a] HRRs based on groundwater analyses results from the FTF PA. [SRS-REG-2007-00002]
[b] HRRs based on intruder analysis results from the FTF PA. [SRS-REG-2007-00002]
[c] HRRs based on uncertainty and sensitivity run results from the FTF PA. [SRS-REG-2007-00002]
[d] HRRs based on potential contribution to worker dose.
[e] Included in Table 1 of 10 CFR 61.55.
[f] Included in Table 2 of 10 CFR 61.55.

The list of FTF HRRs in Table 5.1-1 account for approximately 99% of the current radioactivity in the FTF waste. The short-lived fission products Cs-137 and Sr-90 and their equilibrium daughter products, Ba-137m and Y-90, are by far the predominant sources of radioactivity present in the FTF waste today. Cs-137, and its daughter Ba-137m, are typically considered as a single radionuclide for human health protection purposes because the half-life of Ba-137m is so short that it only exists when Cs-137 is present. The same is true for Sr-90 and its daughter Y-90. Accordingly, the discussions that follow in this Draft FTF 3116 Basis Document focus on Cs-137 or Sr-90 since approaches that are effective in removing Cs-137 and Sr-90 also remove Ba-137m and Y-90, respectively. Approximately 98% of the current radioactivity in the FTF waste is associated with these two radionuclides and their short-lived daughters. [SRR-LWP-2010-00058] Moreover, Cs-137, Sr-90 and their daughters are present in sufficient concentrations in the FTF waste that, without shielding and controls, they produce radiation emissions that would present risk to humans simply due to their proximity without direct inhalation or ingestion. Accordingly, they are of potential acute hazard to occupational workers, the public and the environment.

The remainder of the radionuclides listed are the long-lived isotopes that may pose the greatest risk in the future to human health because of their long life and because they present human health risk if directly inhaled or ingested. These radionuclides combined account for less than 1% of the current radioactivity in the FTF waste today[24]. [SRR-LWP-2010-00058]

5.1.2 Performance Assessment Radionuclides

As explained above, DOE has included in the list of HRRs 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 FTF PA, which includes sensitivity and

[23] Although Tables 1 and 2 in 10 CFR 61.55 specify concentration limits for certain radionuclides in the form of activated metal, DOE includes such radionuclides without regard to whether such radionuclides are in the form of activated metal.
[24] The remaining radionuclides in the FTF waste in combination contribute approximately 1% of the current radioactivity.

uncertainty analyses. The FTF PA applied a rigorous, documented, multi-step, multi-factor screening methodology (including consideration of National Council on Radiation Protection and Measurements [NCRP] information) to 849 radionuclides compiled from radionuclides present in the FTF (based on the SRS Waste Characterization System, process knowledge and available sampling data) as well as radionuclides reported in a variety of published resources. Specifically, DOE used the following approach in the FTF PA to focus on those radionuclides that contribute to the dose for various pathways. [SRS-REG-2007-00002]

For the purpose of determining which radionuclides should be evaluated in the FTF PA, an initial radionuclide screening process was developed and performed, evaluating an initial list of 849 radionuclides compiled from a variety of published resources including the following:

- *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water and Ground*, Volumes I and II, National Council on Radiation Protection and Measurements, which identifies 826 nuclides of interest in determining radiation exposure due to releases to the air, water and ground. [NCRP-123]
- The EPA Risk Assessment Web Site, which provides conversion factors for 824 nuclides which are of interest in determining human health cancer risk from radionuclides in the air, soil and water. [http://www.epa.gov/radiation/heast/]
- *Fission-Product Yields from Neutron-Induced Fission*, Nucleonic - Reference Data Manual, which details 148 radionuclides produced from thermal neutron induced fission of U-235. [Nucleonics_1960]
- *Integrated Data Base Report-1996: U.S. Spent Nuclear Fuel and Radioactive Waste Inventories, Projections and Characteristics*, DOE, which identified 168 nuclides of issue in making material disposition decisions. [DOE/RW-0006]
- *Derivation of Initial Radionuclide Inventories for the Safety Assessment of the Disposal of Used CANDU Fuel*, Atomic Energy of Canada Limited, which lists 211 nuclides of interest in making decisions about disposal of spent fuel from heavy water reactors (note that the five reactors operated at SRS were also heavy water reactors). [AECL-9881]

This initial screening process considered the following information:

- physical properties of each radionuclide such as half-life and decay mechanism,
- waste source and handling based on radionuclide production mechanisms and time since the radionuclide was produced, and
- screening factors for radionuclide ground disposal developed in *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water and Ground* [NCRP-123] which convert a quantity of each radionuclide to a dose.

This initial screening reduced the radionuclide list from 849 down to 159 radionuclides. [CBU-PIT-2005-00228]

Additional screening of the 159 radionuclides was performed to identify the radionuclides to be considered in the initial FTF inventory for the PA. The screening and adjustment included the following:

- Radionuclides were screened out if there were no ancestors present from the specific decay chain or no decay source for the radionuclide.
- The FTF waste production history information was evaluated for the potential for a specific radionuclide to be present in FTF. This step screened out radionuclides not present within the FTF waste.
- In general, radionuclides present due to ingrowth from a decay series were screened out, however, production history was used to retain those radionuclides present at a greater proportion than from the decay series (i.e., Np-237). This screening step eliminated radionuclides that are present only due to the decay of their parent radionuclide (i.e., Ba-137m, Y-90, Ra-226 and Th-229). The inventory of these radionuclides can be controlled by removing the parent radionuclide(s) to the “maximum extent practical.”
- Several radionuclides were added back to the list based on the possibility that these radionuclides may have theoretically been created at various points of production in SRS history.

- Radionuclides with less than a five-year half-life were screened out. This screening reflects that active institutional control will be maintained over the site for 100 years after facility closure. The inventories of these radionuclides will be significantly diminished due to the amount of radioactive decay that will occur during the assumed 100-year institutional control period.

Based on the above screening and adjustment, an additional 105 radionuclides were screened out, thus reducing the FTF PA modeling initial inventory radionuclide number to 54. For the waste tanks and ancillary structures, an initial inventory value for these 54 radionuclides was developed and was used as input into the FTF PA modeling. The results of the FTF PA analyses were then evaluated using the methodology described below to determine which of the 54 radionuclides would be considered HRRs for FTF. [SRR-CWDA-2009-00045, SRS-REG-2007-00002]

5.1.3 Highly Radioactive Radionuclides Based on 100-Meter Groundwater Analysis (For Member of the Public Following Closure)

To determine which radionuclides are HRRs, DOE considered the doses estimated in the FTF PA, as well as the NRC guidance in NUREG 1854, the NRC guidance in Volume 2 of NUREG 1757 (referenced in NUREG 1854), and recommendations by the NRC during consultation to date[25]. While the approach followed by DOE in this Draft FTF 3116 Basis Document differs in some aspects from the approaches followed in prior 3116 Basis Documents, it is a thorough and methodical approach that reflects NRC guidance and recommendations and results in the identification of those radionuclides that may provide more than an insignificant contribution to dose.

Several radionuclides have been included on the HRR list based on an evaluation of the FTF 100-meter groundwater dose, using the groundwater dose results calculated in the FTF PA. For the FTF PA, the 100-meter point is the point of maximum exposure at, or outside of, the FTF 100-meter buffer zone. The groundwater analysis in the FTF PA utilized the initial inventory of 54 radionuclides resulting from the screening analysis described in the previous section as input for the FTF PA model. The model used to perform the groundwater analysis accounted for radioactive decay and ingrowth throughout the assessment period. [SRS-REG-2007-00002] The results of the groundwater analysis were then evaluated to determine the HRRs.

DOE examined the resulting dose contributions to the groundwater analysis, shown in the FTF PA, from all individual radionuclides in the FTF PA initial inventory at the time of closure. Those radionuclides which, in aggregate, would not contribute greater than 1.25 mrem/yr were not considered HRRs. The first step in this groundwater evaluation was to review the resulting doses from the FTF PA groundwater analysis at any time within 20,000 years[26]. The individual doses from each radionuclide were then listed in order of contribution. DOE conducted a quantitative analysis to determine those radionuclides with an aggregate contribution to dose of less than or equal to 1.25 mrem/year[27]. These radionuclides were screened from the HRR list. Based on this evaluation, the remaining radionuclides - I-129, Cs-135, Ra-226, Th-229, U-233, U-234, Np-237, Pu-239 and Pu-240 - were initially identified for consideration. For those radionuclides, the projected inventories (as shown in the FTF PA) at the time of FTF closure were reviewed. For those radionuclides with a relatively insignificant initial inventory (i.e., Ra-226 and Th-229) the associated decay chains were examined. Reduction of these radionuclides is accomplished through

[25] NRC suggested during a public Draft FTF NDAA 3116 Basis Document scoping meeting held on July 13-14, 2010 that DOE consider the guidance in NUREG-1757 in establishing these evaluation thresholds. [SRR-CWDA-2010-00091]

[26] The FTF PA evaluates the consequences of closure activities during the performance period (i.e., 10,000 years) after closure. The DOE also has evaluated for periods beyond 10,000 years in the FTF PA to further inform closure decisions. To ensure that a conservative approach is taken in selection of the HRRs, the calculation of groundwater dose for 20,000 years was used to account for tank degradation for Type I and Type III/IIIA waste tanks beyond 10,000 years. DOE is using a 20,000-year evaluation period for determination of HRRs due to the expected tank liner failures beyond the 10,000-year performance period and resultant peak doses. This tank liner failure analysis is specific to the SRS FTF Type I and Type III/IIIA waste tanks.

[27] This approach is consistent with the guidance and general approach in Volume 2 of NUREG-1757, Consolidated NMSS Decommissioning Guidance [NUREG-1757], 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-referenced 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 the NUREG is applicable in the 3116 context. To ensure that selection of the HRRs is sufficiently conservative, DOE has used 5 percent (i.e., 1.25 mrem/yr) of the 25 mrem/yr all-pathways dose limit.

the removal of the respective parent radionuclides, i.e., U-233 (for Th-229), U-234 (for Ra-226), Pu-238 (for Ra-226). An additional parent radionuclide, Am-241, was included based on progeny ingrowth of Np-237.

Since the completion of the FTF PA, DOE has performed waste characterization of the residuals in both Tank 18 and Tank 19 which showed that Cs-135 and U-233 were significantly lower than projected in the FTF PA. [SRR-CWDA-2010-00117, SRR-CWDA-2010-00118] Moreover, based on the actual Tanks 18 and 19 source terms and the results of the FTF PA analysis, the associated doses from the Cs-135, U-233 and its daughter Th-229 inventories in aggregate with the previously screened radionuclides are not expected to contribute greater than 1.25 mrem/yr to the public groundwater dose.

DOE believes that the screening of radionuclides whose dose contribution in aggregate is less than or equal to 1.25 mrem/yr is sufficiently low, compared to the 25 mrem/yr all-pathways dose limit, to capture all risk-significant radionuclides in those that remain. For the purpose of this Draft FTF 3116 Basis Document, I-129, U-234, Np-237, Pu-238, Pu-239, Pu-240 and Am-241 were included in the HRR list based on the groundwater pathway.

5.1.4 Highly Radioactive Radionuclides Based on Air Pathway Analysis (For Member of the Public Following Closure)

In a similar manner, radionuclides were evaluated for inclusion on the HRR list based on the FTF 100-meter dose from airborne radionuclides, using the air pathway dose results calculated in the FTF PA. In the FTF PA, radionuclides contained in the initial inventory that are susceptible to volatilization were considered in the air pathways analysis. These radionuclides included H-3, C-14, Cl-36, Se-79, Tc-99, Sb-125, Sn-126 and I-129. For the FTF PA, the 100-meter point is the point of maximum exposure at or outside of the FTF 100-meter buffer zone. The FTF PA shows that the air pathway is not a significant contributor to dose, and contributes, in aggregate approximately 0.2 mrem/yr peak dose. [SRS-REG-2007-00002] Therefore, for the purpose of this Draft FTF 3116 Basis Document, no radionuclides were included in the HRR list based on the air pathway.

5.1.5 Highly Radioactive Radionuclides Based on Intruder Pathway Analysis

Several radionuclides have been included on the HRR list based on an evaluation of the FTF inadvertent intruder dose, using the intruder dose results calculated in the FTF PA. The intruder analysis in the FTF PA utilized the initial inventory of 54 radionuclides resulting from the screening analysis described previously as input for the FTF PA model. The model used to perform the intruder analysis did account for radioactive decay and ingrowth throughout the assessment period. [SRS-REG-2007-00002] The results of the intruder analysis were then evaluated to determine the HRRs.

DOE examined the resulting dose contributions to the inadvertent intruder analysis, shown in the FTF PA, from all individual radionuclides in the FTF PA initial inventory at the time of closure. Those radionuclides which, in aggregate, would not contribute greater than 25 mrem/yr were not considered HRRs. The first step in this intruder evaluation was to review the resulting doses from the FTF PA intruder analysis at any time within 20,000 years[28]. The individual doses from each radionuclide were then listed in order of contribution. DOE conducted a quantitative analysis to determined those radionuclides with an aggregate contribution to dose of less than or equal to 25 mrem/yr[29]. These radionuclides were screened from the HRR list. Based on this evaluation the remaining radionuclides - Sr-90, Y-90, Cs-137, Ba-137m, Th-229, U-233 and Np-237 - were initially identified for consideration. For those radionuclides, the projected

[28] The FTF PA evaluates the consequences of closure activities during the performance period (i.e., 10,000 years) after closure. The DOE has evaluated for periods beyond 10,000 years in the FTF PA to further inform closure decisions. To ensure that a conservative approach is taken in selection of the HRRs, the calculation of inadvertent intruder dose for 20,000 years was used to account for tank degradation for Type I and Type III/IIIA waste tanks beyond 10,000 years. DOE is using a 20,000-year evaluation period for determination of HRRs due to the expected tank liner failures beyond the 10,000-year performance period and resultant peak doses. This tank liner failure analysis is specific to the SRS FTF Type I and Type III/IIIA waste tanks.

[29] This approach is consistent with the guidance and general approach in Volume 2 of NUREG-1757, Consolidated NMSS Decommissioning Guidance [NUREG-1757], 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-referenced 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 the NUREG is applicable in the 3116 context. To ensure that selection of the HRRs is sufficiently conservative, DOE has used 5 percent, 25 mrem/yr, of the 500 mrem/yr all-pathways dose recommended in NRC guidance (NUREG-0945 and NUREG-1854).

inventories (as shown in the FTF PA) at the time of FTF closure were reviewed. For the radionuclide with a relatively insignificant initial inventory (i.e., Th-229), the associated decay chain was examined. Reduction of this radionuclide is accomplished through the removal of the respective parent radionuclide (i.e., U-233). An additional parent radionuclide, Am-241, was included based on progeny ingrowth of Np-237.

Since the completion of the FTF PA, DOE has performed waste characterization of the residuals in both Tank 18 and Tank 19 which showed that U-233 was significantly lower than projected in the FTF PA. [SRR-CWDA-2010-00117, SRR-CWDA-2010-00118] Moreover, based on the actual Tanks 18 and 19 source terms and the results of the FTF PA analysis, the associated doses from U-233 and its daughter, Th-229, inventories, in aggregate with the previously screened radionuclides, are not expected to contribute greater than 25 mrem/yr to the public groundwater dose.

Cesium-137, and its daughter Ba-137m, are typically considered as a single radionuclide for human health protection purposes because the half-life of Ba-137m is so short that it only exists when Cs-137 is present. The same is true for Sr-90 and its daughter Y-90. Removal of Ba-137m and Y-90 is accomplished through removal of the parent radionuclides Cs-137 and Sr-90, respectively, during the waste removal process. Therefore, Ba-137m and Y-90 are not considered HRRs based on contribution to the inadvertent intruder dose.

DOE believes the screening of radionuclides whose dose contribution, in aggregate, is less than or equal to 25.0 mrem/yr is sufficiently low, compared to the 500 mrem/yr peak intruder dose recommended in NRC guidance (NUREG-0945 and NUREG-1854), to capture all risk significant radionuclides in those that remain. For the purpose of this Draft FTF 3116 Basis Document, Sr-90, Cs-137, Np-237 and Am-241 were included in the HRR list based on the intruder pathway.

5.1.6 Highly Radioactive Radionuclides Based on Uncertainty and Sensitivity Analyses

Some radionuclides have been included on the HRR list based on an evaluation of the uncertainty and sensitivity analyses included in the FTF PA. [SRS-REG-2007-00002] The FTF PA uncertainty and sensitivity analyses were reviewed to identify those radionuclides shown to have the most influence on the model results.

The purpose of the uncertainty and sensitivity analyses was to consider the effects of uncertainties in the conceptual models used and sensitivities in the parameters used in the mathematical models. While the uncertainty and sensitivity analyses were primarily performed using a probabilistic model, some additional single parameter sensitivity analyses were performed through deterministic modeling. The probabilistic model allows for variability of multiple parameters simultaneously, so concurrent effect of changes in the model can be analyzed. The deterministic model single parameter analysis provides a method to evaluate the importance of the uncertainty around a single parameter of concern. Using both probabilistic and deterministic models for sensitivity analysis versus a single approach provides additional information concerning which parameters are of most importance to the FTF PA model. In addition, as part of the sensitivity analyses, modeling was performed for a period of 100,000 years. [SRS-REG-2007-00002]

The FTF PA considers the uncertainties and sensitivities associated with the projected dose results to a member of the public through uncertainty analysis of the FTF probabilistic model, through sensitivity analysis using the FTF probabilistic model and through sensitivity analysis using the FTF deterministic model. A review of the uncertainty analysis realizations with the highest peak doses show Tc-99, Np-237, Pu-239 and Pu-240 to be significant. A review of the sensitivity analysis performed using the FTF probabilistic model shows Tc-99 and Pu-239 to be significant. A review of the sensitivity analysis performed using the FTF deterministic model, including a barrier analysis, show Tc-99, U-234, Np-237, Pu-239 and Am-241 to be significant. Based on an evaluation of the FTF PA results for these different analyses, Tc-99, U-234, Np-237, Pu-239, Pu-240 and Am-241 were included in the HRR list. The FTF PA also considered the effects on the Intruder Analysis of uncertainties in the conceptual models used and sensitivities in the parameters used in the mathematical models. Probabilistic intruder sensitivity analyses performed identified Tc-99 and Pu-239 as potentially significant to the inadvertent intruder results. A review of the FTF probabilistic sensitivity analyses identified Sr-90, Y-90, Cs-137, Ba-137m and Am-241 as potentially significant to the inadvertent intruder results. Cesium-137, and its daughter Ba-137m, are typically considered as a single radionuclide for human health protection purposes because the half-life of Ba-137m is so short that it only exists when Cs-137 is present. The same is true for Sr-90 and its daughter Y-90. Removal of Ba-137m and Y-90 is accomplished through removal of the parent radionuclides Cs-137 and Sr-90, respectively, during the waste removal process. Therefore, Ba-137m and Y-90 are not considered HRRs based on an evaluation of the uncertainty and sensitivity analyses. Strontium-90, Tc-99, Cs-137, Pu-239 and Am-241 were included in the HRR list based on an evaluation of the results for the FTF PA inadvertent intruder uncertainty and sensitivity analyses.

Based on the FTF PA uncertainty and sensitivity analyses, Sr-90, Tc-99, Cs-137, U-234, Np-237, Pu-239, Pu-240 and Am-241 were included in the HRR list.

5.1.7 Highly Radioactive Radionuclides Summary

The results of the HRR evaluation are summarized in Table 5.1-2. The table provides the results of the evaluation for each of the 54 radionuclides contained in the initial FTF PA inventory. Radionuclides were considered HRRs based on the evaluations and considerations discussed above. Based on these evaluations and considerations, Sr-90, Tc-99, I-129, Cs-137, U-234, Np-237, Pu-238, Pu-239, Pu-240 and Am-241 are considered the HRRs in the FTF stabilized residuals, FTF waste tanks and FTF ancillary structures at the closure of FTF.

Table 5.1-2: Radionuclides Contained in FTF PA Initial Inventory

		Evaluation Results[a]							
		Groundwater Analysis		Air Pathway Analysis		Intruder Analysis			
Radionuclide with Initial Inventory	Half-Life (yr)	Dose > 1.25 mrem/yr	Impact on Progeny > 1.25 mrem/yr	Dose > 1.25 mrem/yr	Impact on Progeny > 1.25 mrem/yr	Dose > 25.0 mrem/yr	Impact on Progeny > 25.0 mrem/yr	Uncertainty & Sensitivity Analysis	Contribution to Worker Dose
H-3[c]	1.23E+01	x	x	x	x	x	x	x	x
C-14[b]	5.70E+03	x	x	x	x	x	x	x	x
Al-26	7.17E+05	x	x	x	x	x	x	x	x
Cl-36	3.01E+05	x	x	x	x	x	x	x	x
K-40	1.25E+09	x	x	x	x	x	x	x	x
Ni-59[b]	7.60E+04	x	x	x	x	x	x	x	x
Ni-63[c]	1.00E+02	x	x	x	x	x	x	x	x
Co-60[c]	5.30E+00	x	x	x	x	x	x	x	x
Se-79	2.95E+05	x	x	x	x	x	x	x	x
Sr-90[c]	**2.89E+01**	x	x	x	x	✓	x	✓	✓
Y-90	7.31E-03	x	x	x	x	x	x	x	x
Zr-93	1.53E+06	x	x	x	x	x	x	x	x
Nb-93m	1.61E+01	x	x	x	x	x	x	x	x
Nb-94[b]	2.03E+04	x	x	x	x	x	x	x	x
Tc-99[b]	**2.11E+05**	x	x	x	x	x	x	✓	x
Pd-107	6.50E+06	x	x	x	x	x	x	x	x
Sn-126	2.30E+05	x	x	x	x	x	x	x	x
Sb-126	1.24E+01	x	x	x	x	x	x	x	x
Sb-126m	1.92E+01	x	x	x	x	x	x	x	x

✓ Included on the HRR list based on specific evaluation threshold.
x Does not meet specific evaluation threshold.
a Radionuclides considered HRRs if one or more evaluation thresholds are met.
b Included in Table 1 of 10 CFR 61.55.
c Included in Table 2 of 10 CFR 61.55.
Note: HRRs for this Draft FTF 3116 Basis Document are highlighted.
[SRS-REG-2007-00002]

Table 5.1-2: Radionuclides Contained in FTF PA Initial Inventory (Continued)

Radionuclide with Initial Inventory	Half-Life (yr)	Evaluation Results[a]							
		Groundwater Analysis		Air Pathway Analysis		Intruder Analysis			
		Dose > 1.25 mrem/yr	Impact on Progeny > 1.25 mrem/yr	Dose > 1.25 mrem/yr	Impact on Progeny > 1.25 mrem/yr	Dose > 25.0 mrem/yr	Impact on Progeny > 25.0 mrem/yr	Uncertainty & Sensitivity Analysis	Contribution to Worker Dose
I-129[b]	**1.57E+07**	✓	x	x	x	x	x	x	x
Cs-135	2.36E+06	x	x	x	x	x	x	x	x
Cs-137[c]	**3.00E+01**	x	x	x	x	✓	x	✓	✓
Ba-137m	4.85E-06	x	x	x	x	x	x	x	x
Sm-151	9.00E+01	x	x	x	x	x	x	x	x
Eu-152	1.35E+01	x	x	x	x	x	x	x	x
Eu-154	8.59E+00	x	x	x	x	x	x	x	x
Pt-193	5.00E+01	x	x	x	x	x	x	x	x
Ra-226	1.60E+03	x	x	x	x	x	x	x	x
Ac-227	2.17E+01	x	x	x	x	x	x	x	x
Th-229	7.88E+03	x	x	x	x	x	x	x	x
Th-230	7.54E+04	x	x	x	x	x	x	x	x
Pa-231	3.27E+04	x	x	x	x	x	x	x	x
U-232	6.89E+01	x	x	x	x	x	x	x	x
U-233	1.59E+05	x	x	x	x	x	x	x	x
U-234	**2.46E+05**	✓	✓	x	x	x	x	✓	x
U-235	7.04E+08	x	x	x	x	x	x	x	x
U-236	2.34E+07	x	x	x	x	x	x	x	x
U-238	4.47E+09	x	x	x	x	x	x	x	x
Np-237[b]	**2.14E+06**	✓	x	x	x	✓	x	✓	x
Pu-238[b]	**8.77E+01**	x	✓	x	x	x	x	x	x
Pu-239[b]	**2.41E+04**	✓	x	x	x	x	x	✓	x
Pu-240[b]	**6.56E+03**	✓	x	x	x	x	x	✓	x
Pu-241[b]	1.43E+01	x	x	x	x	x	x	x	x
Pu-242[b]	3.75E+05	x	x	x	x	x	x	x	x
Pu-244[b]	8.00E+07	x	x	x	x	x	x	x	x
Am-241[b]	**4.32E+02**	x	✓	x	x	x	✓	✓	x
Am-242m[b]	1.41E+02	x	x	x	x	x	x	x	x
Am-243[b]	7.37E+03	x	x	x	x	x	x	x	x
Cm-243[b]	2.91E+01	x	x	x	x	x	x	x	x
Cm-244[b]	1.81E+01	x	x	x	x	x	x	x	x
Cm-245[b]	8.50E+03	x	x	x	x	x	x	x	x
Cm-247[b]	1.56E+07	x	x	x	x	x	x	x	x
Cm-248[b]	3.48E+05	x	x	x	x	x	x	x	x
Cf-249[b]	3.51E+02	x	x	x	x	x	x	x	x

✓ Included on the HRR list based on specific evaluation threshold.
x Does not meet specific evaluation threshold.
a Radionuclides considered HRRs if one or more evaluation thresholds are met.
b Included in Table 1 of 10 CFR 61.55.
c Included in Table 2 of 10 CFR 61.55.
Note: HRRs for this Draft FTF 3116 Basis Document are highlighted.
[SRS-REG-2007-00002]

5.2 Removal of Highly Radioactive Radionuclides to the Maximum Extent Practical

The NDAA Section 3116(a) provides that certain waste resulting from reprocessing is not high-level waste if the Secretary of Energy, in consultation with the NRC, determines, among other things, that the waste has had HRRs removed "to the maximum extent practical[30]." Section 5.2 and Section 5.3 demonstrates that the FTF residual waste, tanks and ancillary structures will have had HRRs removed to the MEP upon cessation of waste removal activities. 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" (*A Dictionary of Modern English Usage*); "adapted or designed for actual use"; "useful" (http://infoplease.com/ipd/A0598638.html); selected "mindful of the results, usefulness, advantages or disadvantages, etc., of [the] action or procedure" (http://dictionary.cambridge.org/define.asp?key=practical*2+0&dict=A); fitted to "the needs of a particular situation in a helpful way"; "effective or suitable." Therefore, the determination as to whether a particular HRR will be removed to the MEP will vary from situation to situation, based not only on the available technologies but also on the overall costs and benefits[31] of deploying a technology with respect to the conditions in a specific FTF waste tank or ancillary structure. The MEP 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 HRR 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 using such technologies. What may be removal to MEP 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 prior or later point in time.

Moreover, it may not be practical to undertake further removal of certain radionuclides because further removal is not sensible or useful in light of the overall benefit to human health and 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[32] to potential doses to workers, the public, and the hypothetical human intruder.

The HRRs have been and, for tanks and equipment to be cleaned in the future, will be removed from FTF waste tanks and ancillary structures to the MEP for the purpose of removal from service[33] and eventual closure of the waste tanks and ancillary structures. Removal of HRRs to the MEP in FTF waste tanks and ancillary structures occurs through a systematic progression of waste removal and cleaning activities using proven technologies to a point where further removal of HRRs is not sensible or useful in light of the overall benefit to human health, safety and the environment.

5.2.1 Current Status of F-Tank Farm Waste Removal Activities

The FTF includes 22 waste storage tanks. Two of these waste tanks, Tanks 17 and 20, previously underwent final waste removal activities, were removed from service in 1997, were filled with a

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[30] The NDAA Section 3116 does not specify "remedial goals" or other numerical objectives, and does not require DOE to develop any such removal goals or objectives.

[31] While prior NRC and DOE requirements for waste determinations called for removal "to the maximum extent *technically* and *economically* practical" [NRC_03-02-93; DOE M 435.1-1], NDAA 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.

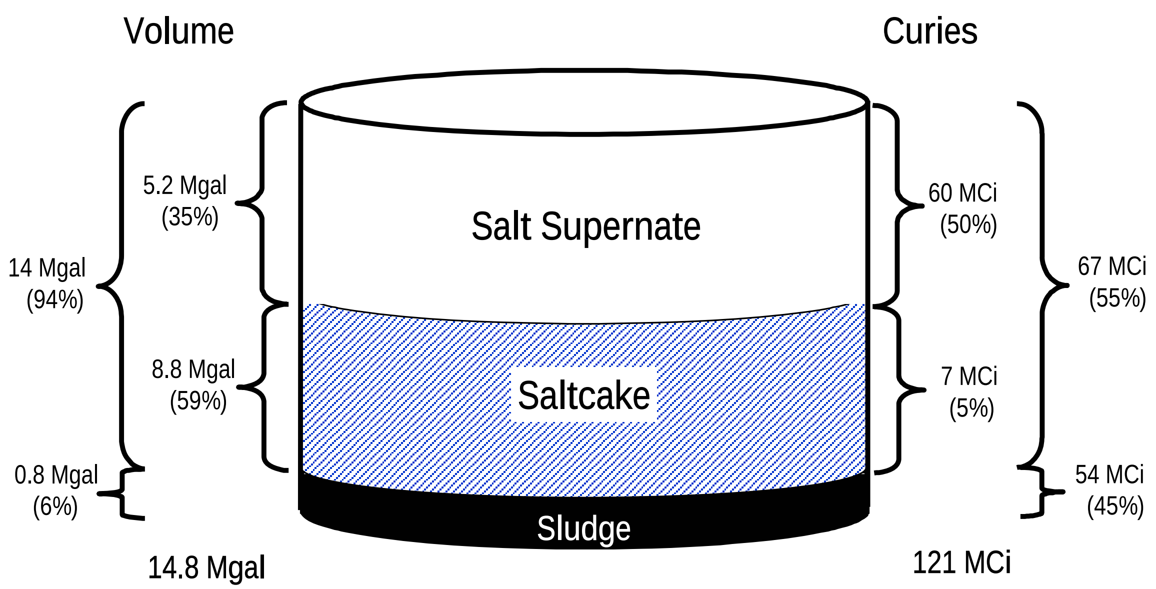
[32] The DOE normally would view radionuclides as making a clearly insignificant contribution if the contribution to dose from those radionuclides, in both the expected case and considering sensitivity analyses, does not exceed any of the following: (1) 10% of the 25-mrem/yr all-pathways annual dose 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. 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* (NUREG-1757), 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.

[33] The term "removal from service" refers to the protocols set forth in the State-approved General Closure Plan (GCP), as described in Section 8.0 of this Draft FTF 3116 Basis Document, to stabilize the waste tank or ancillary structure and prepare it for final closure.

combination of grouting materials, and are not encompassed within this Draft FTF 3116 Basis Document. [PIT-MISC-0002, PIT-MISC-0004] All of the remaining Type I or Type IV waste tanks in FTF are no longer actively receiving fresh canyon waste and many are currently undergoing waste removal activities. Tank 18 and Tank 19 have undergone extensive waste removal campaigns and tank residual samples have been collected for the purposes of characterizing the remaining tank residuals. It is anticipated that Tank 18 and Tank 19 will be the next two waste tanks removed from service in FTF. Tank 5 and Tank 6 have completed bulk waste removal and have undergone extensive heel removal activities. Tank 5 and Tank 6 are anticipated to be the next set of tanks removed from service following Tank 18 and Tank 19. Tank 5 and Tank 6 will represent the first waste tanks removed from service that are Type I tanks and have extensive interior structures (i.e., support columns and cooling coils). Tank 3, Tank 4, Tank 7 and Tank 8 have all undergone initial waste removal activities. Waste removal activities have not been initiated in Tank 1 or Tank 2, or any of the ten Type III/IIIA tanks located in FTF.

As of March 31, 2010, FTF stored approximately 121,000,000 curies in approximately 15,000,000 gallons of waste. The sludge portion of this waste represents approximately six percent of the volume but contains approximately 45 percent of the radioactivity. Of the approximate 14,000,000 gallons of salt waste, approximately 8,800,000 gallons is in the form of saltcake with the remaining approximately 5,200,000 gallons being concentrated supernate. The concentrated supernate accounts for approximately 50 percent of the total radioactivity in the FTF. Figure 5.2-1 graphically presents the breakdown of the waste in FTF in terms of both volume and curies for each of the three primary waste types. [SRR-LWP-2010-00040]

Figure 5.2-1: FTF Waste Tank Composite Inventory



FTF Inventory values as of March 31, 2010

5.2.2 Waste Removal Technologies

The DOE 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, and a new generation of waste removal equipment has been selected for use at SRS. DOE has over 40 years of experience in successfully removing waste from the SRS waste tanks. This experience encompasses removal of all waste types (supernate, sludge and saltcake).

In 2003, DOE used a systematic process to identify, evaluate and select equipment for waste removal tasks to accelerate waste removal and the removal of waste tanks from service. This process is formally documented in a Systems Engineering Evaluation. The evaluation investigated options for bulk waste mixing, waste transfer and residual waste heel removal. The evaluation graded the options on weighted selection criteria such as technical maturity, effectiveness, reliability, reusability, radiological control requirements, integration with the tank farm system and cost. Knowledgeable tank farm operations, engineering, plant support and maintenance personnel identified potential technology candidates based on experience, literature, worldwide web research and contacts with other knowledgeable personnel in the DOE Complex and commercial industry. The team recommended using a combination of mechanical removal technologies and chemical removal technologies, if necessary, to perform waste removal in the tanks. [G-ESR-G-00051]

In addition to the mechanical and chemical waste removal technologies, as the result of a March 2006 DOE-sponsored Tank Cleaning Technical Exchange, DOE identified a new vacuum removal technology for heel removal applications. [CBU-PIT-2006-00067]

The DOE is utilizing mechanical, chemical and vacuum heel removal technologies at SRS in various combinations and sequences depending on the unique characteristics of the waste and conditions in each tank. The DOE will continue to consider new technological developments relevant to waste tank cleaning. A range of potential technologies for evaluation will potentially include technologies developed and/or used at other DOE sites, in domestic commercial industry and in international applications.

The following discusses the current technologies and how they are being utilized at SRS.

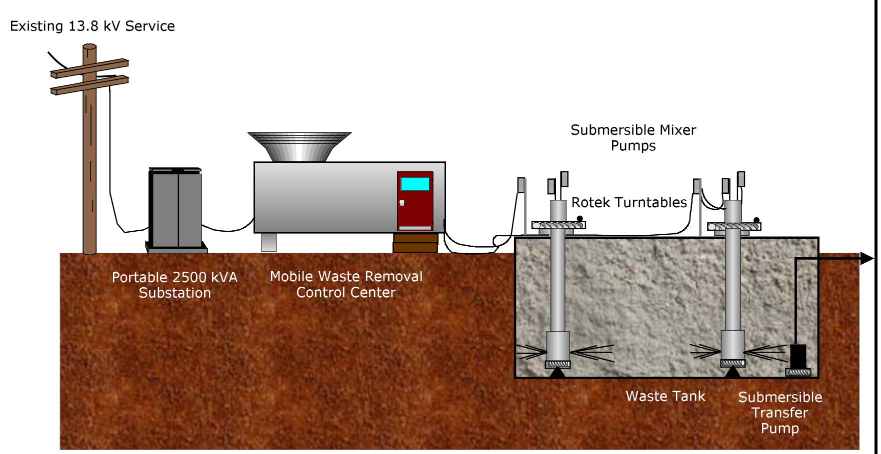
5.2.2.1 Mechanical Heel Removal Cleaning

As documented in the Systems Engineering Evaluation, the team of knowledgeable and experienced engineers and operations personnel recommended using floor-mounted, canned SMPs for bulk waste mixing and a mast-mounted STP for waste transfer.

Based on the recommendations of the team, DOE selected and implemented mechanical removal techniques that use liquid (chemically-treated water[34] and/or supernate) as the media for mixing. Mixer pumps used for bulk waste removal are also used in the subsequent heel removal process and may be augmented by spraying and lancing. Spraying and lancing within the waste tanks is performed by inserting a nozzle through an open riser in the waste tank and directing the liquid at a targeted location. Lancing typically is used to refer to a higher pressure, more concentrated spray pattern aimed at breaking-up or moving the solids within the waste tank. A recycle system, also referred to as a "feed and bleed" system, may be employed to enhance the efficiency and effectiveness of the mechanical removal of the solids.

The technology consists of a Mobile Substation that provides power, a Mobile Waste Removal Control Center that provides local control and monitoring capabilities, SMPs for mixing and suspending waste solids and an STP for waste transfer. These mobile units have the capability of being co-located near any tank or tanks scheduled for waste removal. This concept efficiently performs waste removal using mobile and reusable equipment (Figure 5.2-2)[35].

Figure 5.2-2: Submersible Mixer Pump Waste Removal Diagram



[34] Chemically-treated water is typically utilized when significant volumes will be added to the waste tanks to minimize the potential corrosion to the carbon steel primary tank walls and floors and the secondary annular pans, as applicable.
[35] Figure 5.2-2 depicts two SMPs located in the waste tank, however, during actual cleaning operations DOE may deploy from one to four SMPs within a waste tank based upon the particular tank configuration and waste characteristics.

A key component of the mechanical waste removal equipment is the SMP because the ability to mix and suspend waste solids has a direct impact on the volume of solids remaining after mechanical heel removal. The SMPs are variable speed, single-stage centrifugal pumps with a 305-HP motor that can operate up to 1,600 rpm. The SMPs utilize the tank liquid waste to cool the motor and lubricate the upper and lower bearings. The SMPs are rotated by a turntable assembly that provides the motive force for oscillation or allows for stationary indexing operation. The SMPs have a rotating foot attached to the lower end of the pump, which allows the SMP to rest on the tank floor and oscillate. SMPs used for bulk waste removal are also used in the subsequent residual heel removal process and can be augmented by spraying or lancing. In some cases, a recycle system or feed and bleed system may be employed to enhance mechanical heel removal. Two discharge nozzles give the SMPs the capability to produce an effective cleaning radius of up to 50 feet. [M-CLC-G-00344] While obstructions such as cooling coils and support columns affect the effective cleaning radius of SMPs, the use of multiple SMPs in an obstructed tank can make a significant contribution to removal of waste. Experience in Tank 5 and Tank 6 has demonstrated that two or three SMPs can successfully suspend the majority of the sludge solids in a waste tank with internal obstructions. A Type I tank, such as Tank 5 and Tank 6, represents the most challenging tank for waste removal activities due, in part, to a limited number of access points compared to a Type III/IIIA tank, the presence of roof support columns in the Type I tanks, and horizontal coiling coil runs at the bottom of the waste tank including stacked horizontal runs (often referred to as "fences") that were "field to fit" during the time of waste tank construction versus only having vertical cooling coils in the waste zone in Type III and IIIA tanks. Pump configurations are shown in Figure 5.2-3 and Figure 5.2-4.

Figure 5.2-3: Submersible Mixing Pump and Submersible Transfer Pump

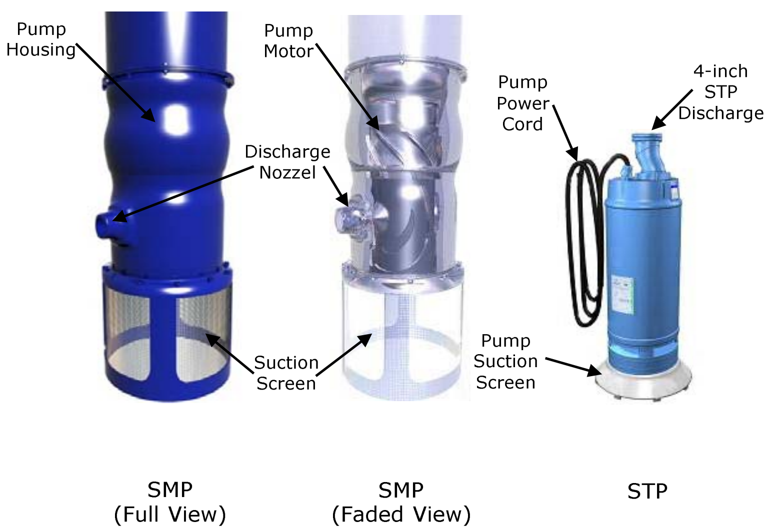
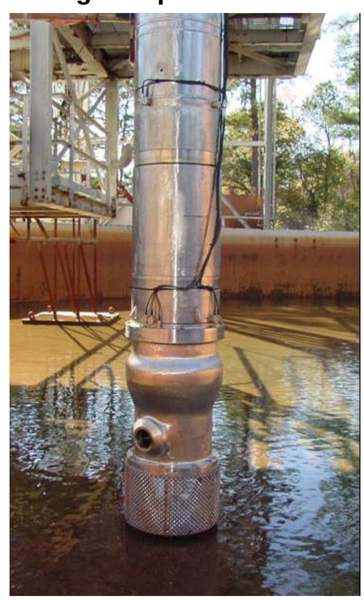


Figure 5.2-4: Submersible Mixing Pump in a Test Tank



The first deployments and operations of SMPs in FTF led to successful removal of sludge from two Type I tanks. Seven mechanical heel removal phases using SMPs in Tank 5 reduced the volume of sludge solids from approximately 34,000 gallons to approximately 3,500 gallons. In Tank 6, SMP operations during eleven mechanical heel removal phases resulted in the reduction of sludge solids from approximately 25,000 gallons to approximately 6,000 gallons. These first uses of SMPs provided the opportunity to gather and evaluate data to refine and enhance operational parameters such as mixer speed, mixer orientation and strategy (oscillation and fixed position) and coordination of mixer and transfer pump operations to optimize waste removal effectiveness in future tanks. Mechanical heel removal using SMPs successfully reduced the volume of sludge to the level required for chemical heel removal in Tank 5 and Tank 6. [M-ESR-F-00107, M-ESR-F-00147, M-ESR-F-00132]

5.2.2.2 Chemical Heel Removal Cleaning

The DOE also successfully uses chemical heel removal technology that employs oxalic acid for chemical treatment of the heel to dissolve solids that cannot be removed by mechanical methods and water addition alone. The oxalic acid may also be sprayed into the tank to further clean contaminants from the internal tank surfaces (e.g., walls, cooling coils, support columns, equipment). At the conclusion of chemical heel removal, the interior of the waste tank are washed with water to rinse oxalic acid from internal surfaces and dislodge loose contamination.

Chemically-aided cleaning techniques have been evaluated for additional levels of waste removal following mechanical heel removal. A team of knowledgeable and experienced engineers and scientists assessed the current knowledge base and collected and evaluated information available on chemical-based methods for removing residual solids from the waste tanks. [WSRC-TR-2003-00401] As part of this study, the team developed recommendations for chemical treatments to remove residual solids. The cleaning agents identified included:

- oxalic acid,
- a mixture of oxalic acid and citric acid,
- a combination of oxalic acid with hydrogen peroxide,
- nitric acid,
- formic acid, and
- organics.

The results of the evaluation support oxalic acid as the cleaning agent of choice. Nitric acid, formic acid and oxalic acid with hydrogen peroxide were all closely grouped for the next best choice. The mixture of oxalic acid and citric acid rated poorly (primarily due to the fact that it performed less well than oxalic acid and the presence of citrate could adversely impact downstream treatment operations, such as the DWPF). Organics rated even more poorly due to large uncertainties in performance and downstream impacts.

The use of oxalic acid was recommended for a number of reasons. First, oxalic acid has been widely studied and used several times to clean waste tanks at SRS and at other sites within the DOE Complex. Its effect on downstream waste treatment process (e.g., DWPF) and evaporator operations is better known. Oxalic acid has been shown to be effective for a wide variety of sludge types and out-performed nitric acid and other chemical cleaning agents in head-to-head laboratory tests. Lastly, oxalic acid is less corrosive to the carbon steel tanks than nitric acid or a combination of oxalic acid and hydrogen peroxide. [WSRC-TR-2003-00401]

Oxalic acid cleaning of tanks was successfully demonstrated through the cleaning of Tank 16[36], located in HTF, in the early 1980s. Tanks 5 and 6 were cleaned using three large batches of 8 weight percent oxalic acid similar to Tank 16 cleaning. This process is referred to as Bulk Oxalic Acid Cleaning. If needed, oxalic acid may be sprayed into the tank to clean contaminants from internal tank surfaces (e.g., walls, cooling coils, support columns, equipment, etc.). [LWO-SPT-2008-00033]

Oxalic acid cleaning in Tank 16 (part of an overall waste removal program which also employed mechanical cleaning) resulted in the removal of over 99% of the activity. [DPST-81-441, DPSP-80-17-23] The oxalic acid was again used as the chemical heel removal method in Tank 5 and Tank 6 in 2008 and 2009. Oxalic acid was applied multiple times in each tank using various methods (e.g., downcomer, spray nozzle, mixer agitation and non-agitation soak times) which provided data for process evaluation, improved effectiveness and overall process optimization. Oxalic acid cleaning in Tank 5 reduced the volume of residual solids to approximately 3,300 gallons, while the residual solids volume in Tank 6 was reduced to approximately 3,500 gallons. [M-ESR-F-00160, M-ESR-F-00165] A significant portion of the remaining waste is non-radioactive oxalate compounds that formed during the chemical cleaning process. Formation of these non-radioactive oxalate compounds is demonstrated by a greater than 30 percent increase in waste volume between the first and second chemical cleaning cycles and the second and

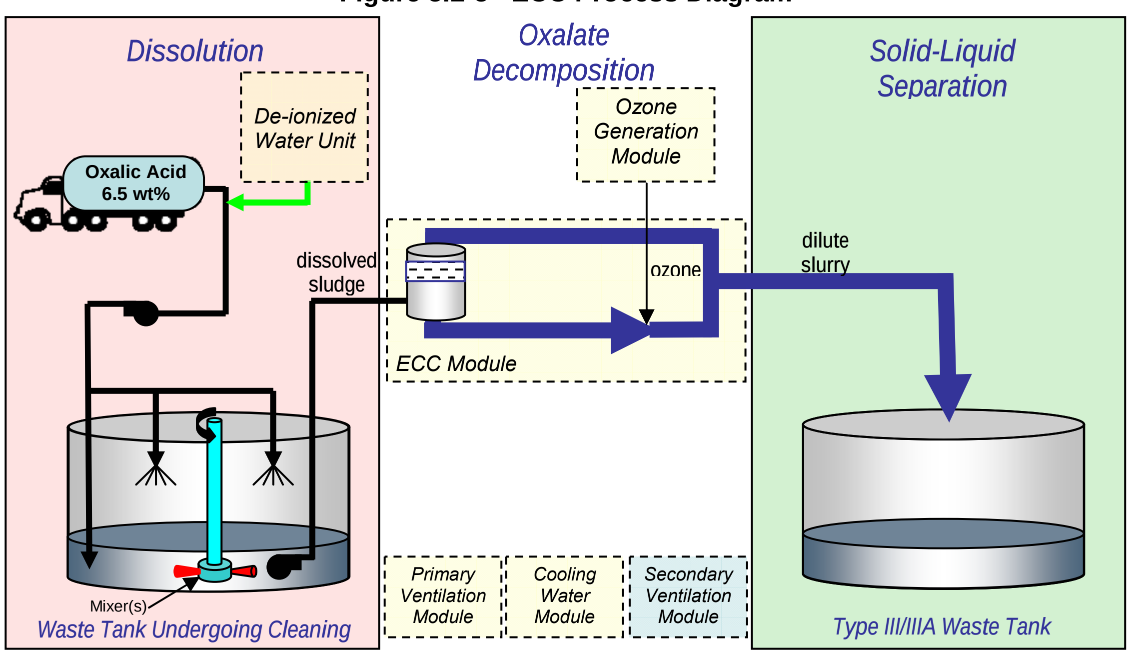
[36] Discussion on Tank 16 cleaning is included to demonstrate successful deployment of the oxalic acid cleaning process; however, Tank 16 is located in the HTF at SRS and is not within the scope of this Draft FTF 3116 Basis Document.

third chemical cleaning cycles for Tank 6 and between the first and second chemical cleaning cycles in Tank 5. [M-ESR-F-00158, M-ESR-F-00165]

As discussed above, chemical cleaning of waste tanks using oxalic acid produces sodium oxalates in the solids slurry that will be eventually included as feed to DWPF. Because of sodium limits and oxalate restrictions on the DWPF feed, preparation of the feed results in a significant amount of additional material being generated that eventually must be processed through SWPF and disposed of in the Saltstone Disposal Facility[37]. The oxalic acid flowsheet evaluations have considered the downstream effects of oxalates on the DWPF process and salt processing to determine quantities of oxalic acid that can be tolerated by the Liquid Waste System. Modeling shows that for every tank that undergoes chemical cleaning, about 51,000 kg of new sodium oxalates (a non-radioactive compound) solids will be created for feed to DWPF. In addition, approximately 500,000 gallons of salt waste will be created. These quantities of oxalates result in additional wash cycles for DWPF feed, increased likelihood of feed breaks to DWPF and extension of the operating life of the entire Liquid Waste System. [SRR-STI-2010-00015] The oxalates are also anticipated to create evaporator foaming and scaling problems. [LWO-SPT-2008-00033] Due to these downstream impacts, the amount of Bulk Oxalic Acid Cleaning in FTF will be carefully controlled to optimize the cleaning effectiveness and the downstream waste treatment processes.

ECC is a new oxalic acid chemical cleaning process being tested and designed by DOE for use in FTF tanks. ECC is based on adapting proven techniques from the commercial reactor and steam generating industries to the cleaning of carbon steel waste tanks. The ECC process uses dilute oxalic acid to dissolve residual waste solids and clean the tank internals. The oxalates in the resulting acid stream are then destroyed using ozone. [SRR-STI-2010-00015] A dilute slurry containing the remaining dissolved metals and associated radionuclides is fed to a settling tank (i.e., FTF waste tank) with the liquid fraction undergoing subsequent evaporation. Since this process can be utilized with minimal impacts on the Liquid Waste System, ECC can continue until the process is no longer effective in removing the residuals. This chemical cleaning process minimizes the impacts of the use of oxalic acid for residual heel cleaning and potentially permits additional cleaning opportunities. [LWO-SPT-2008-00033] Figure 5.2-5 provides a conceptual ECC Process diagram[38].

Figure 5.2-5 ECC Process Diagram



[37] See Appendix A for a brief description of DWPF, SWPF and Saltstone Disposal Facility operations.
[38] The ECC process is still in the stage of being tested and designed by DOE. The figure depicted represents a conceptual diagram of the ECC process and is subject to change as development of the process advances.

5.2.2.3 Vacuum Heel Removal Cleaning

As the result of a March 2006 DOE-sponsored Tank Cleaning Technical Exchange, a new vacuum technology was identified. The DOE has adapted and successfully used this new technology in the unobstructed Type IV tanks, Tanks 18 and 19. This technology used an ultra-high-pressure water eductor to vacuum residual solids and transport the slurry to a receipt tank. This technology was initially deployed in Type IV tanks with no internal obstructions due the size of the device and the large accompanying tether system.

To deploy this technology in Tank 18 and Tank 19, DOE utilized a cleaning device, called a Mantis, which consists of a mechanical crawler and an eductor assembly that made up a retrieval system utilizing an ultra-high-pressure water eductor to vacuum residual solids and transport the slurry to a receipt tank (Figure 5.2-6). The process system consists of a remotely controlled, in-tank Mantis, an umbilical hose containing hydraulic supply lines and the high-pressure water hoses, in-tank waste retrieval hose, a diesel-driven ultra-high-pressure water pump, a motor-driven high pressure water pump, hydraulic pump skid, a diesel generator, above-ground hose-in-hose transfer lines, WMC and support equipment. The device was inserted into the tank through a 24-inch riser in a folded position. Once inside the tank, the device was unfolded into its operational configuration.

Figure 5.2-6: Mantis

The Mantis was remotely driven around the waste tank bottom by an operator located in the Control Center. A high pressure hydro-lance at its front was used to break up waste mounds and an eductor was used to vacuum waste from the floor of the waste tanks. The waste traveled through the eductor in-tank waste retrieval hose up into a tee spool piece located on top of the tank riser and then through an above-ground transfer line that terminated inside a WMC installed inside a riser on the receipt tank. An immersion mill, located near the bottom of the WMC, size-reduced solid waste particles so that the particles can be more easily re-suspended in future waste removal activities. [WSRC-TR-2007-00327]

The deployment of this new cleaning device allowed removal of waste from Tanks 18 and 19 to a greater extent than the technologies available when waste removal was previously discontinued due to diminishing returns in 2003 and 2001, respectively.

The working end effecter of the Mantis, the ultra-high-pressure eductor system, effectively vacuumed residual solids and transported the slurry to the receipt tank. The cooling coils in Type I, Type III and Type IIIA tanks severely restrict the mobility of large tethered mechanical crawlers such as the Mantis platform. However, DOE recognizes the potential for future use of vacuum technology deployed on other platforms specifically tailored for applications in tanks with obstructions.

5.2.3 Optimization of Existing Technologies

DOE continues to optimize its existing technologies that have been successfully deployed for waste removal. DOE is continuing to pursue small-scale robotic technologies for waste removal and sampling applications in tanks with extensive internal obstructions. For example, a small robotic crawler was developed to sample Tanks 18 and 19 (Figure 5.2-7) and is being adapted for applications in future tanks. Such tactical applications of tailored robotic platforms will continue to be used in future waste removal activities in FTF.

Figure 5.2-7: Robotic Sampler at Test Facility

5.3 Removal to the Maximum Extent Practical

As described above, extensive waste removal operations have occurred at SRS and specifically within FTF. Based on waste removal experience to date and anticipated new technologies, FTF waste removal activities will result in significant collective removal of all waste including HRRs[39].

Removal of HRR's begins with the removal of the solids and liquid from a waste tank or ancillary structure in a bulk waste removal phase. Following bulk waste removal, heel removal is performed using a mix of technologies described above, as appropriate, accounting for the physical configuration of the tank and the chemical characteristics of the waste.

Throughout the process, DOE continually evaluates the ongoing effectiveness of the technology being implemented and optimizes the existing technologies. In addition, DOE evaluates the usefulness and practicality of additional technology deployment once the existing technology has reached the point of diminished effectiveness for HRR removal. The DOE's approach consists of the following phases: initial technology selection, technology implementation, technology execution, technology effectiveness evaluation and additional technology evaluation.

The FTF Type IV tanks, Tanks 17, 18, 19 and 20, have all undergone waste removal and tank cleaning activities resulting in a relatively small quantity of resultant tank residuals. For example, in 1998 following bulk waste removal in Tank 19, DOE used a systematic selection process which was documented in a Systems Engineering Evaluation to select the best available technology at the time for heel removal activities in Tank 19. [PIT-MISC-0040] Heel removal activities using the selected mechanical removal technology were carried out in 2001. In 2001, a similar selection process that was used for the selection of the heel removal technology for Tank 19 was also used to select a removal technology for the heel in Tank 18, similarly a Type IV waste tank. [WSRC-RP-2001-00024] This Tank 18 technology selection took into account additional technology studies conducted since the issuance of the Tank 19 Systems

[39] Experience to date with waste removal for Tanks 18 and 19 resulted in removal of approximately 99% of the radioactivity based on a starting point of the maximum historical radionuclide inventory in those tanks. See SRR-CWDA-2009-00030. In this regard, Section 3116 does not specify "remedial goals" or other numerical objectives and does not require DOE to develop any such removal goals or objectives. Although the cleaning methodologies are expected to collectively remove 99% of HRRs, based on a starting point of the maximum historical radionuclide inventory in the overall FTF, individual tanks or ancillary structures may not achieve this level of HRR removal on an individual basis. Demonstration that waste removal within a particular waste tank or ancillary structure has achieved 99% removal of HRRs is not, by itself, a justification for stopping HRR removal activities. In addition, demonstration that residual radionuclide inventory of a given waste tank or ancillary structure is below that assumed in the FTF PA is not sole justification to conclude cleaning activities on an individual tank or ancillary structure.

Engineering Evaluation. Heel removal activities were carried out in Tank 18 in 2002 utilizing a different mechanical removal technology. In these initial Tank 18 and Tank 19 heel removal campaigns a series of waste removal phases were carried out in each of the tanks until it was no longer practical to continue with the mechanical removal technologies that were being utilized. In 2006, following initial heel removal campaigns using the tailored mechanical removal techniques, it was determined that it was practical to deploy an alternative vacuum technology, the Mantis (as described above in Section 5.2.2.3), that could result in significant additional waste removal within these tanks.

Throughout the heel removal activities in Tank 18 and Tank 19 utilizing the Mantis, DOE continually worked to optimize the effectiveness of the Mantis and minimize the impact on the rest of the Liquid Waste System by adjusting how the sprays were utilized, attempting different vacuuming patterns, using the hose/cable bundle to drag the solids into a concentrated area, or turning off the sprays when possible to improve removal efficiency and reduce space impacts on the receipt tank. During the Mantis campaigns on both Tank 18 and Tank 19, the Mantis became ineffective due to failure of one or more of the in-tank components on the equipment. In both cases, DOE evaluated the costs of repairing the Mantis and the anticipated effectiveness once repaired and determined that it was practical to make the repairs and continue the heel removal operations. The Mantis equipment was utilized within Tank 18 and Tank 19 until it was no longer effective. Factors leading to this decision included such things as: visual observation of remaining tank residuals, transfer line radiation readings, significant increase in ratio of water additions to solids removed and significant equipment degradation. Figures 5.3-1 and 5.3-2 depict the overall waste removal effectiveness for Tanks 18 and 19 respectively. Once it was determined that the existing Mantis equipment was no longer effective, alternative HRR removal technologies were reviewed to determine practicality for design, construction, deployment and operation. This alternative technology analysis considered the current tank conditions as well as the impact to the overall Liquid Waste System from the operation of such technologies. [SRR-CWDA-2009-00030] Based on this technology review, which included consideration of costs, worker dose, and other potential impacts, DOE determined that deployment of an alternate technology in Tank 18 and Tank 19 was not practical.

Figure 5.3-1: Tank 18 Waste Removal

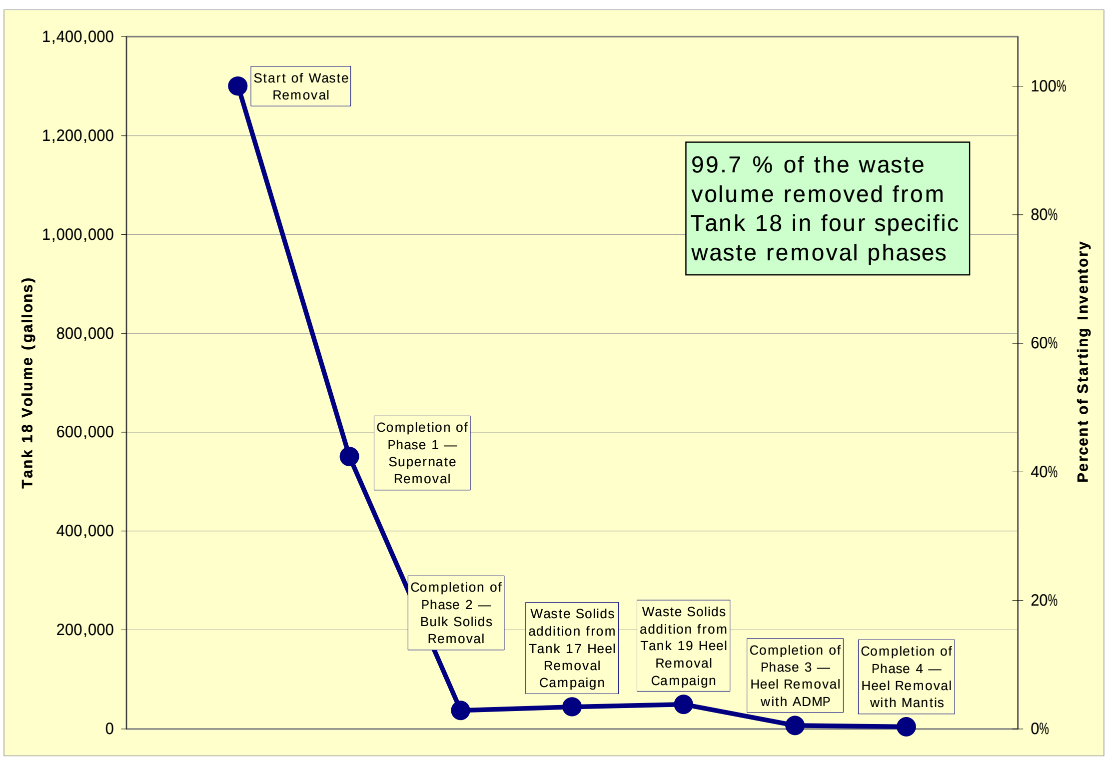
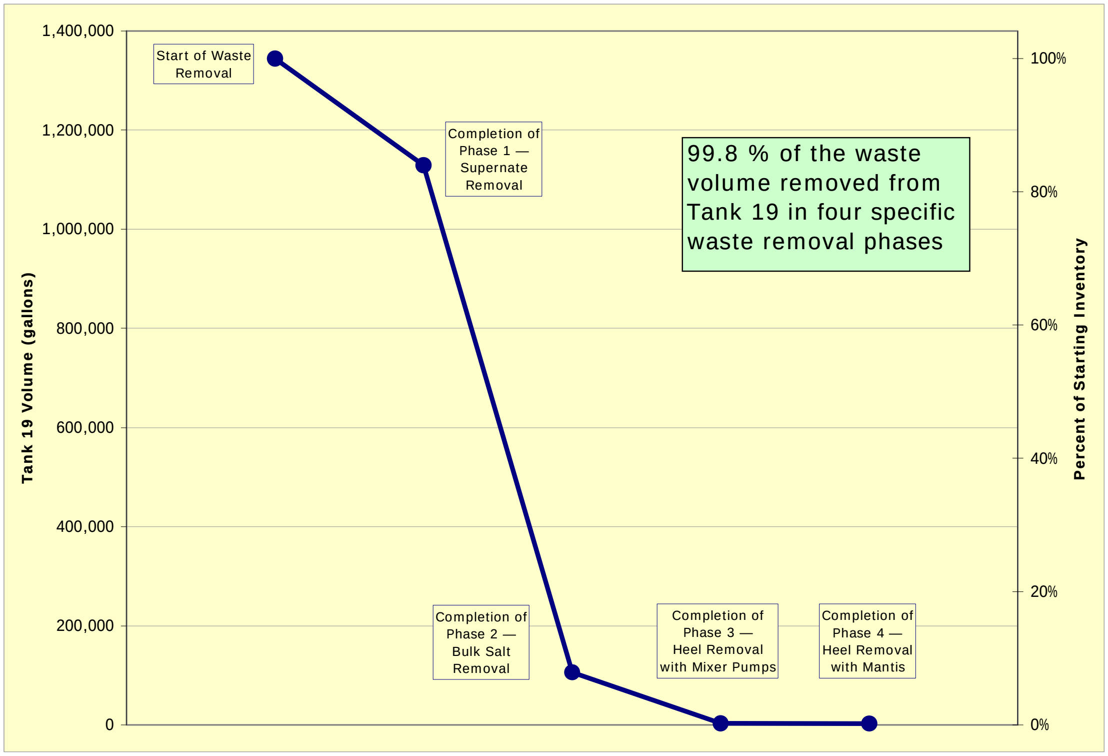


Figure 5.3-2: Tank 19 Waste Removal



Two of the eight Type I tanks in FTF have also undergone extensive heel removal campaigns. These two Type I tanks, Tank 5 and Tank 6, originally contained approximately 34,000 and 25,000 gallons of sludge solids, respectively, at the conclusion of their bulk waste removal campaigns. A Type I tank, such as Tank 5 and Tank 6, represents the most challenging tank for waste removal activities due, in part, to a limited number of access points, horizontal coiling coil runs at the bottom of the waste tank including stacked horizontal runs (often referred to as “fences”) that were “field to fit” during the time of waste tank construction and the presence of roof support columns. Experience in Tank 5 and Tank 6 demonstrates DOE’s successful deployment of innovative technologies capable of removing HRRs even under the most challenging conditions. As described in Section 5.2.2, in 2003, DOE performed a Systems Engineering Evaluation to identify, evaluate and select equipment for waste removal tasks to accelerate waste removal and the removal of waste tanks from service. This evaluation resulted in the selection of new mechanical removal technologies in combination with chemical removal technologies, if necessary.

These new mechanical technologies, in combination with the chemical removal technology, were first utilized in Tank 5 and Tank 6. Throughout the initial mechanical heel removal operations in both Tank 5 and Tank 6, DOE performed numerous waste removal phases consisting of liquid addition followed by SMP operation and then pump down of the slurried waste. Prior to each of these phases, DOE evaluated the results from the previous phases and optimized the removal effectiveness by adjusting the indexing of the SMPs or utilizing a hydro-lance to disperse the solids from areas the SMPs were not effectively cleaning. Once it was determined that the existing mechanical cleaning method had reached a point of diminished effectiveness, chemical heel removal cleaning, described in Section 5.2.2.2, was utilized until it too reached a point of diminished effectiveness. The initial plan for both Tank 5 and Tank 6 was to first perform mechanical cleaning of the tank heels and follow that with chemical heel removal. However, at the conclusion of the chemical heel removal cleaning campaign in both tanks, DOE evaluated the tank conditions and determined that it would be practical to deploy additional cleaning methods within the tanks. In Tank 5, for example, following the implementation of three chemical cleaning cycles using a bulk oxalic acid flowsheet, a new mechanical cleaning method utilizing the three existing SMPs in the tank was deployed. Instead of adding liquid, attempting to slurry the solids and then pumping out the solution in a batch fashion, which required the SMP to be turned off at a certain point in the pump down, the operations were modified to allow for a continuous “feed and bleed” to occur. This new methodology was deployed until it was determined that the new methodology had reached diminished returns. A modified version of this "feed and bleed" mechanical cleaning technology was also utilized in Tank 6 as a final cleaning campaign.

Effective radionuclide removal is expected to be achieved during cleaning of the remaining tanks, and the cleaning and/or flushing of ancillary equipment, for a number of reasons. As discussed above, DOE has successfully removed waste from Type IV tanks (including Tanks 18 and 19) as well as Type I tanks (Tanks 5 and 6), which present the most challenging conditions in the FTF. The cleaning process employed is thorough, and the process is reviewed and documented during cleaning to maximize effectiveness. DOE will continue to use such measures as visual (remote) observation of remaining tank residuals against benchmarks in the tanks (or ancillary equipment), transfer line radiation readings, sampling and analysis, radiation monitoring, and equipment operating parameters to evaluate efficiency and effectiveness of cleaning operations. Moreover, removal activities on a given tank or ancillary structure will not be considered complete until it is clearly demonstrated and documented, for each individual tank or ancillary structure, that further deployment of the technology is no longer useful or sensible, and that other proven technologies have been evaluated and would not be practical. These documented considerations will take into account a variety of factors including such things as the conditions in the specific waste tank or ancillary structure, the status of the FTF and the overall Liquid Waste System (e.g., available waste tank volume), available proven technologies, the potential benefits from long-term risk reduction from continued HRR removal, increased radiation exposure to site workers or the public due to removal activities, increased risk associated with impacts to other DOE missions involving risk-reducing activities, direct monetary expenditures and effectiveness of available technologies[40].

5.4 Conclusion

Removal of HRRs to the MEP in FTF waste tanks and ancillary structures occurs through a systematic progression of waste removal and cleaning activities using proven technologies to a point where further removal of HRRs is not sensible or useful in light of the overall benefit to human health, safety and the environment. The preceding subsections demonstrate that the FTF waste tanks, ancillary structures and their associated stabilized residuals will have had HRRs removed to the MEP at the time of closure.

[40] Typically, the cost-benefit analysis will be relatively simple and will focus on the financial costs for implementation of new technologies versus the decrease in the potential future doses resulting from the additional removal of residuals. [NUREG-1854].

6.0 RADIONUCLIDE CONCENTRATIONS OF STABILIZED RESIDUALS, TANKS AND ANCILLARY STRUCTURES

Section Purpose

The purpose of this section is to demonstrate whether the FTF stabilized residuals at closure will meet concentration limits for Class C low-level waste as set out in 10 CFR Part 61, Section 61.55.

Section Contents

This section provides the methodology and assumptions to demonstrate whether the FTF stabilized residuals at closure meet Class C concentration limits.

Key Points

- DOE is using the NRC guidance in NUREG-1854, Category 3 – Site-Specific Averaging in its approach to determining whether the stabilized residuals meet Class C concentration limits.
- The Category 3 approach involves the use of the site-specific intruder-driller scenarios analyzed in the FTF PA.
- In addition, DOE, in consultation with the NRC[41], has derived site-specific concentration averaging expressions for FTF waste based upon the site-specific intruder-driller scenarios and the guidance in NUREG-1854.
- While DOE believes there is a reasonable basis to conclude that none of the stabilized residuals will exceed the Class C concentration limits in 10 CFR 61.55, DOE nevertheless is also consulting with the NRC on DOE's disposal plans, as described in the Draft FTF 3116 Basis Document, to take full advantage of the NDAA Section 3116 consultation process.

The NDAA Section 3116(a) 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

(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.

[41] NRC suggested during a public Draft FTF NDAA 3116 Basis Document scoping meeting held on July 13-14, 2010 that DOE consider this approach. [SRR-CWDA-2010-00091]

6.1 Waste Concentrations

For the purposes of making a determination under NDAA Section 3116(a)(3), regardless of whether the waste exceeds or does not exceed the concentration limits for Class C low-level waste as set out in 10 CFR 61.55, the Secretary of Energy, in consultation with the NRC, 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 7.0 of this Draft FTF 3116 Basis 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 8.0 of this Draft FTF 3116 Basis Document, information is presented that demonstrates that the waste will be disposed of in compliance with State-approved closure plans.

In situations where the waste exceeds the concentration limits for Class C low-level waste, NDAA Section 3116(a)(3)(B)(iii) provides for consultation with NRC about the disposal plans for the waste. [NDAA_3116]

As discussed in this section, under DOE's disposal plans, the stabilized residuals within the waste tanks and ancillary structures, the waste tanks, and the ancillary structures (including integral equipment) in the FTF are not expected to exceed concentration limits for Class C low-level waste. Nevertheless, DOE is also consulting with the NRC pursuant to the consultation process in NDAA Section 3116(a)(3)(B)(iii) to take full advantage of the consultation process established by NDAA Section 3116. In this regard, DOE is specifically requesting in this Draft FTF 3116 Basis Document that NRC identify what changes, if any, NRC would recommend to DOE's disposal plans as described in the Draft FTF 3116 Basis Document, and DOE intends to consider the NRC recommendations, as appropriate, in the development of DOE's plans. In the following subsections, the methodology for comparison to the Class C concentration limits for radionuclides included in 10 CFR 61.55 is presented. The radionuclides and their associated limits are specified in two separate tables within 10 CFR 61.55 which are reproduced in Table 6.1-1 and Table 6.1-2.

Table 6.1-1: 10 CFR 61.55 Table 1 Class C Concentration Limits

Radionuclides (Long-lived)	Concentration (Ci/m³)
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 nuclides with half-life greater than five years	[1]100
Pu-241	[1]3,500
Cm-242	[1]20,000

(1) Units are in nanocuries per gram.
[10 CFR 61]

Table 6.1-2: 10 CFR 61.55 Table 2 Class C Concentration Limits

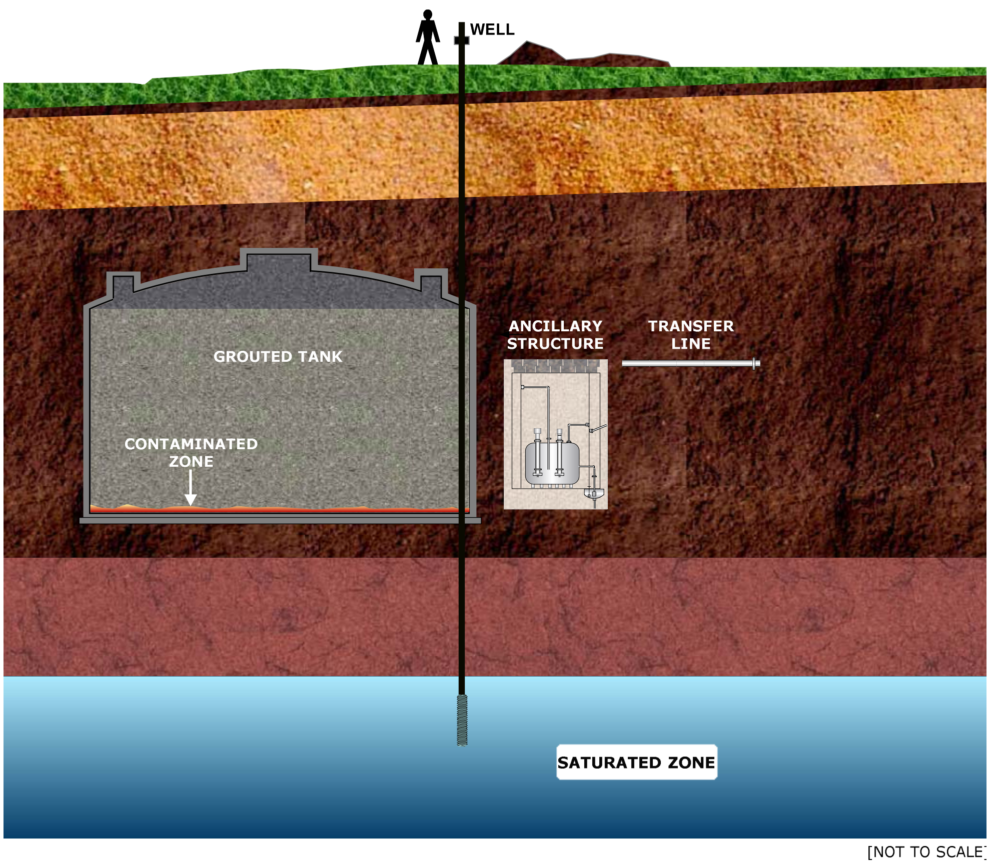
Radionuclides (Short-lived)	Concentration (Ci/m³) Column 1 [Class A]	Column 2 [Class B]	Column 3 [Class C]
Total of all nuclides with less than 5 year half-life	700	(1)	(1)
H-3	40	(1)	(1)
Co-60	700	(1)	(1)
Ni-63	3.5	70	700
Ni-63 in activated metal	35	700	7000
Sr-90	0.04	150	7000
Cs-137	1	44	4600

(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.
[10 CFR 61]

6.2 Approach to Waste Concentrations for F-Tank Farm Residuals

Prior NRC guidance to determine concentrations for comparison with Class C concentration limits of 10 CFR 61.55 was based on excavation as the likely pathway to expose an inadvertent member of the public to waste in a commercial shallow land burial site. [NUREG-1854] Due to the disposal depth of the FTF stabilized residuals in the waste tanks and the ancillary structures, the basement excavation scenario associated with development of 10 CFR 61.55 Table 1 and 2 is not applicable to the FTF waste tanks and ancillary structures. A more appropriate scenario for the purposes of calculation and comparison with Class C concentration limits is one that assumes the inadvertent intruder drills a groundwater well and drills through a waste tank or ancillary structure. (Figure 6.2-1)

Figure 6.2-1: Intruder-Driller Scenarios for Concentration Calculations



Consistent with more recent NRC staff guidance, this Draft FTF 3116 Basis Document follows the Category 3—Site-Specific Averaging approach set forth in NUREG-1854, using the intruder-drilling scenario. [NUREG-1854] This approach utilizes a risk-informed approach that takes into consideration such things as the specific conditions of the FTF site, the final form of the stabilized residuals, site-specific parameters and the final closure configuration.

The following section describes the methodology, and presents the inputs and assumptions, DOE used to compare the concentration of the stabilized residuals in FTF at closure to the Class C concentration limits.

6.3 Methodology

The Category 3—Site-Specific Averaging approach to concentration averaging reflects site-specific conditions of FTF and the final form of the stabilized residuals to account for the volume, concentration and accessibility of the residual material. In order to account for the site-specific conditions relative to

FTF, DOE has developed, consistent with the Category 3—Site-Specific Averaging methodology, averaging expressions for FTF based on the results of the inadvertent intruder analysis performed within the FTF PA. [SRS-REG-2007-00002] As discussed in the following sections, the concentrations of the stabilized residuals have been compared, utilizing these averaging expressions, against the concentration limits for Class C low-level waste as set out in 10 CFR 61.55 Table 1 and Table 2. For Tank 18, this comparison was based on the actual residual inventory in the tank following cleaning. For the ancillary structures, this comparison was based on the projected inventories at closure in the FTF PA.

For purposes of comparison to the Class C concentration limits, and to align with the inputs used in developing the averaging expressions for FTF, the residual inventory used for these calculations are decayed to the inventory that will be present at the time of closure (assumed to be 2020 for the purposes of analysis in the FTF PA). [SRS-REG-2007-00002] As discussed below, the radionuclide concentrations of the stabilized residuals are compared, using the sum of fractions methodology and the FTF averaging expressions, to the concentration limits for Class C low-level waste as set out in 10 CFR 61.55 Table 1 and Table 2.

In order to demonstrate compliance with, among other things, the performance objectives set out in 10 CFR Part 61, Subpart C, as required by NDAA Section 3116(a)(3), DOE developed a performance assessment covering closure activities within FTF. [SRS-REG-2007-00002] To demonstrate compliance with 10 CFR 61.42 the FTF PA is used to demonstrate that there is reasonable assurance the dose to an inadvertent intruder will remain below 500 mrem/yr taking into consideration a variety of intruder scenarios. DOE utilized the inadvertent human intruder analysis in the FTF PA to develop the FTF averaging expressions used for waste classification.

The FTF PA models used to simulate the performance of the FTF closure system take into account the release of radiological contaminants from the waste tanks and the associated ancillary structures in the FTF and simulates transport of the radiological contaminants through soil and groundwater to the assessment point. The models use numerous FTF-specific input parameters to represent the FTF closure system behavior over time. Many of the input parameters are based on site-specific data (e.g., soil and cementitious materials distribution coefficients) used in transport modeling. In addition, site-specific information is used to model the behavior of individual barriers within the FTF, such as the waste tank carbon steel primary tanks and secondary liners (as applicable), and cementitious barriers. Numerous bioaccumulation factors (e.g., soil-to-plant transfer factors), human health exposure parameters (e.g., water ingestion rates, vegetable consumption data) and dose conversion factors are used in the computer modeling to calculate doses for each of the exposure pathways. All of these parameters factor into development of the FTF averaging expressions. A detailed discussion of the FTF PA Intruder analyses is provided in Section 7.1.5 of this Draft FTF 3116 Basis Document and the FTF PA. [SRS-REG-2007-00002]

The stabilized contaminant materials after FTF closure will be primarily located in areas protected by significant materials (e.g., grouted waste tanks, diversion box cell covers and valve box shielding) which are clearly distinguishable from the surrounding soil and make drilling an unlikely scenario based on regional drilling practices. Regional drilling conditions are such that a barrier such as the closure cap erosion barrier, tank top or grout fill are situations that would cause drillers to stop operations and move drilling location. The most vulnerable location for stabilized residuals is in a transfer line which may be near grade-level prior to closure and are of a small size (typically a 3-inch diameter or less) which makes them the most credible stabilized contaminants vulnerable during any intruder drilling operations although the probability of hitting a transfer line is small due to the small surface area of transfer lines versus the large FTF footprint. However, for the purposes of developing averaging expressions for FTF, it is assumed that the structures would be penetrated and that construction of the well would be completed.

The following subsections describe how the sum of fractions is calculated for the FTF waste tanks and ancillary structures.

6.3.1 Methodology Inputs

The following inputs are used for the concentration calculations. Generally, the inputs and assumptions underestimate or do not take credit for certain masses or volumes that would lower the calculated radionuclide contribution to the sum of fractions.

- The residual inventory used for the concentration calculations is the total inventory of the residual material within the waste tank or ancillary structure and includes all material on the floor, walls, piping, cooling coils and any structure that will be abandoned in the tank.
- The residual material layer in the waste tanks and ancillary structures, with the exception of transfer lines, is assumed to be spread evenly across the floor of the waste tank or ancillary structure. The residual material within transfer lines is assumed to be spread evenly over the internal surface of the transfer line.
- The volume of the residual material used in the calculations will be determined on an individual basis for each waste tank or ancillary structure at the time it is being removed from service.
- For the purpose of calculating mass-based concentrations, the density of the residual material within the waste tanks is assumed to be the same as the grout used for stabilization.
- Site-specific averaging expressions for FTF, as described in Section 6.3.2, are utilized for comparison against the concentration limits for Class C low-level waste as set out in Table 1 and Table 2 of 10 CFR 61.55.
- The FTF averaging expressions utilizing transfer line volumes, mass and surface area, have been established on the basis of one linear foot of transfer line piping. This basis was used only for the purpose of standardizing the format for use in performing the calculation and does not impact the calculated radionuclide contributions to the sum of fractions. The ratio between these parameters is constant and the length of pipe does not impact the results of the equations.
- The projected inventories for the ancillary structures (other than the transfer lines discussed above) are bounded by the residuals projected for the waste tanks. [SRS-REG-2007-00002]

6.3.2 Site-Specific FTF Waste Concentration Calculation Averaging Expressions

As described above, the Category 3—Site-Specific Averaging approach to concentration averaging contemplates consideration of site-specific conditions of FTF and the stabilized residuals. In development of Table 1 and 2 of 10 CFR 61.55, the underlying assumption was that the concentration limits and disposal requirements ensure that an inadvertent intruder (e.g., assuming excavation to a depth of 10 feet for construction of a house) would not receive a dose exceeding an equivalent of 500 mrem/yr to the whole body[42]. At closure, the depth of the stabilized residuals within the FTF waste tanks and ancillary structures will be well below (i.e., greater than 10 feet) the FTF closure cap and a robust intruder barrier (e.g., grouted waste tanks, diversion box cell covers and valve box shielding), as described in the FTF PA, will be in place. [SRS-REG-2007-00002] Therefore, the intruder-construction scenario is considered inapplicable. Based on the depth to the stabilized residuals and the presence of a robust intruder barrier, the “Deep waste, intruder barrier” scenario from Table 3-2 of NUREG-1854 is being utilized. In order to account for the site-specific conditions relative to FTF, site-specific averaging expressions for FTF, based on the results of the Inadvertent Intruder Analyses performed within the FTF PA, have been developed.

The FTF PA provides the estimated dose to an intruder who resides within the boundary of the FTF after the period of institutional control (100 years). The intruder is assumed to be exposed via various pathways from water collected from a 1-meter well and from drill cuttings. The groundwater associated with the 1-meter well is contaminated from all the sources within the FTF (waste tanks, transfer lines and other ancillary structures). In addition, drill cuttings that pull up contamination from striking a transfer line are deposited on the ground surface. The base case in the FTF PA assumes that a 3-inch transfer line is penetrated by a driller and the cuttings are spread among the garden – thus an additional source is added to the contaminated well source. The impact of drilling into a 4-inch transfer line has been presented in the FTF PA with respect to the chronic intruder. The impact of drilling into a waste tank was also considered in the FTF PA with respect to the acute intruder, the well driller. Since the likelihood of a well driller penetrating a waste tank is very remote based on local drilling practices that would terminate the

[42] NUREG-1854 states, *“Although multiple scenarios were considered, the limiting intruder scenario that was used (in deriving the concentration limits for waste classification found in Table 1 and Table 2 of 10 CFR 61.55) was an intruder construction scenario (NRC, 1981). This scenario involved excavation of a foundation for a house. Approximately 232 m^3 [8,190 ft^3] of waste was assumed to be exhumed, and the excavation was assumed to be to a depth of 3 m [10 ft]…The underlying assumption of the values in Tables 1 and 2 of 10 CFR 61.55 is that the concentration limits <u>and</u> disposal requirements ensure that an inadvertent intruder would not receive a dose exceeding an equivalent of 5 mSv [500 mrem] to the whole body.”* [NUREG-1854]

drilling once significant resistance is encountered, a chronic intruder was not assessed. [SRS-REG-2007-00002]

Because the stabilized residuals in FTF at closure are expected to have multiple radionuclides from Table 1 and Table 2 of 10 CFR 61.55, the sum of fractions approach for comparing to Class C concentration limits was applied. The sum of fractions approach requires that the concentration of each of the Table 1 and Table 2 radionuclides contained in the stabilized residuals be divided by the appropriate Table 1 or Table 2 Class C concentration limit. The resulting fraction for each of the radionuclides are then totaled for the applicable 10 CFR 61.55 Table 1 or Table 2 radionuclides. If the sum of the fractions is less than 1.0 for the individual tables, the waste is below the Class C concentration limits set out in 10 CFR 61.55. Consistent with the Category 3—Site-Specific Averaging approach, the averaging expressions used to determine the individual radionuclide contribution to the sum of fractions is represented by the following equation:

$$SOF_i = \frac{C_R}{Table\_Value_i} xSiteFactor_i$$

where:

SOF_i	=	Radionuclide "i" contribution to the sum of the fractions
C_R	=	Concentration of the drilled source for radionuclide "i" at closure
$Table\_Value_i$	=	Class C concentration limit from 10 CFR 61.55 Table 1 or Table 2 for radionuclide "i"
$Site\ Factor_i$	=	Site-specific factor for radionuclide "i" based on site-specific conditions within the FTF after closure

The FTF averaging expressions, based on the above equation, that DOE is utilizing are shown below.

6.3.2.1 FTF Waste Tank Waste Concentration Calculation Averaging Expressions

For FTF waste tanks individual radionuclide concentrations for the sum of the fractions calculations are determined with the following equations:

For volume-based concentrations:

$$SOF_i = \frac{1}{Table\_Value_i} x \frac{I_R}{V_R} xSiteFactor_i$$

where:

SOF_i	=	Radionuclide "i" contribution to the sum of the fractions
$Table\_Value_i$	=	Class C concentration limit in Ci/m^3 from 10 CFR 61.55 Table 1 or Table 2 for radionuclide "i"
I_R	=	Total tank residuals inventory for radionuclide "i" decayed to date of closure (i.e., 2020), units in curies
V_R	=	Total volume of residuals remaining in the waste tank, units in m^3
$Site\ Factor_i$	=	Site-specific factor for radionuclide "i" at closure (Table 6.3-1 or Table 6.3-2, see Section 6.3.2.3 for derivation of site-specific factors)

For mass-based concentrations:

$$SOF_i = \frac{1}{Table\_Value_i} x \frac{I_R}{(V_R) \times (\rho_G) \times (1{,}000{,}000)} xSiteFactor_i$$

where:

SOF_i	=	Radionuclide "i" contribution to the sum of the fractions
$Table\_Value_i$	=	Class C concentration limit in nCi/g from 10 CFR 61.55 Table 1 for radionuclide "i"
I_R	=	Total tank residuals inventory for radionuclide "i" decayed to date of closure (i.e., 2020), units in nanocuries
V_R	=	Total volume of residuals remaining in the waste tank, units in m^3
ρ_G	=	Density of stabilizing grout, units in g/cm^3
$Site\ Factor_i$	=	Site-specific factor for radionuclide "i" at closure (Table 6.3-1, see Section 6.3.2.3 for derivation of site-specific factors)

The calculated fractions are totaled for the applicable 10 CFR 61.55 Table 1 or Table 2 radionuclides. If the sum of the fractions is less than 1.0 for the individual tables, the waste is below the Class C concentration limits set out in 10 CFR 61.55.

6.3.2.2 FTF Transfer Line Waste Concentration Calculation Averaging Expressions

For FTF transfer lines the individual radionuclide concentrations for the sum of the fractions calculations are determined with the following equations:

For volume-based concentrations:

$$SOF_i = \frac{1}{Table\_Value_i} x \frac{(I_R) \times (SurfaceArea)}{V_{TL}} xSiteFactor_i$$

where:

SOF_i	=	Radionuclide "i" contribution to the sum of the fractions
$Table\_Value_i$	=	Class C concentration limit in Ci/m^3 from 10 CFR 61.55 Table 1 or Table 2 for radionuclide "i"
I_R	=	Transfer line residuals inventory for radionuclide "i" decayed to date of closure (i.e., 2020), units in Ci/ft^2
Surface Area	=	Transfer line internal surface area for one linear-foot of transfer line piping, units in ft^2
V_{TL}	=	Total volume of piping material for one linear-foot of transfer line piping, units in m^3
$Site\ Factor_i$	=	Site-specific factor for radionuclide "i" at closure (Table 6.3-3 or Table 6.3-4, see Section 6.3.2.3 for derivation of site-specific factors)

For mass-based concentrations:

$$SOF_i = \frac{1}{Table\_Value_i} x \frac{(I_R) \times (SurfaceArea)}{M_{TL}} xSiteFactor_i$$

where:

SOF_i	=	Radionuclide "i" contribution to the sum of the fractions

$Table\_Value_i$	=	Class C concentration limit in nCi/g from 10 CFR 61.55 Table 1 for radionuclide "i"
I_R	=	Transfer line residuals inventory for radionuclide "i" decayed to date of closure (i.e., 2020), units in nCi/ft^2
$Surface\ Area$	=	Transfer line internal surface area for one linear-foot of transfer line piping, units in ft^2
M_{TL}	=	Total mass of piping material for one linear-foot of transfer line piping, units in g
$Site\ Factor_i$	=	Site-specific factor for radionuclide "i" at closure (Table 6.3-3, see Section 6.3.2.3 for derivation of site-specific factors)

The calculated fractions are totaled for the applicable 10 CFR 61.55 Table 1 or Table 2 radionuclides. If the sum of the fractions is less than 1.0 for the individual tables, the waste is below the Class C concentration limits set out in 10 CFR 61.55.

6.3.2.3 Site-Specific Factors for Use in FTF Averaging Expressions

The site-specific factors in the FTF averaging expressions discussed previously were developed to account for site-specific conditions while ensuring the same protection as the concentration limits in Table 1 and Table 2 and the Part 61.55 analysis provides. To develop the site-specific factors, the results of the FTF PA inadvertent intruder analyses along with the FTF PA inventory at closure were utilized. The FTF PA probabilistic model was utilized to determine the dose to the chronic intruder assuming the 1-meter well contaminated source and one of three drill cutting sources including a 3-inch diameter transfer line, a 4-inch diameter transfer line or a waste tank. Because it is the primary contributor to the peak dose in the FTF, Tank 18 was used for the waste tank calculations. The peak dose for each radionuclide, regardless of the time of the peak, was determined and site-specific factors were developed based on the assumed concentrations at closure from the drill cutting source. For the waste tank analysis, the time period evaluated started at 500 years after closure and for transfer lines the time period started at 100 years after closure. Although the FTF closure design does provide a robust intruder barrier that would prevent intrusion into the waste for the first 500 years after closure, for conservatism, the FTF PA evaluated the transfer line scenario beginning at 100 years after closure. Therefore, in selecting the peak doses for the transfer line scenarios, the peak dose was not just the dose after 500 years but also included any individual radionuclide peaks which may have occurred between the 100- and 500-year period.

To determine, based on the inadvertent intruder analysis performed within the FTF PA, the individual radionuclide site-specific factors that would result in an inadvertent intruder under the FTF site-specific conditions receiving an equivalent dose, 500 mrem/yr, to that used in developing the 10 CFR 61.55 concentration limits, the following equation was used:

$$SiteFactor_i = \frac{Table\_Value_i}{C_{PA}} \, x \, \frac{Dose_i}{500 mrem / yr}$$

where:

$Site\ Factor_i$	=	Site-specific factor for radionuclide "i" at closure
$Table\_Value_i$	=	Class C concentration limit from 10 CFR 61.55 Table 1 or Table 2 for radionuclide "i"
C_{PA}	=	Concentration, based on the FTF PA inventory at closure, of the drilled source for radionuclide "i"
$Dose_i$	=	Peak dose, based on results of the FTF PA, that occurs beyond 100 years (for transfer lines) or beyond 500 years (for waste tank) after closure, for radionuclide "i", units in mrem/yr

Using values for the FTF PA closure inventory along with the mass, for mass based limits, and volume, for volume based limits, of the residual material, the radionuclide concentrations at closure were determined. The calculated concentration for each radionuclide, the peak dose for each radionuclide and the equation developed above, were then used to determine the site-specific factors, on a radionuclide basis, for the three different sources (i.e., 3-inch diameter transfer line, a 4-inch diameter transfer line or a waste tank) to be used in the site-specific averaging expression as shown in Section 6.3.2.1 and 6.3.2.2.

Recognizing that the peak dose for a specific radionuclide may be dominated by the contaminated groundwater source and not the drilling source, the site-specific factor for the transfer lines was set at the limiting value based on either the 3-inch diameter transfer line or the 4-inch diameter transfer line.

Table 6.3-1 and Table 6.3-2 provide the FTF waste tank site-specific factors for 10 CFR Table 1 and Table 2 radionuclides, respectively, which are used in the FTF averaging expressions. Table 6.3-3 and Table 6.3-4 provide the FTF transfer line site-specific factors for 10 CFR Table 1 and Table 2 radionuclides, respectively, which are used in the FTF averaging expressions. [SRR-CWDA-2010-00122]

Table 6.3-1: FTF Waste Tank Site-Specific Factors for 10 CFR 61.55 Table 1 Radionuclides

Table 1 Radionuclide	Site-Specific Factor
C-14	4.4E-01
Ni-59	3.0E-01
Nb-94	8.8E-01
Tc-99	1.8E-01
I-129	1.9E+00
Np-237	1.1E+00
Pu-238	1.8E-05
Pu-239	9.0E-04
Pu-240	1.5E-03
Pu-241	5.9E-04
Pu-242	2.1E-03
Pu-244	4.5E-03
Am-241	8.9E-04
Am-242m	1.9E-03
Am-243	1.7E-02
Cm-243	1.5E-04
Cm-244	2.7E-05
Cm-245	2.3E-02
Cm-247	1.0E-01
Cm-248	3.9E-01
Cf-249	5.2E-04

Note: Radionuclides listed are from SRR-CWDA-2009-00045. Only the 10 CFR 61.55 Table 1 radionuclides in the tank inventory are listed.

Table 6.3-2: FTF Waste Tank Site-Specific Factors for 10 CFR 61.55 Table 2 Radionuclides

Table 2 Radionuclide	Site-Specific Factor
H-3	(1)
Co-60	(1)
Ni-63	7.1E-04
Sr-90	3.3E-03
Cs-137	1.3E-02

(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.

Note: Radionuclides listed are from SRR-CWDA-2009-00045. Only the 10 CFR 61.55 Table 2 radionuclides in the tank inventory are listed.

Table 6.3-3: FTF Transfer Line Site-Specific Factors for 10 CFR 61.55 Table 1 Radionuclides

Table 1 Radionuclide	Site-Specific Factor
C-14	3.0E+02
Ni-59	5.2E+00
Nb-94	6.9E-01
Tc-99	3.6E-01
I-129	7.9E+02
Np-237	3.6E+02
Pu-238	5.1E-04
Pu-239	8.6E-02
Pu-240	2.6E-01
Pu-241	6.7E-02
Pu-242	6.5E-01
Pu-244	3.5E+00
Am-241	4.4E-03
Am-242m	1.6E-02
Am-243	2.0E+00
Cm-243	7.0E-05
Cm-244	1.2E-05
Cm-245	7.4E+03
Cm-247	8.8E+13
Cm-248	1.5E+15
Cf-249	1.7E+09

Note: Radionuclides listed are from SRS-REG-2007-00002. Only the 10 CFR 61.55 Table 1 radionuclides in the transfer line inventory are listed.

Table 6.3-4: FTF Transfer Line Site-Specific Factors for 10 CFR 61.55 Table 2 Radionuclides

Table 2 Radionuclide	Site-Specific Factor
H-3	(1)
Co-60	(1)
Ni-63	4.5E-03
Sr-90	1.4E+01
Cs-137	3.9E+01

(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.

Note: Radionuclides listed are from SRS-REG-2007-00002. Only the 10 CFR 61.55 Table 2 radionuclides in the transfer line inventory are listed.

6.4 Waste Concentration Calculations

The following subsections provide calculations of radionuclide concentrations and compare those concentrations to the Class C concentration limits set out in 10 CFR 61.55.

6.4.1 Waste Tank Concentration Calculation

For this calculation, the best estimate residual radionuclide inventory and residual volume for Tank 18 based on actual final characterization results is used. [SRR-CWDA-2010-00117] This inventory was selected on the basis that Tank 18 is the primary contributor to the peak dose in the FTF. The contribution of each radionuclide to the sum of the fractions was calculated using the FTF averaging expressions presented in Section 6.3.2.1 for a mass or volume basis as necessary. For example, using the inventory values for C-14 and Pu-241, the mass- and volume-based averaging expressions from Section 6.3.2.1 become:

For the volume-based C-14 fraction of Class C concentration limit:

$$SOF_{[C-14]} = \frac{1}{8.0\text{E}+00 Ci/m^3} x \frac{(9.0\text{E}-01\, Ci)}{15.2\, m^3} x4.4\text{E}-01 = 3.3\text{E-}03$$

For the mass-based Pu-241 fraction of Class C concentration limit :

$$SOF_{[Pu-241]} = \frac{1}{3.5E+03 nCi/g} x \frac{(2.7\text{E}+11\, nCi)}{(15.2m^3)\times(2.06g/cm^3)\times(1.0\text{E}+06cm^3/m^3)} x5.9\text{E}-04 = 1.5\text{E-}03$$

The remainder of the Table 1 and Table 2 radionuclides are calculated similarly and the results are presented in Table 6.4-1 and Table 6.4-2.

Table 6.4-1: Sum of the Fractions Calculation Using the Closure Inventory for Tank 18 (10 CFR 61.55 Table 1 Radionuclides)

Table 1 Radionuclide	Tank Inventory (Ci)	Class C Concentration Limit[a]	Fraction of Class C Concentration Limit
C-14	9.0E-01	8.0E+00 Ci/m3	3.3E-03
Ni-59	3.3E-01	2.2E+02 Ci/m3	3.0E-05
Nb-94	5.5E-04	2.0E-01 Ci/m3	1.6E-04
Tc-99	9.0E-01	3.0E+00 Ci/m3	3.6E-03
I-129	2.7E-04	8.0E-02 Ci/m3	4.2E-04
Np-237	1.9E-01	1.0E+02 nCi/g	6.7E-02
Pu-238	1.3E+03	1.0E+02 nCi/g	7.5E-03
Pu-239	2.8E+02	1.0E+02 nCi/g	8.1E-02
Pu-240	6.5E+01	1.0E+02 nCi/g	3.1E-02
Pu-241	2.7E+02	3.5E+03 nCi/g	1.5E-03
Pu-242	2.7E-02	1.0E+02 nCi/g	1.8E-05
Pu-244	6.2E-06	1.0E+02 nCi/g	8.9E-09
Am-241	1.6E+02	1.0E+02 nCi/g	4.6E-02
Am-242m	3.8E-02	1.0E+02 nCi/g	2.3E-05
Am-243	2.3E+00	1.0E+02 nCi/g	1.3E-02
Cm-243	1.8E-02	1.0E+02 nCi/g	8.6E-07
Cm-244	9.8E+01	1.0E+02 nCi/g	8.5E-04
Cm-245	1.2E-02	1.0E+02 nCi/g	8.8E-05
Cm-247	2.1E-06	1.0E+02 nCi/g	6.7E-08
Cm-248	9.5E-05	1.0E+02 nCi/g	1.2E-05
Cf-249	2.3E-03	1.0E+02 nCi/g	3.8E-07
		Sum of the Fractions	**2.5E-01**

[a] Values from 10 CFR 61.55 Table 1.
Note: Inventory values are from SRR-CWDA-2010-00117. Only the 10 CFR 61.55 Table 1 radionuclides in the tank inventory are listed.

Table 6.4-2: Sum of the Fractions Calculation Using the Closure Inventory for Tank 18 (10 CFR 61.55 Table 2 Radionuclides)

Table 2 Radionuclide	Tank Inventory (Ci)	Class C Concentration Limit(Ci/m^3)[a]	Fraction of Class C Concentration Limit
H-3	1.4E-02	(1)	NA
Co-60	1.2E+00	(1)	NA
Ni-63	1.7E+01	7.0E+02	1.1E-06
Sr-90	3.2E+03	7.0E+03	7.8E-05
Cs-137	1.2E+04	4.6E+03	1.7E-03
		Sum of the Fractions	**1.8E-03**

[a] Values from 10 CFR 61.55 Table 2.
(1) There are no limits established for these radionuclides in Class C waste.
NA - Not Applicable
Note: Inventory values are from SRR-CWDA-2010-00117. Only the 10 CFR 61.55 Table 2 radionuclides in the tank inventory are listed.

The sum of the fractions from Table 6.4-1 and Table 6.4-2 are 2.5E-01 and 1.8E-03, respectively. Using the inventory values assumed for this calculation, the stabilized residuals would not exceed Class C concentration limits.

6.4.2 Transfer Line Concentration Calculation

For this calculation, the FTF PA estimated residual inventory for a 3-inch transfer line is used[43]. [SRS-REG-2007-00002] This inventory was selected on the basis that over 99% of the transfer lines located in FTF are 3-inch transfer lines or smaller. The contribution of each radionuclide to the sum of the fractions was calculated using the averaging expressions presented in Section 6.3.2.2 for a mass or volume basis as necessary. For example, using the inventory values for C-14 and Pu-241, and assuming schedule 40 3-inch piping, the mass- and volume-based averaging expressions from Section 6.3.2.2 become:

For the volume-based C-14 fraction of Class C concentration limit:

$$SOF_{[C-14]} = \frac{1}{8.0\mathrm{E}+00Ci/m^3} x \frac{(4.22\mathrm{E}-08\,Ci/\,ft^2)\times(8.03\mathrm{E}-01\,ft^2)}{4.38\mathrm{E}-04\,m^3} x3.0\mathrm{E}+02 = 2.9\text{E-}03$$

For the mass-based Pu-241 fraction of Class C concentration limit:

$$SOF_{[Pu-241]} = \frac{1}{3.5E+03nCi/g} x \frac{(1.23\mathrm{E}+05\,nCi/\,ft^2)\times(8.03\mathrm{E}-01\,ft^2)}{3{,}438\,g} x6.7\mathrm{E}-02 = 5.5\text{E-}04$$

The remainder of the Table 1 and Table 2 radionuclides are calculated similarly and the results are presented in Table 6.4-3 and Table 6.4-4.

[43] The inventory used for this calculation is based on the 3-inch transfer line inventory developed to support the FTF PA. [SRS-REG-2007-00002]

Table 6.4-3: Sum of the Fractions Calculation Using the FTF PA, Revision 1 Inventory for a 3-inch Transfer Line (10 CFR 61.55 Table 1 Radionuclides)

Table 1 Radionuclide	3-inch Transfer Line Inventory[44] (Ci/ft^2)	Class C Concentration Limit[a]	Fraction of Class C Concentration Limit
C-14	4.22E-08	8.0E+00 Ci/m3	2.9E-03
Ni-59	1.66E-06	2.2E+02 Ci/m3	7.2E-05
Nb-94	3.73E-08	2.0E-01 Ci/m3	2.4E-04
Tc-99	1.48E-05	3.0E+00 Ci/m3	3.3E-03
I-129	7.03E-11	8.0E-02 Ci/m3	1.3E-03
Np-237	7.81E-08	1.0E+02 nCi/g	6.6E-02
Pu-238	1.82E-04	1.0E+02 nCi/g	2.2E-04
Pu-239	7.13E-05	1.0E+02 nCi/g	1.4E-02
Pu-240	2.62E-05	1.0E+02 nCi/g	1.6E-02
Pu-241	1.23E-04	3.5E+03 nCi/g	5.5E-04
Pu-242	2.16E-07	1.0E+02 nCi/g	3.3E-04
Pu-244	1.02E-10	1.0E+02 nCi/g	8.3E-07
Am-241	5.79E-04	1.0E+02 nCi/g	6.0E-03
Am-242m	8.32E-07	1.0E+02 nCi/g	3.1E-05
Am-243	9.45E-08	1.0E+02 nCi/g	4.4E-04
Cm-243	1.39E-08	1.0E+02 nCi/g	2.3E-09
Cm-244	2.67E-05	1.0E+02 nCi/g	7.5E-07
Cm-245	3.43E-10	1.0E+02 nCi/g	5.9E-03
Cm-247	1.28E-22	1.0E+02 nCi/g	2.6E-05
Cm-248	2.95E-23	1.0E+02 nCi/g	1.0E-04
Cf-249	8.00E-25	1.0E+02 nCi/g	3.2E-12
		Sum of the Fractions	**1.2E-01**

[a] Values from 10 CFR 61.55 Table 1.
Note: Inventory values are from SRS-REG-2007-00002. Only the 10 CFR 61.55 Table 1 radionuclides in the transfer line inventory are listed.

Table 6.4-4: Sum of the Fractions Calculation Using the FTF PA, Revision 1 Inventory for a 3-inch Transfer Line (10 CFR 61.55 Table 2 Radionuclides)

Table 2 Radionuclide	3-inch Transfer Line Inventory[45] (Ci/ft^2)	Class C Concentration Limit(Ci/m^3) [a]	Fraction of Class C Concentration Limit
H-3	8.08E-07	(1)	NA
Co-60	2.31E-05	(1)	NA
Ni-63	1.38E-04	7.0E+02	1.6E-06
Sr-90	3.10E-02	7.0E+03	1.1E-01
Cs-137	5.79E-03	4.6E+03	9.0E-02
		Sum of the Fractions	**2.0E-01**

[a] Values from 10 CFR 61.55 Table 2.
(1) There are no limits established for these radionuclides in Class C waste.
NA - Not Applicable
Note: Inventory values are from SRS-REG-2007-00002. Only the 10 CFR 61.55 Table 2 radionuclides in the tank inventory are listed.

The sum of the fractions from Table 6.4-3 and Table 6.4-4 are 1.2E-01 and 2.0E-01, respectively. Using the inventory values assumed for this calculation, the stabilized residuals in this 3-inch transfer line would not exceed Class C concentration limits.

[44] See footnote 43.
[45] See footnote 43.

6.5 Conclusion

As demonstrated by the above discussion, the stabilized FTF waste at closure meets concentration limits for Class C low-level waste as set out in 10 CFR 61.55. Nevertheless, DOE is also consulting with the NRC pursuant to the consultation process in NDAA Section 3116(a)(3)(B)(iii) to take full advantage of the consultation process established by NDAA Section 3116. In this regard, DOE is specifically requesting in this Draft FTF 3116 Basis Document that NRC identify what changes, if any, NRC would recommend to DOE's disposal plans as described in the Draft FTF 3116 Basis Document, and DOE intends to consider the NRC recommendations, as appropriate, in the development of DOE's plans.

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 stabilized residuals in the FTF waste tanks and ancillary structures, the waste tanks, and the ancillary structures (including integral equipment) will be disposed of in compliance with the performance objectives for land disposal of low-level waste found in 10 CFR Part 61, Subpart C, Sections 61.41 through 61.44.

Section Contents

This section describes key parameters and results from the FTF PA that demonstrate compliance with the performance objectives in 10 CFR 61.41 and 10 CFR 61.42, DOE regulatory and contractual requirements which ensure compliance with 10 CFR 61.43 and relevant factors of FTF siting, design, use, operation and closure, which ensure compliance with 10 CFR 61.44.

Key Points

- Based on the FTF PA, there is reasonable assurance that the 10 CFR 61.41 and 10 CFR 61.42 performance objectives will be met.
- The DOE is using an assumed institutional control period of 100 years for the purpose of analysis although the SRS Land Use Plan [PIT-MISC-0041] calls for Federal ownership in perpetuity.
- For the purpose of calculating doses to a member of the public, a 100-meter buffer zone around the FTF boundary is assumed.
- The FTF PA evaluates a performance period of 10,000 years, and provides additional data beyond this period of time, for the purpose of making risk-informed decisions related to the closure of FTF.
- The FTF PA analysis demonstrates compliance with the performance objective in 10 CFR 61.41 based on compliance with a 25 mrem/yr peak all-pathways Total Effective Dose Equivalent (TEDE) to a hypothetical member of the public.
- The FTF PA analysis demonstrates compliance with the performance objective in 10 CFR 61.42 based on compliance with a dose limit of 500 mrem/yr to a future hypothetical inadvertent intruder of the closed FTF.
- The DOE regulatory and contractual requirements for FTF facilities and activities establish dose limits based on 10 CFR 835 and relevant DOE Orders. These dose limits correspond to the radiation protection standards set out in 10 CFR 20, as cross-referenced in 10 CFR 61.43.
- The FTF waste tanks will be filled with grout to provide long-term stability.
- The FTF ancillary structures may be filled with appropriate fill materials, as necessary, to provide long-term stability. There are currently no plans to grout or fill the FTF transfer lines.

The NDAA Section 3116(a) 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)(i) [Will be disposed of] 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, Subpart C Performance Objectives

The 10 CFR 61, Subpart C, Sections 61.40 through 61.44 detail performance objectives the NRC established for land disposal of radioactive waste. These performance objectives address protection of the general population from radioactivity releases, individuals from inadvertent intrusion on the disposal site, protection of workers and the public during disposal facility operations and the stability of the disposal site after closure. The following subsections discuss the 10 CFR 61.40 through 10 CFR 61.44 performance objectives.

7.1.1 10 CFR 61.40

10 CFR 61.40 states:

> *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 §§61.41 through 61.44.*

The 10 CFR 61.40 general provisions require "reasonable assurance" that exposures are within the limits of the subsequent performance objectives for 10 CFR 61.41 through 10 CFR 61.44 and are discussed below.

7.1.2 F-Tank Farm Performance Assessment [SRS-REG-2007-00002]

The DOE has developed an FTF PA which provides the technical basis and results demonstrating that the 10 CFR 61.41 and 10 CFR 61.42 performance objectives will be met after FTF closure. These analyses were performed using a variety of modeling codes including the PORFLOW deterministic code and GoldSim probabilistic code. As required by the DOE Manual 435.1-1, maintenance of the FTF PA will include future revisions to incorporate new information, update model codes, etc., as appropriate.

The FTF PA details the analysis performed to provide "reasonable assurance" that the stabilized residuals, waste tanks and ancillary structures will be disposed of in compliance with the 10 CFR 61.41 and 61.42 performance objectives in conjunction with closure of the FTF. Individual FTF system behaviors are evaluated within the FTF PA for various waste tank and ancillary structure configurations, including a base (expected) case, which provides results reflecting the closure system behavior. The FTF PA provides the development and calculation of the following doses:

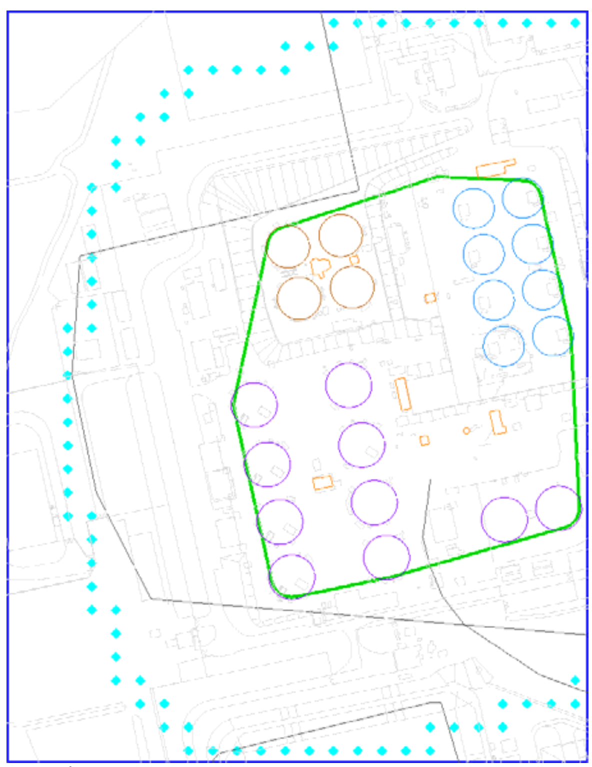
- potential radiological doses to a hypothetical member of the public and
- potential radiological doses to a hypothetical inadvertent intruder.

These calculations were performed to provide information over a minimum of 10,000 years. In addition, uncertainty and sensitivity analyses were used to ensure reasonably conservative information is available to develop risk-informed conclusions related to the closure of FTF.

The following general definitions and assumptions are used in the FTF PA and will serve as the basis for future FTF PA revisions.

FTF Boundary: The FTF boundary is the line of demarcation enclosing the FTF waste tanks (Figure 7.1-1).

Figure 7.1-1: FTF Boundary and Buffer Zone



Legend:
Boundary   Buffer Zone (100 meters from FTF

Buffer Zone: The buffer zone is the radial area that encompasses the FTF 100 meters from its boundary (Figure 7.1-1).

Institutional Control: Institutional control is a 100-year period in which DOE retains ownership and control of FTF such that FTF facility maintenance and controls will be performed to prevent inadvertent intrusion and protect public health and the environment[46].

Performance Period: The performance period is the 10,000 years following final closure activities. The FTF PA evaluates impacts of closure activities during the performance period (i.e., 10,000 years) after closure. The DOE has evaluated for periods beyond 10,000 years in the FTF PA to further inform closure decisions[47].

Uncertainty and Sensitivity: Uncertainty and sensitivity analyses are employed to consider the effects of uncertainties in the conceptual models and sensitivity of simulation results to the parameters in the mathematical models. The sensitivity analyses consider sensitivity of results to parameters both individually and collectively. The FTF PA includes uncertainty and sensitivity analyses.

7.1.3 10 CFR 61.41

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 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems 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 is reasonably achievable.*

7.1.3.1 General Approach

To Demonstrate compliance with this performance objective, a 25 mrem/yr peak all-pathways TEDE is used, rather than individual organ doses. The NRC states in NUREG-1854 that use of the 25 mrem/yr all-pathways TEDE is used by the NRC in making the assessment for compliance with the whole body, thyroid and any other organ limits in 10 CFR 61.41 and is protective of human health and the environment.

In addition NUREG-1854 states:

> *…incidental waste determinations may use total effective dose equivalent (TEDE) without specific consideration of individual organ doses. Intruder calculations should be based on 5 mSv [500 mrem] TEDE limit, without specific consideration of individual organ doses, to ensure consistency between 10 CFR 61.41 and 10 CFR 61.43. Because of the tissue weighting factors and the magnitude of the TEDE limit, specific organ dose limits are not necessary for protection from deterministic effects.*

The hypothetical future member of the public is assumed to be located at the boundary of the DOE controlled area until the assumed active institutional control period ends (i.e., 100 years after closure), at which point the receptor is assumed to move to the point of maximum exposure at or outside of the FTF 100-meter buffer zone. For the purposes of demonstrating compliance with 10 CFR 61.41, the peak all-pathways dose at or outside of the 100-meter buffer zone during the 10,000-year performance period will be used.

The pathways for release to a member of the public considered in the FTF PA analyses are discussed below. The scenarios are not assumed to occur until after the assumed 100-year institutional control period ends. [SRS-REG-2007-00002]

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[46] To ensure a conservative analysis relative to potential public risk, DOE is using an institutional control period of 100 years. As described in Section 7.1.7.4 of this Draft FTF 3116 Basis Document, the SRS Land Use Plan requires Federal ownership and control of the site well beyond 100 years after closure.

[47] The FTF PA evaluates a performance period of 10,000 years, and provides additional data beyond this period of time, for the purpose of making risk-informed decisions related to the closure of FTF.

7.1.3.2 Public Release Pathways Dose Analysis

The primary water sources for the member of the public release pathways are either a well drilled into the groundwater aquifers or a GSA stream. The bounding dose scenario and associated exposure pathways for the member of the public was determined to be an agricultural resident who uses water from a well for domestic purposes. The bounding public dose scenario and associated exposure pathways are documented in the FTF PA. The following exposure pathways involving the use of contaminated[48] well water were considered (Figure 7.1-2):

- direct ingestion of well water,
- ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that drink well water,
- ingestion of vegetables grown in garden soil irrigated with well water,
- ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that eat fodder from pasture irrigated with well water, and
- ingestion and inhalation of well water while showering.

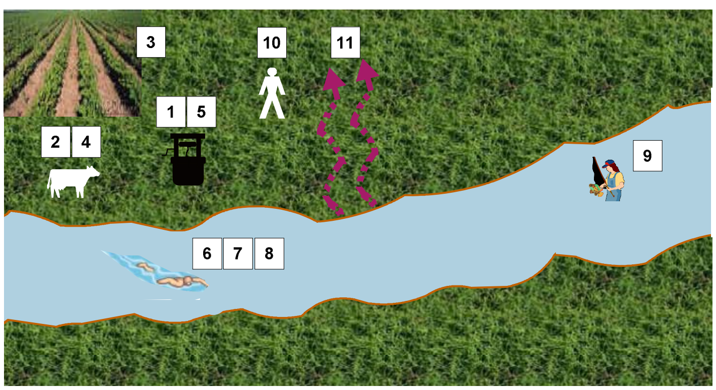
The following exposure pathways involving the use of contaminated surface water (from the applicable stream) for recreational use are assumed to occur:

- direct irradiation during recreational activities (e.g., swimming, fishing) from stream water,
- dermal contact with stream water during recreational activities (e.g., swimming, fishing),
- incidental ingestion and inhalation of stream water during recreational activities, and
- ingestion of fish from the stream water.

Additional exposure pathways could involve releases of radionuclides into the air from the water taken from the well (i.e., volatile radionuclides such as C-14 and I-129). Exposures from the air pathway in the FTF PA are:

- direct plume shine and
- inhalation.

Figure 7.1-2: Scenario with Well Water as Primary Source



[NOT TO SCALE]

1. Direct ingestion of well water
2. Ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that drink well water
3. Ingestion of vegetables grown in garden soil irrigated with well water
4. Ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that eat fodder from pasture irrigated with well water
5. Ingestion and inhalation of well water while showering
6. Direct irradiation during recreational activities (e.g., swimming, fishing) from stream water
7. Dermal contact with stream water during recreational activities (e.g., swimming, fishing)
8. Incidental ingestion and inhalation of stream water during recreational activities
9. Ingestion of fish from the stream water
10. Direct plume shine
11. Inhalation

Secondary and indirect pathways that contribute relatively minor doses to a receptor when compared to direct pathways (e.g., ingestion of milk and meat) include:

- inhalation of well water used for irrigation,
- inhalation of dust from the soil that was irrigated with well water,
- ingestion of soil that was irrigated with well water, and
- direct radiation exposure from radionuclides deposited on the soil that was irrigated with well water.

[48] Contaminated in this context refers to radioactive materials that have been projected to migrate from the closed FTF.

The point of assessment for the groundwater wells used in the member of the public scenario is located 100 meters from the FTF, as shown in Figure 7.1-1. The peak concentrations used to determine the peak doses for the member of the public exposure pathways are calculated and documented in the FTF PA. The groundwater concentrations used are peak concentrations for each radionuclide at the given point of assessment, from any of the aquifers.

The groundwater concentrations were calculated based on the FTF PA conceptual model. The conceptual model is used to simulate the performance of the FTF closure system during the 10,000-year period and beyond following FTF closure and is comprised of both near-field and far-field models that represent the FTF closure system and the environmental media through which radionuclides may migrate. The conceptual model was used to simulate transport of the radiological contaminants through soil and groundwater to the 100-meter assessment point and nearby streams.

The conceptual model used numerous FTF-specific input parameters to represent the FTF closure system behavior over time. Many of the input parameters are based on site-specific data (e.g., soil and cementitious materials distribution coefficients) used in transport modeling. In addition, site-specific information is used to model the behavior of individual barriers within the FTF conceptual model, such as the waste tank carbon steel primary tanks and secondary tanks or annular pans, if applicable, and cementitious barriers. The models and model inputs used in the FTF conceptual model to calculate groundwater concentrations are described in detail in the FTF PA.

The groundwater peak dose for the member of the public is calculated in the FTF PA using site-specific input parameters, and the bounding dose scenario exposure pathways and peak concentrations discussed previously. Numerous bioaccumulation factors (e.g., soil-to-plant transfer factors), human health exposure parameters (e.g., water ingestion rates, vegetable consumption data) and dose conversion factors are used in the computer modeling to calculate doses for each of the exposure pathways, and these parameters are documented in the FTF PA.

An air-pathway analysis was also performed in addition to the groundwater analysis to determine the dose contribution from the air pathway. This analysis utilized an atmospheric screening methodology to identify radionuclides for air-pathway modeling based on waste tank radionuclide projected inventories and the limited number of radionuclides susceptible to volatilization. Computer modeling was performed to calculate the transport of radionuclides through the stabilized waste form and the closure cap to the surface of FTF. An air-pathway dose was then calculated based on the specific curies of each radionuclide assumed to be transported to the surface of FTF. The air pathway analysis and groundwater analysis are combined to determine an all-pathways peak dose for a member of the public.

In addition to the all-pathways peak dose analyses, additional analyses are provided in the FTF PA to characterize the context of uncertainty and sensitivity surrounding the FTF PA all-pathways peak dose results. These evaluations focused on the key uncertainties and sensitivities identified during calculation of the member of the public dose. The uncertainty analyses provide information regarding how collective uncertainty in model input parameters is propagated through the model to the various model results. The sensitivity analyses provide information as to how various individual input parameters affect dose results. Together the uncertainty and sensitivity analyses provide assurance that the impacts of variability and uncertainty in the member of the public dose analyses are understood and addressed.

The uncertainty and sensitivity analyses were primarily performed using a probabilistic model, with some additional single parameter sensitivity analyses (e.g., inventory sensitivity analysis, distribution coefficient sensitivity analysis, alternate configuration sensitivity analysis) performed through deterministic modeling. The probabilistic model allows for variability of multiple parameters simultaneously, so concurrent effects of changes in the model can be analyzed. The deterministic model single parameter analyses provide a method to evaluate the importance of the uncertainty around a single parameter of concern. The deterministic model single parameter analyses included a comprehensive barrier analyses that identified barriers to waste migration and evaluated the capabilities of each barrier as understood from the results of the FTF PA. The barrier analyses assessed the contribution of individual barriers (e.g., closure cap, grout, contamination zone, waste tank liner and waste tank concrete) by comparing contaminant flux results under various barrier conditions. Using both probabilistic and deterministic models for sensitivity analysis versus a single approach provides additional information concerning which parameters are of most importance to the FTF PA model. [SRS-REG-2007-00002]

7.1.3.3 Results of the Analysis

The FTF PA modeling was used to determine an all-pathways dose to a member of the public for comparison with the 10 CFR 61.41 performance objectives. The FTF PA, Revision 1, projected the peak all-pathways dose to the FTF public receptor (i.e., individual greater than or equal to 100 meters from the FTF) to be 2.5 mrem/year, which is less than the 25 mrem/yr performance objective during the 10,000-year performance period[49]. The 25 mrem/yr peak all-pathways dose includes the groundwater pathways and air pathways associated with all 22 FTF waste tanks and associated ancillary structures with the groundwater pathway being the most significant contributor. [SRS-REG-2007-00002]

The FTF PA modeling performed for the uncertainty and sensitivity analyses was used to determine the projected dose to the FTF public receptor for the base case, as well as other tank configurations, over a wide range of variability in input parameters. The FTF PA, Revision 1, uncertainty analysis projected a 4.8 mrem/yr peak of the mean dose to a member of the public within the 10,000-year performance period for 1,000 base case realizations. In addition, the FTF PA, Revision 1, uncertainty and sensitivity analysis projected a 14 mrem/yr peak of the mean dose to a member of the public within the 10,000-year performance period when all modeled tank and ancillary structure configurations are considered. In both cases, the peak of the mean dose is less than the 25 mrem/yr performance objective within the 10,000-year performance period[50]. [SRS-REG-2007-00002]

Since there are 29 unique and independent inventory sources modeled in the FTF model, there is significant temporal and spatial complexity inherent in the modeling system. The uncertainty and sensitivity analyses (in particular the barrier analyses) demonstrated that the impact of individual parameters and/or specific barriers can be variable, with the impact depending to a great extent upon the tank type and/or radionuclide involved[51]. [SRS-REG-2007-00002] Additional discussion regarding the radionuclides most impacting the dose results can be found in Section 5.1 of this Draft FTF 3116 Basis Document.

Compliance with the 25 mrem/yr peak all-pathways dose limit is demonstrated by the peak FTF all-pathways base (expected) case dose of 2.5 mrem/yr calculated in the FTF PA for the 10,000-year performance period. In addition, the uncertainty and sensitivity analyses included in the FTF PA provide sufficient information on parameter sensitivities and modeling uncertainties to provide reasonable assurance that the 25 mrem/yr all-pathways dose limit will be met during the 10,000-year performance period. [SRS-REG-2007-00002]

7.1.3.4 Conclusion

As demonstrated in the preceding discussion, reasonable assurance is provided that the performance objective at 10 CFR 61.41 will not be exceeded.

7.1.4 As Low As Reasonably Achievable

The NRC performance objective in 10 CFR 61.41 also provides that reasonable effort shall be made to maintain releases of radioactivity in effluents to the environment as low as reasonable achievable (ALARA). The FTF PA was developed in accordance with the comparable requirement in DOE Manual 435.1-1:

> *Performance assessments shall include a demonstration that projected releases of radionuclides to the environment shall be maintained as low as reasonably achievable (ALARA).*

[49] The 2.5 mrem/year dose reflects the current results of the base (expected) case through deterministic (PORFLOW) modeling for FTF PA, Revision 1, and is estimated to occur at year 10,000 following closure of FTF. The peak FTF all-pathways base case dose provided in the FTF PA is not to be considered a limit. As required by DOE Manual 435.1-1, maintenance of the FTF PA will include future updates to incorporate new information, update model codes, analysis of actual residual inventories, etc., as appropriate.

[50] The 4.8 mrem/yr and 14 mrem/yr doses reflect the current results of the uncertainty analyses projected through probablilistic modeling for FTF PA, Revision 1. The peak of the means doses provided in the FTF PA are not to be considered limits. As required by DOE Manual 435.1-1, maintenance of the FTF PA will include future updates to incorporate new information, update model codes, analysis of actual residual inventories, etc., as appropriate.

[51] Detailed discussion of the uncertainty and sensitivity analyses can be found in Section 5.6 (pages 568 to 749) of the FTF PA Revision 1. The impact of the various integrated conceptual model segments on the dose results is detailed in Section 7.1 of the FTF PA Revision 1. [SRS-REG-2007-00002]

As discussed previously, the FTF PA provides the information to demonstrate compliance with the 25 mrem all-pathways dose performance objective, including stabilization[52] of the residual waste using grout to minimize releases to the environment. [SRS-REG-2007-00002] Section 5.2 of this Draft FTF 3116 Basis Document provides the information to show that the residual waste inventory in the waste tanks will be removed to the maximum extent practical. These factors demonstrate reasonable effort to maintain releases of radioactivity ALARA.

7.1.5 10 CFR 61.42

Provisions in 10 CFR 61.42 require:

> *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.*

7.1.5.1 General Approach

Demonstration of compliance with a 500 mrem/yr peak intruder dose will be used to demonstrate compliance with the 10 CFR 61.42 performance objective. The requirement of 10 CFR 61.42 exhibits the NRC's intent to protect persons who inadvertently intrude on the waste. The performance objective does not place quantitative limits on exposure. However, the 10 CFR 61 Final EIS suggests a dose limit of 500 mrem/yr for the waste classification scheme in 10 CFR 61.55. Consequently, the NRC uses 500 mrem/yr dose limit for evaluating impacts to an inadvertent intruder for purposes of 10 CFR 61.42[53]. [NUREG-0945, NUREG-1854]

The 10 CFR 61.42 regulations do not specify use of a particular scenario to demonstrate compliance. In developing intruder scenarios, the DOE assumes that humans will continue land use activities, which are consistent with past (e.g., recent decades) and present regional practices, after the end of the assumed active institutional control period.

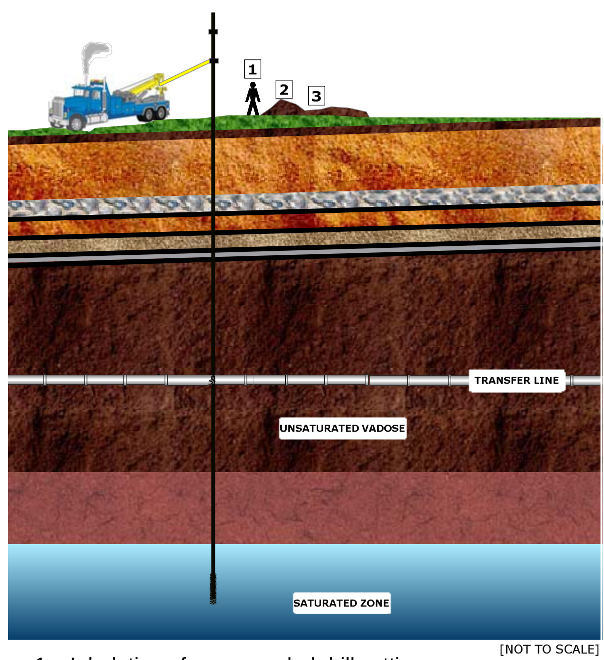
To calculate the dose to an inadvertent intruder, potential intruder scenarios were considered in the FTF PA and the bounding Acute Intruder and Chronic Intruder dose scenarios were determined to be the Acute Intruder-Drilling Scenario and Chronic Intruder-Agricultural (Post-Drilling) Scenario respectively.

7.1.5.2 Acute Intruder-Drilling Scenario

The bounding Acute Intruder scenario analyzed in the FTF PA is an Acute Intruder-Drilling Scenario. This scenario assumes that after the end of active institutional controls a well is drilled within the FTF buffer zone. The well is assumed to be used for domestic water use and irrigation. Because no other natural resources have been identified in the FTF, no additional drilling scenarios are considered. In a drilling scenario, an Acute Intruder is assumed to be the person or persons who install the well and are exposed to drill cuttings during well installation.

The exposure pathways for this acute drilling scenario include (Figure 7.1-3):

Figure 7.1-3: Acute Intruder-Drilling Scenario



1. Inhalation of resuspended drill cuttings
2. External exposure to the drill cuttings
3. Inadvertent drill cuttings ingestion

[52] Stabilization of the FTF waste tanks will be carried out by filling the tanks with grout after completion of waste removal activities. Ancillary structures will be filled, as necessary, to prevent subsidence of the structure or final closure cap. The DOE currently does not plan to grout the FTF transfer lines.

[53] For additional information, DOE Manual 435.1-1 also establishes a 100 mrem/yr chronic dose limit for evaluating impacts of an inadvertent intruder.

- inhalation of resuspended drill cuttings,
- external exposure to the drill cuttings, and
- inadvertent drill cuttings ingestion.

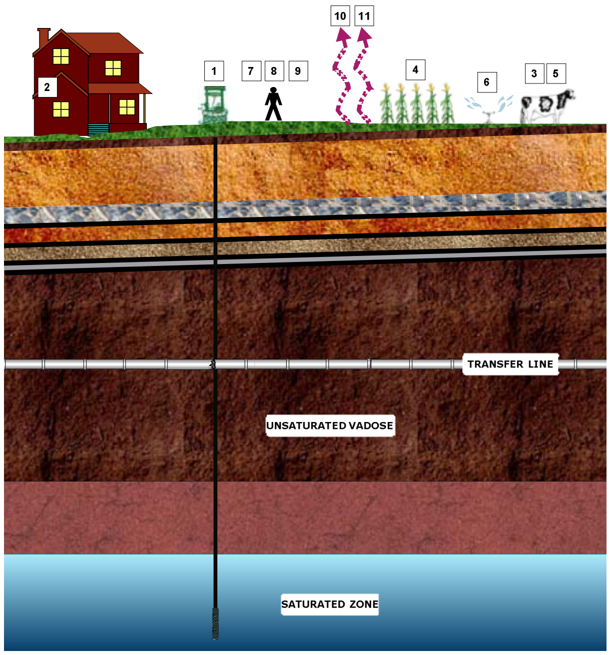
7.1.5.3 Chronic Intruder-Agricultural (Post-Drilling) Scenario

The bounding chronic intruder scenario analyzed in the FTF PA is Chronic Intruder-Agricultural (Post-Drilling) Scenario. This scenario assumes that after the end of active institutional controls, a farmer lives within the FTF buffer zone and consumes food crops grown, and meat and milk from animals raised there, using water from a well drilled within the FTF buffer zone. The Chronic Intruder-Agricultural Scenario (i.e., post-drilling) is an extension of the Acute Intruder-Drilling Scenario. This scenario assumes that an intruder lives in a building near the well drilled as part of the intruder-drilling scenario and engages in agricultural activities within the FTF buffer zone. Excavation to the surface of the stabilized contaminants in the waste tanks was not considered credible because its depth is more than 40 feet below the closure cap. Therefore, the intruder-agricultural scenario was retained for the ancillary structures inventory and specifically a waste transfer line. This is because it is less protected than a diversion box, valve box or pump pit, which are protected by thick shield covers, equaling several feet of concrete. The soil used for agricultural purposes is assumed to be contaminated by both drill cuttings and well water used for irrigation.

The intruder is exposed to (Figure 7.1-4):

- direct ingestion of well water,
- ingestion and inhalation of well water while showering,
- ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that drink well water,
- ingestion of vegetables grown in garden soil irrigated with well water and containing contaminated drill cuttings,
- ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that eat fodder from pasture irrigated with well water,
- inhalation of well water used for irrigation,
- inhalation of dust from the soil that was irrigated with well water,
- ingestion of soil that was irrigated with well water, and
- direct radiation exposure from radionuclides deposited on the soil that was irrigated with well water.

Figure 7.1-4: Chronic Intruder-Agricultural (Post-Drilling) Scenario



1. Direct ingestion of well water
2. Ingestion and inhalation of well water while showering
3. Ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that drink well water
4. Ingestion of vegetables grown in garden soil irrigated with well water and containing contaminated drill cuttings
5. Ingestion of milk and meat from livestock (e.g., dairy and beef cattle) that eat fodder from pasture irrigated with well water
6. Inhalation of well water used for irrigation
7. Inhalation of dust from the soil that was irrigated with well water
8. Ingestion of soil that was irrigated with well water
9. Direct radiation exposure from radionuclides deposited on the soil that was irrigated with well water
10. Direct plume shine
11. Inhalation

The intruder may also be exposed to a release of volatile radionuclides (e.g., C-14 and I-129) from the drill cuttings and contaminated well water. These pathways include direct plume shine and inhalation.

7.1.5.4 Intruder Release Pathways Dose Analysis

As discussed previously, the bounding Acute Intruder and Chronic Intruder dose scenarios are the Acute Intruder-Drilling Scenario and Chronic Intruder-Agricultural (Post-Drilling) Scenario respectively. These bounding intruder dose scenarios and associated exposure pathways are documented in the FTF PA. The water source for the intruder release pathways is a well drilled into the groundwater aquifers. The contaminated drill cuttings in the intruder release pathways are from drilling into a waste transfer line.

The point of assessment for the groundwater wells used in the intruder scenario is located one meter from the FTF boundary (Figure 7.1-1). The peak concentrations used to determine the peak doses for the intruder release exposure pathways are calculated and documented in the FTF PA. The groundwater concentrations used are peak concentrations for each radionuclide at the given point of assessment, from any of the aquifers.

The groundwater concentrations were calculated based on the FTF PA conceptual model. The conceptual model is used to simulate the performance of the FTF closure system during the 10,000-year period following FTF closure. The conceptual model is comprised of models that represent the FTF closure system and the environmental media through which radionuclides may migrate. The conceptual model was used to simulate transport of the radiological contaminants through soil and groundwater.

The conceptual model used numerous FTF-specific input parameters to represent the FTF closure system behavior over time. Many of the input parameters are based on site-specific data (e.g., soil and cementitious materials distribution coefficients) used in transport modeling. In addition, site-specific information is used to model the behavior of individual barriers within the FTF conceptual model, such as the waste tank carbon steel primary tanks and secondary tanks or annular pans (as applicable), and cementitious barriers. The models and model inputs used in the FTF conceptual model to calculate groundwater concentrations and the waste transfer line drill cutting inventory are described in detail in the FTF PA.

The peak intruder dose is calculated in the FTF PA using site-specific input parameters and the bounding dose scenario exposure pathways and peak concentrations discussed previously. Numerous bioaccumulation factors (e.g., soil-to-plant transfer factors), human health exposure parameters (e.g., water ingestion rates, vegetable consumption data) and dose conversion factors are used in the computer modeling to calculate doses for each of the exposure pathways, and these parameters are documented in the FTF PA.

In addition to the intruder peak dose analyses, additional analyses are provided in the FTF PA to characterize the context of uncertainty and sensitivity surrounding the FTF PA intruder peak dose results. These evaluations focused on the key uncertainties and sensitivities identified during calculation of the intruder dose. The uncertainty analyses provide information regarding how collective uncertainty in model input parameters is propagated through the model to the various model results. The sensitivity analyses provide information as to how various individual input parameters affect dose results. Together the uncertainty and sensitivity analyses provide assurance that the impacts of variability and uncertainty in the intruder dose analyses are understood and addressed.

The uncertainty and sensitivity analyses were primarily performed using a probabilistic model, with some additional single parameter sensitivity analyses (e.g., inventory sensitivity analysis, distribution coefficient sensitivity analysis, alternate configuration sensitivity analysis) performed through deterministic modeling. The probabilistic model allows for variability of multiple parameters simultaneously, so concurrent effects of changes in the model can be analyzed. The deterministic model single parameter analyses provide a method to evaluate the importance of the uncertainty around a single parameter of concern. The deterministic model single parameter analyses included comprehensive barrier analyses that identified barriers to waste migration and evaluated the capabilities of each barrier as understood from the results of the FTF PA. The barrier analyses assessed the contribution of individual barriers (e.g., closure cap, grout, contamination zone, waste tank liner and waste tank concrete) by comparing contaminant flux results under various barrier conditions. Using both probabilistic and deterministic models for sensitivity analyses versus a single approach provides additional information concerning which parameters are of most importance to the FTF PA model. [SRS-REG-2007-00002]

7.1.5.5 Results of the Analysis

The FTF PA modeling was used to determine an inadvertent intruder dose for comparison with the 10 CFR 61.42 performance objective. The FTF PA, Revision 1, projected the peak inadvertent intruder (i.e., individual within the FTF boundary) dose to be 73 mrem/yr chronic dose and 1.6 mrem/yr acute dose, which is less than the 500 mrem/yr performance objective during the 10,000-year performance period[54]. The 500 mrem/yr inadvertent intruder dose considers releases associated with the closure of all 22 waste tanks and related ancillary structures within FTF.

The FTF PA modeling performed for the uncertainty and sensitivity analyses was used to determine the projected dose to an inadvertent intruder for the base case, as well as other tank configurations, over a wide range of variability in input parameters. The FTF PA, Revision 1, uncertainty analysis projected a 639 mrem/yr peak of the mean chronic dose to an inadvertent intruder within the 10,000-year performance period for 1,000 base case realizations. This peak of the mean dose is greater than the 500 mrem/yr performance objective due to the tendency for many of the stochastic distributions used in the uncertainty analysis to be conservatively biased high, thereby skewing the uncertainty analysis results to the high dose side of their distributions[55].

Since there are 29 unique and independent inventory sources modeled in the FTF model, there is significant temporal and spatial complexity inherent in the modeling system. The uncertainty and sensitivity analyses (in particular the barrier analyses) demonstrated that the impact of individual parameters and/or specific barriers can be variable, with the impact depending to a great extent upon the tank type and/or radionuclide involved[56]. Additional discussion regarding the radionuclides most impacting the dose results can be found in Section 5.1 of this Draft FTF 3116 Basis Document.

Demonstration of compliance with the 500 mrem/yr peak inadvertent intruder dose is provided by the fact that peak FTF base case inadvertent intruder dose calculated in the FTF PA is less than or equal to 500 mrem/yr during the 10,000-year performance period. In addition, the uncertainty and sensitivity analyses included in the FTF PA provide sufficient information on parameter sensitivities and modeling uncertainties during the 10,000-year performance period. [SRS-REG-2007-00002]

7.1.5.6 Conclusion

The preceding discussion demonstrates that there is reasonable assurance that the 10 CFR 61.42 performance objective will not be exceeded after FTF closure.

7.1.6 10 CFR 61.43

Provisions in 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 §61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable.*

This requirement references 10 CFR 20, which contains radiological protection standards for workers and the public. The DOE requirements for occupational radiological protection are provided in 10 CFR 835

[54] The 73 mrem/yr chronic dose and 1.6 mrem/yr acute dose reflect the current results of the base (expected) case through deterministic (PORFLOW) modeling for FTF PA, Revision 1, and is estimated to occur at year 10,000 and 100 respectively, following closure of FTF. The peak FTF inadvertent intruder dose values provided in the FTF PA are not to be considered as limits. As required by DOE Manual 435.1-1, maintenance of the FTF PA will include future updates to incorporate new information, update model codes, analysis of actual residual inventories, etc., as appropriate.

[55] The 639 mrem/yr dose reflects the current results of the uncertainty analyses results projected through probablilistic modeling for FTF PA, Revision 1. The peak of the means dose provided in the FTF PA is not to be considered a limit. As required by DOE Manual 435.1-1, maintenance of the FTF PA will include future updates to incorporate new information, update model codes, analysis of actual residual inventories, etc., as appropriate.

[56] Detailed discussion of the uncertainty and sensitivity analyses can be found in Section 5.6 (pages 568 to 749) of the FTF PA Revision 1. The impact of the various integrated conceptual model segments on the dose results is detailed in Section 7.1 of the FTF PA Revision 1. [SRS-REG-2007-00002]

and those for radiological protection of the public and the environment are provided in DOE Order 5400.5. [DOE O 5400.5]

Consistent with NDAA Section 3116(a), the cross-referenced "standards for radiation protection" in 10 CFR 20 that are considered in detail in this Draft FTF 3116 Basis Document 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)[57]. [NDAA_3116] Consistent with NUREG-1854, the following sections explain that these dose limits correspond to the dose limits in 10 CFR 835 and relevant DOE Orders which establish DOE regulatory and contractual requirements for DOE facilities and activities. The following subsections show the FTF closure meets these dose limits and that doses will be maintained ALARA[58]. Table 7.1-1 provides a crosswalk between the standards set forth in 10 CFR 20 and the applicable DOE requirements.

Table 7.1-1: Crosswalk Between Applicable 10 CFR 20 Standards and DOE Requirements

10 CFR 20 Standard	DOE Requirement	Basis Document Section	Title
10 CFR 20.1101(d)	DOE Order 5400.5	7.1.6.1	*Air Emissions Limit for Individual Member of the Public*
10 CFR 20.1201(a)(1)(i)	10 CFR 835.202 (a)(1)	7.1.6.2	*Total Effective Dose Equivalent Limit for Adult Workers*
10 CFR 20.1201(a)(1)(ii)	10 CFR 835.202 (a)(2)	7.1.6.3	*Any Individual Organ or Tissue Dose Limit for Adult Workers*
10 CFR 20.1201(a)(2)(i)	10 CFR 835.202 (a)(3)	7.1.6.4	*Annual Dose Limit to the Lens of the Eye for Adult Workers*
10 CFR 20.1201(a)(2)(ii)	10 CFR 835.202 (a)(4)	7.1.6.5	*Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers*
10 CFR 20.1201(e)	DOE Order 440.1A	7.1.6.6	*Limit on Soluble Uranium Intake*
10 CFR 20.1208(a)	10 CFR 835.206 (a)	7.1.6.7	*Dose Equivalent to an Embryo/Fetus*
10 CFR 20.1301(a)(1)	DOE Order 5400.5 (II.1.a)	7.1.6.8	*Total Effective Dose Equivalent Limit for Individual Members of the Public*
10 CFR 20.1301(a)(2)	10 CFR 835.602 10 CFR 835.603	7.1.6.9	*Dose Limits for Individual Members of the Public in Unrestricted Areas*
10 CFR 20.1301(b)	10 CFR 835.208	7.1.6.10	*Dose Limits for Individual Members of the Public in Controlled Areas*

[57] The introductory "notwithstanding" phrase to NDAA Section 3116 makes it clear that the provisions of NDAA 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, NDAA 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 NDAA Section 3116(a)(1) through NDAA Section 3116(a)(3) 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 NDAA Section 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 NDAA 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 NDAA 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 FTF 3116 Basis Document 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 closure operations at FTF. Therefore, this limit is not discussed further in this Draft FTF 3116 Basis Document. 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 FTF 3116 Basis Document.

[58] 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 or training.

7.1.6.1 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.*

The DOE similarly limits effective dose equivalent from air emissions to the public at 10 mrem/yr in DOE Order 5400.5. The DOE is also subject to and complies with the EPA requirement in 40 CFR 61.92, which has the same limit[59]. The estimated dose per year from airborne emissions to the maximally exposed individual member of the public located at or beyond the SRS boundary from all operations at SRS ranged from 0.04 mrem to 0.11 mrem from 1997 through 2008. [WSRC-TR-97-00322, WSRC-TR-98-00312, WSRC-TR-99-00299, WSRC-TR-2000-00328, WSRC-TR-2001-00474, WSRC-TR-2003-00026, WSRC-TR-2004-00015, WSRC-TR-2005-00005, WSRC-TR-2006-00007, WSRC-TR-2007-00008, WSRC-STI-2008-00057, SRNS-STI-2009-00190] These values (0.04 mrem to 0.11 mrem from 1997 to 2008) for the SRS operations, not only FTF closure operations, are well below the dose limit specified in 10 CFR 20.1101(d) of 10 mrem, 0.1 mSv per year.

7.1.6.2 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).*

The DOE regulation in 10 CFR 835.202 (a)(1) has the same annual dose limit for the annual occupational dose to general employees[60]. For the occupational dose to adults during FTF closure, the total effective dose (TED) per year will be controlled using the ALARA principles, and will be below 5 rem as described in 5Q Manual, Chapter 2, *Radiological Standards*. Occupational doses to workers have been well below the annual limits specified in 10 CFR 20.1201(a)(1)(i) for all SRS work activities. Since 1995, the highest dose received by an SRS worker is 1,808 mrem/yr. [SRR-CWDA-2010-00025] The highest total dose received by an FTF worker from 1995 - 2009 was 545 mrem. [PIT-MISC-0062, SRR-CWDA-2010-00025] The TED to workers from FTF closure is expected to remain well below the DOE/NRC limit since the highest dose received by an FTF worker in 1997 was 215 mrem. Since Tank 17 and Tank 20 were operationally closed in 1997, this dose is of particular interest in addressing the dose potential for a worker during closure operations. Given that the entire FTF highest dose was 215 mrem in 1997, reasonable assurance is provided that doses received by a worker during closure activities will be below 5 rem. [PIT-MISC-0062, SRR-CWDA-2010-00025]

7.1.6.3 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 –*

---

[59] 40 CFR 61.92 provides in relevant part 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.

[60] The DOE regulation requires that the occupational dose per year for general employees shall not exceed both a TED of 5 rems which is the sum of the equivalent dose to the whole body for external exposures and the committed effective dose, which includes the weighted internal exposures to any other organ or tissue other than the skin or the lens of the eye.

(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 similar[61] to the dose limit specified in 10 CFR 835.202 (a)(2). For the occupational dose to adults during FTF closure, 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. The SRS Standard Number 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. [5Q Manual, Chapter 2, WSRC-TM-95-1]

7.1.6.4 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, and 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 the DOE regulation at 10 CFR 835.202 (a)(3). For the occupational dose to adults during FTF closure, the annual dose limit to the eye lens will be controlled using the ALARA principles, and will be below 15 rem/yr. The SRS Standard Number 01064 provides the design basis annual occupational exposure limits for the eye lens cannot exceed 3 rem/yr, which is well below the NRC limit of 15 rem/yr. [5Q Manual, Chapter 2, WSRC-TM-95-1]

7.1.6.5 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 at 10 CFR 835.202 (a)(4). For the occupational dose to adults during FTF closure, 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 ALARA principles, and will be below a shallow-dose equivalent of 50 rem/yr. [5Q Manual, Chapter 2]

7.1.6.6 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[.]

[61] 10 CFR 835.202(a)(2) also excludes exposure to skin as well as exposure to the lens of the eye and the dose term is Committed Equivalent Dose.

In addition to the adult annual dose limits during FTF closure, the soluble uranium intake by an individual is controlled to less than 10 milligrams per week. The DOE Order 440.1A soluble uranium intake requirements are the more restrictive concentrations in the American Conference of Governmental Industrial Hygienists Threshold Limit Values (0.2 milligrams per cubic meter, 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 milligrams per cubic meter). The soluble uranium OSHA PEL limit, which equates to a soluble uranium intake of 2.4 milligrams per week, is the more restrictive of the two. The soluble uranium intake, if any, during FTF closure will be controlled to 2.4 milligrams per week, which is below the NRC limit in 10 CFR 20.1201(e). [4Q1.1, Procedure 101A]

7.1.6.7 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).

The DOE regulation at 10 CFR 835.206 (a) has the same dose limit. For the embryo/fetus occupational dose during FTF closure, 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. Furthermore, after pregnancy declaration, DOE provides a mutually agreeable assignment option 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. In addition, personnel dosimetry is provided and used to carefully track exposure as controlled by the 5Q Manual, Chapter 2.

7.1.6.8 Total Effective Dose Equivalent Limit 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[.]

Provisions in DOE Order 5400.5 (II.1.a) similarly limit public doses to less than 100 mrem/yr. However, the DOE 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 FTF closure, the TED limit to an individual member of the public will be controlled to less than 0.1 rem/yr. [5Q Manual, Chapter 2] The air pathway is the predominant pathway for doses to the public from SRS operations. The air pathway doses 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). [WSRC-TR-97-00322, WSRC-TR-98-00312, WSRC-TR-99-00299, WSRC-TR-2000-00328, WSRC-TR-2001-00474, WSRC-TR-2003-00026, WSRC-TR-2004-00015, WSRC-TR-2005-00005, WSRC-TR-2006-00007, WSRC-TR-2007-00008, WSRC-STI-2008-00057, SRNS-STI-2009-00190]

7.1.6.9 Dose Limits for Individual Members of the Public in Unrestricted Areas (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.

The DOE regulation at 10 CFR 835.602 establishes the expectation that TED in controlled areas will be less than 0.1 rem in a year. For individual members of the public during FTF closure, 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. The 5Q Manual, Chapter 2, also restricts the TED 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, radioactive materials areas have been established for radioactive material accumulation possibly resulting 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 0.00005 mrem/hour (hr) (100 mrem/2,000 hrs = 0.05 mrem/hr or 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. [5Q Manual, Chapter 2] Therefore, SRS implementation of the provisions at 10 CFR 835.602 and 10 CFR 835.603 provides limits protective of the dose limit specified in 10 CFR 20.1301(a)(2). 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[62]. [5Q Manual, Chapter 5]

7.1.6.10 Dose Limits for Individual Members of the Public in Controlled Areas (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.

The DOE regulation at 10 CFR 835.208 has the same dose limit. The TED limit to an individual member of the public granted access to controlled areas during FTF closure will be controlled to 0.1 rem/yr. 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[63]. [5Q Manual, Chapter 5]

7.1.6.11 As Low As Reasonably Achievable (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…[.]

The DOE has a similar requirement, and the DOE 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 FTF closure, 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, the DOE regulation at 10 CFR 835.101(c) requires the contents of each Radiation Protection Program

[62] 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.

[63] 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 Section 7.1.6.8.

(RPP) to include formal plans and measure for applying the ALARA process to occupational exposure as further discussed in Section 7.1.6.12.1 of this Draft FTF 3116 Basis Document.

7.1.6.12 Reasonable Assurance

Measures that provide reasonable assurance that FTF closure will comply with the applicable dose limits and with the ALARA provisions include the documented RPP, the Documented Safety Analysis (DSA), design, regulatory and contractual enforcement mechanisms and access controls, training and dosimetry. These measures are discussed in the following subsections.

7.1.6.12.1 SRS Radiation Protection Program

The DOE regulates occupational radiation exposure at its facilities through 10 CFR 835, which 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 10 CFR 835, if violated, provide a basis for the assessment of civil penalties under the Atomic Energy Act of 1954, Section 234A, as amended. [42 USC 2282a]

Pursuant to 10 CFR 835, activities at SRS, including FTF closure operations, must be conducted in compliance with the documented RPP for SRS as approved by DOE. The key RPP elements include monitoring of individuals and work areas, access control to areas containing radiation and radioactive materials, use of warning signs and labels, methods to control the spread of radioactive contamination, radiation safety training qualification, objectives for the design of facilities, criteria for radiation and radioactive material workplace levels, and continually updated records to document compliance with the provisions of 10 CFR 835. The RPP also includes formal plans and measures for applying the ALARA process.

The 10 CFR 835 requirements, as contained in the RPP, are incorporated in the Standards/Requirement Identification Document system (S/RIDs). The S/RID system links the requirements of 10 CFR 835 to the site-level and lower-level implementing policies and procedures that control radiological work activities conducted across the site. 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 10 CFR 835.

7.1.6.12.2 Documented Safety Analysis

The FTF operates under a DSA in accordance with 10 CFR Part 830. As the first step in the development of the DSA, a formal Hazard Analysis (HA) was performed to systematically present the results of potential process-related hazards, Natural Phenomena Hazards and external hazards that can affect the public, workers and environment through the occurrence of single or multiple failures. [DOE-STD-3009-94] The HA was performed by subject matter experts including operations, engineering, industrial hygiene, radiological protection, environmental compliance and maintenance professionals.

The HA consisted of three phases:

1. hazard identification
2. hazard classification
3. hazard evaluation [DOE-STD-3009-94]

The hazard identification phase identifies possible radiological and chemical hazardous materials associated with normal and abnormal operations as well as potential energy sources to disperse hazardous materials into the environment.

The hazard classification phase evaluates for the maximum possible quantities of hazardous materials, which are then evaluated against DOE criterion to determine the overall hazard classification. [DOE-STD-1027-92]

The hazard evaluation phase identifies possible normal and abnormal operational events that could expose the public and workers to hazardous material and, therefore, are evaluated to establish the magnitude of the risk. Additionally, the consequence and frequency of each operational event must be determined and risk level identified. The purpose of identifying the risk level is to determine which operational events pose risk (and thus require additional evaluation) and those events which present negligible risk to the public and workers.

As waste is removed from the waste tanks during the closure process, the DSA requires controls on the waste tanks commensurate with the risk of the material remaining in the waste tank. These controls include engineering controls (e.g., physical isolation requirements on transfer lines and motive forces) and administrative controls (e.g., limits on waste transfers and equipment operation).

The DSA identifies hazards in the HA that could impact the public, facility workers and the environment during normal operations and accident conditions. The DSA also discusses summary descriptions of key SRS safety management programs.

In part, these administrative controls require: a facility manager be assigned who is accountable for safe operation and in command of activities necessary to maintain safe operation, personnel who carry out radiological controls functions have sufficient organizational freedom to ensure independence from operating pressure, that personnel receive initial and continuing training including radiological control training and an RPP shall be prepared consistent with 10 CFR 835. In addition, the design requirements implement 10 CFR 835 and, in particular, implement ALARA principles.

7.1.6.12.3 Radiological Design for Protection of Occupational Workers and the Public

The FTF radiological facilities and facility modifications are designed to meet the requirements of 10 CFR 835 Subpart K. The SRS Standard Number 01064 provides the requirements necessary to ensure compliance with 10 CFR 835. [WSRC-TM-95-1] The standard refers to 10 CFR 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 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.

The facility design also incorporates radiation zoning criteria to ensure exposure limits 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. [WSRC-TM-95-1] 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.

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

7.1.6.12.4 Regulatory and Contractual Enforcement

Any violation of the 10 CFR 835 requirements is subject to civil penalties pursuant to the Atomic Energy Act of 1954, Section 234A, as amended, 42 USC 2011 et seq., as implemented by DOE regulations in 10 CFR Part 820. In addition, the requirements in 10 CFR 835 and all applicable DOE Orders are incorporated into all contracts with DOE contractors. The DOE enforces these contractual requirements through contract enforcement measures, including the reduction of contract fees. [48 CFR 970]

7.1.6.12.5 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. [5Q Manual, Chapter 5, 5Q Manual, Chapter 6]

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. [5Q Manual, Chapter 5]

7.1.6.12.6 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. The highest annual dose received by an SRS worker from 1995-2009 was 1,808 mrem TED and the highest total dose received by an FTF worker from 1995 – 2009 was 545 mrem compared to the DOE Administrative Control Limit of 2,000 mrem/yr and the 10 CFR 835 limit of 5,000 mrem/yr. [PIT-MISC-0062, SRR-CWDA-2010-00025]

In addition, for all work activities, the average TED exposure for workers receiving a TED dose at SRS has been 134 mrem/yr over the last five years, 2004-2008. [10-ORAU-0098]

7.1.7 10 CFR 61.44

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.*

This section outlines the relevant factors of FTF siting, design, use, operation and closure, which ensure compliance with 10 CFR 61.44 for the purpose of the Draft FTF 3116 Basis Document.

7.1.7.1 Siting

A site characteristics review of demography, geography, meteorology, climatology, ecology, geology, seismology, and hydrogeology is presented in Section 3.0 of the FTF PA [SRS-REG-2007-00002] and Section 2.0 of this Draft FTF 3116 Basis Document, and is briefly summarized below. The SRS is located in south-central South Carolina, approximately 100 miles from the Atlantic Coast. The major physical feature at SRS is the Savannah River, approximately 20 miles along the southwestern boundary of the site. The FTF is an active waste storage facility located within the GSA of SRS, approximately 5 miles from the boundary of SRS. The nearest towns are New Ellenton, South Carolina (5 miles), Jackson, South Carolina (5 miles), Snelling, South Carolina (5 miles) and the more populated areas include Aiken, South Carolina (12 miles) and Augusta, Georgia (15 miles). As of 2008, the region of influence population was over 550,000.

The general climate for the SRS region is a humid subtropical climate characterized by short mild winters and long warm and humid summers. An average of 49 inches of rainfall each year and the average monthly temperature has a low of 46 degrees in the winter to 81 degrees in the summer. SRS supports an abundant terrestrial and semi-aquatic wildlife. The areas around FTF have grasses, forests, and swamps. An abundance of terrestrial, avian, wetlands, and aquatic wildlife live within SRS. FTF itself is a heavy industrial complex surrounded by fencing and covered in asphalt and, therefore, few animals are seen near the tanks; however burrowing animals in the surrounding areas are common. At closure, a cover is expected to be designed incorporating the latest barrier technologies to limit burrowing animals and growth of plants with taproot system; however, those are expected to infiltrate, in time, after a loss of institutional controls.

The principal geology of the region is characterized by unconsolidated soils. The vadose zone is comprised of cross-bedded, poorly sorted sands with clay lenses indicating fluvial deposition with occasional transitional marine influence. It is represented by wide differences in lithology and presents a very complex system of transmissive and confining beds.

The GSA is bounded by two surface waters: Upper Three Runs and Fourmile Branch. These waters eventually feed into the Savannah River. The aquifers of primary interest for FTF are the Upper Three Run and Gordon Aquifers. Other aquifers do not contribute to the potential dose to the workers, public or intruder and were not included in the models.

Because SRS is not located within a region of active plate tectonics characterized by volcanism, volcanology is not an issue of concern for SRS. The seismic history of the southeastern U.S. is dominated by the Charleston earthquake of 1886 with an estimated magnitude of 7.0. The most recent seismic event within a 50-mile radius of SRS was in March 2009 with a magnitude of 2.6. In the past approximately 30 years there have been four earthquakes with epicenter locations within SRS boundaries, all with a magnitude less than 3.0. [SRS-REG-2007-00002]

7.1.7.2 Design

The closure design of the FTF waste tanks and ancillary structures provide long-term stability, which is consistent with the performance objective.

There are multiple elements of the FTF design that will serve to minimize infiltration of water through the waste tanks and ancillary structures. The waste tank concrete vaults, steel tanks and secondary tanks or annular pans, where applicable, serve to significantly retard water flow through the waste tanks. The concrete structures, steel wall liners, if applicable, and transfer line encasements or outer jackets will serve to significantly retard water flow into ancillary structures. The FTF design features are described in detail in the FTF PA. In addition, the waste tanks and ancillary structures are expected to be covered with a closure cap[64], which further limits the water infiltration into the waste tanks and ancillary structures.

Because the waste tanks will be filled with grout at closure, significant structural failure (i.e., collapse) is not likely. For tank types with an annulus, the annulus will also be grouted for stability and to minimize void spaces. The impact of potential waste tank degradation (e.g., cracking or corrosion leading to increased water infiltration) is considered in the FTF PA analysis. Ancillary structures such as diversion boxes, pump pits and pump tanks will be filled, as necessary, to prevent subsidence.

Multiple waste tank design elements will serve as inadvertent intruder barriers. The FTF closure cap, concrete tops on Type I, III and IIIA waste tanks, grout-filled domed roof of the Type IV waste tanks[65] and waste tank reducing grout are considered sufficient barriers to prevent drilling into the waste tanks, given regional well drilling practices and the presence of nearby land without underground rock or concrete obstructions. [SRS-REG-2007-00002]

7.1.7.3 Use/Operation

The use/operation of FTF waste tanks and ancillary structures will support long-term stability consistent with the performance objective. During operations, corrosion control and structural integrity programs are implemented to maintain design features utilized for waste containment (e.g., waste tanks and ancillary structures). These programs ensure that tanks are monitored for structural integrity via mechanisms such as a tank inspection program and a tank leak detection system. Programs such as these will be

[64] The closure cap design described in the FTF PA is based on the best information available at the time the FTF PA was developed. [SRS-REG-2007-00002] The design information utilized is for planning purposes sufficient to support evaluation of the closure cap as part of the integrated site conceptual model evaluated in the FTF PA. Any actual closure cap design will be finalized closer to the time of FTF closure in accordance to the FFA for SRS (e.g., Section IX.E.(2).) [WSRC-OS-94-42], to take advantage of possible advances in materials and closure cap technology that could be used to improve the design. The final closure cap design will minimize water infiltration into the waste tanks and ancillary structures, and the likelihood of intrusion into the waste.

[65] Grout used to fill the domed roof of the Type IV waste tanks will have a minimum 2,000 psi nominal compressive strength at 28 days.

maintained throughout FTF use and operation. The FTF waste tanks and ancillary structures monitoring continues after closure via the site Groundwater Protection Program. [SRNS-TR-2009-00076]

7.1.7.4 Closure

Final FTF closure will support long-term stability consistent with this performance objective. In this context, long-term stability of the closed FTF site means that the stabilized residuals in the waste tanks and ancillary structures, the waste tanks, and the ancillary structures (including integral equipment) maintains structural integrity under the closure conditions for hundreds to thousands of years following closure. A stable closure system prevents subsidence of, and minimizes water intrusion into, the closed site and mitigates migration of residual material into the environment. In addition, a carefully designed closure site minimizes the likelihood of inadvertent intrusion into the system and disturbance of the stabilized residuals.

The waste tank systems (i.e., primary tank, secondary tank or annular pan, and concrete roofs and vaults, as applicable) and the ancillary structures themselves will provide the primary stability for the closed FTF site. Grouting of the waste tanks and backfilling of ancillary structures with an appropriate fill material, as necessary, will prevent subsidence and will minimize the migration of radioactive material into the environment over time, and will support long-term stability of both the tank structures and the waste form[66]. Grout used to fill the domed roof of the Type IV waste tanks will have a minimum 2,000 psi nominal compressive strength at 28 days to deter intrusion. Type I and Type III/IIIA tanks have sufficient thicknesses of reinforced concrete roofs to deter such intrusion. Grouting of the waste tanks, filling of the ancillary structures, and grout chemical and mechanical characteristics are discussed in detail in the FTF PA. [SRS-REG-2007-00002]

A closure cap is expected to be designed and constructed over the the FTF site following grouting of the FTF waste tanks and backfilling of the ancillary structures, as necessary[67]. The closure cap design described in the FTF PA is based on the best information available at the time the FTF PA was developed. [SRS-REG-2007-00002] The design information utilized is for planning purposes sufficient to support evaluation of the closure cap as part of the integrated site conceptual model evaluated in the FTF PA. The actual closure cap design will be finalized closer to the time of FTF closure, to take advantage of possible advances in materials and closure cap technology that could be used to improve the design. The final closure cap design will minimize water infiltration into the waste tanks and ancillary structures, and the likelihood of intrusion into the waste. [SRS-REG-2007-00002]

The DOE will maintain GSA ownership, which includes the FTF. The SRS Land Use Plan requires Federal ownership and control of the site well beyond 100 years after tank closure. [PIT-MISC-0041]

7.1.7.5 Conclusion

As previously discussed, the site conditions do not present hazards that impact FTF stability. In addition, the FTF closure methods will result in a facility closure that does not require ongoing maintenance. Therefore, closure of the FTF complies with 10 CFR 61.44 performance objective.

[66] It may be shown through calculation and other means that backfilling certain ancillary structures will not be required for the purpose of long-term stability. For example, there are no current plans to grout or fill the FTF transfer lines.
[67] Final closure of FTF will performed per the requirements of the FFA for SRS (e.g., Section IX.E.(2). [WSRC-OS-94-42]

8.0 STATE-APPROVED CLOSURE PLAN

Section Purpose

The purpose of this section is to demonstrate removal from service and stabilizing of the FTF waste tanks and ancillary structures, as appropriate, will be performed pursuant to a State-approved closure plan.

Section Contents

This section discusses the State of South Carolina regulation of the waste tanks and ancillary structures and shows that removal from service of the FTF waste tanks and ancillary structures will be pursuant to State-approved Closure Modules, consistent with the FTF General Closure Plan (GCP).

Key Points

- The FTF waste storage and removal are governed, in part, by a SCDHEC industrial wastewater construction permit.
- The overall plan for removing from service and stabilizing the FTF waste tanks and ancillary structures, referred to as the FTF General Closure Plan, requires approval by SCDHEC.
- A specific Closure Module for each waste tank and ancillary structure, consistent with the requirements of the FTF GCP, will be developed and submitted to the SCDHEC for approval. The State must grant this approval before final stabilization activities may proceed.

The NDAA Section 3116(a) 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)(ii) [Will be disposed of] 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.*

8.1 State-Approved Closure Plan

The FTF waste storage and removal operations are governed by an SCDHEC industrial wastewater construction permit issued March 3, 1993. [DHEC_03-03-1993] 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) and all regulations implementing the Act. The State of South Carolina has authority for approval of wastewater treatment facility operational closure under Chapter 61, Articles 67 and 82 of the SCDHEC Regulations. [SCDHEC R.61-67, SCDHEC R.61-82]

The FTF GCP addresses the State's regulatory authority relevant to removing the FTF waste tanks and ancillary structures from service. The GCP sets forth the general protocol by which DOE intends to remove from service the FTF waste tanks and ancillary structures to protect human health and the environment. The FTF GCP requires approval by SCDHEC. Prior to approval by SCDHEC, the FTF GCP will be made available to the public for review and comment. [LWO-RIP-2009-00009]

Before final stabilization activities commence[68], individual waste tank and ancillary structure closure plans, referred to as Closure Modules, describing closure details will be developed and submitted to SCDHEC for approval[69]. Prior to approval, the Closure Modules will be made available to the public for review and comment as deemed appropriate by SCDHEC. The Closure Modules will describe the waste tank(s) or ancillary structure(s) being covered, waste removal activities performed and effectiveness, justification that additional waste removal is not technically practical from an engineering perspective[70], characteristics of remaining residuals and the stabilization process. The Closure Modules will provide analysis for each waste tank or ancillary structure demonstrating conformance with the performance objectives set forth in the GCP.

8.2 Conclusion

As explained above, the FTF waste tanks and ancillary structures will be removed from service, stabilized and operationally closed pursuant to State-approved Closure Modules, consistent with the FTF General Closure Plan. Thus, the FTF waste tanks, ancillary structures and the stabilized residuals "[will be disposed of] 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."

[68] Final stabilization activities in this context refers to the addition of grout to the waste tanks, annulus, and cooling coils, or, in the case of ancillary structures, grout or other appropriate fill material, as necessary, for the purpose of stabilizing the structure.

[69] Each individual waste tank and ancillary structure is required to be covered by a Closure Module approved by SCDHEC. Closure Modules may be written for individual waste tanks and ancillary structures or for groupings of waste tanks and ancillary structures.

[70] Per the FFA, the waste tanks will be cleaned until Department of Energy-Savannah River Operations Office, SCDHEC and EPA agree that waste removal may cease. The Closure Module provides the basis for agency agreement.

9.0 CONCLUSION

As demonstrated in the preceding sections of this Draft FTF 3116 Basis Document, the stabilized residuals within the waste tanks and ancillary structures, the waste tanks, and the ancillary structures (including integral equipment) located at FTF at the time of closure are not high-level waste pursuant to the criteria set forth in NDAA Section 3116(a). This Draft FTF 3116 Basis Document will be finalized after DOE has completed consultation with NRC and, although not required by NDAA Section 3116, after public review and comment.

10.0 REFERENCES

Note 1: Reference numbers with a **REDACTED** designator were classified: ***"Official Use Only, Exemption 2 — Not Releasable to the Public or Foreign Nationals without prior approval from DOE-SR."*** However, the restricted information does not pertain to this document, thus a redacted copy with the ***Restricted Information Removed*** has been provided.

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11.0 GLOSSARY

A	
Actinide	Group of elements of atomic number 89 through 103 including thorium, uranium, neptunium, plutonium, americium and curium.
Ancillary Structures	Structures associated with the waste storage tanks, such equipment as transfer line piping, pump tanks or evaporators, which are used to distribute or control the transfer of waste from one storage point to another storage point, or are used to volume reduce the waste.
Air Content	Amount of air incorporated into the grout as the result of mixing and placement.
Annulus	Also referred to as the secondary containment of a waste tank, surrounds the primary tank of Types I, II, III and IIIA waste tanks, providing a location for collection of any leakage from the primary tank.
B	
Base Case	The waste tank system configuration modeling case within the FTF PA that represents the best estimate of expected conditions for the FTF closure system based on currently available information.
Basemat	Concrete pad upon which the waste tank is constructed. The pad has close tolerances for tank leveling and the concrete is quality controlled to ensure the structural integrity to tank foundation. The basemat is also referred to as floor slab or foundation.
Bleed Water	Water that separates from the grout as the result of solids settling.
C	
Chromated Cooling Water	Coolant comprised of choromate-inhibited water that circulates through the cooling coils of waste tanks to remove radioactive decay heat and other sources of heat (e.g., steam heat loads, ventilation heat loads or mechanical heat loads from pumping/mixing operations).
Closure Plan	Plan that presents the environmental regulatory standards and guidelines pertinent to the closure of the tanks and describes that process for evaluating and selecting the closure configuration (i.e., residual inventory and form).
Compressive Strength	Force per unit area required to break an unconfined grout or concrete sample.
Concentration	Amount (e.g., in grams or moles) per volume of a substance.
Conductivity Probes	A simple electrical device that works on the principle that liquids conduct electricity more readily than air. If a liquid comes in contact with the probe it will complete an electrical circuit and send a signal for indication or alarm purposes.
Cooling coils	Coils installed in the tanks to remove the decay heat that is generated by the waste in the tanks. Arrangements and designs of cooling coils differ, depending on the type of tank. Type I and II tanks, in addition to having vertical cooling coils, also have cooling coils across the bottom of the tank to provide a means for cooling the bottom of the tank.
Contamination	In the context of this Draft FTF 3116 Basis Document, contamination refers to radioactive materials that have been projected to migrate from the closed FTF.
Core pipe	Internal pipe of transfer line that comes into contact with the waste materials. The core pipe is usually located within a jacket pipe.

Curie	A unit of radioactivity - the quantity of nuclear material that has 3.7E+10 disintegrations per second.
D	
Diffusion	Movement of contaminants from an area of higher concentration to an area of lower concentration.
Diversion Box	A shielded reinforced concrete structure containing transfer line nozzles to which jumpers are connected in order to direct waste transfers to the desired location.
E	
E_h	The symbol for oxidation/reduction reaction (redox) potential in millivolts.
Evaporator	Steam-heated, water-cooled system installed in the tank farms to concentrate underground waste storage tank contents, in order to reduce the liquid waste volume.
Exposure	Being exposed to ionizing radiation or to radioactive material.
F	
Federal Facility Agreement	Agreement between EPA, DOE and SCDHEC that directs the comprehensive remediation of the Savannah River Site. It contains requirements for (1) site investigation and remediation of releases and potential releases of hazardous substances and (2) interim status corrective action for releases of hazardous wastes or hazardous constituents.
Fly Ash	A mineral admixture used in grout to enhance finishing characteristics, make the mix more economical and to improve pumping. It is finer in consistency than cement and its particles are round. These fine particles make the mix finish easier and pump easier.
G	
General Separations Area	Centralized area of SRS including, E, F, H, J, S and Z Areas that are the heavily industrialized areas of SRS.
Grout	A cement mixture, sufficiently fluid, which can be pumped into equipment cavities creating a watertight bond and increasing the strength of the existing structural foundation. Capable of slowing the vertical movement or migration of water.
H	
H-Modified Process	The modified PUREX process in H Canyon for separation of special nuclear materials and enriched uranium from irradiated targets.
Hydraulic Conductivity	Velocity of water flow through saturated materials (e.g., concrete, grout, soil).
I	
Institutional Control	A 100-year period in which DOE retains ownership and control of FTF such that FTF facility maintenance and controls will be performed to prevent inadvertent intrusion and protect public health and the environment.

L	
Leak Detection Boxes	Structures that provide for the collection and detection of leakage from the transfer lines.
Line Encasement (Sealed Concrete Trench)	Enclosed core pipes in a covered reinforced concrete encasement below ground. Any core pipe leakage into the encasement and in-leakage of groundwater into the encasement will gravity drain to a catch tank.
N	
NDAA Section 3116	The Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 Section 3116 was passed by Congress on October 9, 2004 and signed by the President on October 28, 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.
O	
Oxalic Acid	A relatively strong organic acid, being about 10,000 times stronger than acetic acid.
Oxidation Potential (E_h)	The measure of a material to oxidize or lose electrons.
P	
Permeability	Capability of a material to let pass other molecules or particles.
pH	A measure of the acidity or alkalinity of a solution, numerically equal to 7 for neutral solutions, increasing with increasing alkalinity and decreasing with increasing acidity.
Pitting	Localized corrosion of a metal surface, confined to a point or small area that takes the form of cavities.
Porosity	Grout porosity is generally defined as the percentage of total volume of cured grout that is not occupied by the starting cementitious materials and the products that result from reaction of these cementitious materials with water.
Primary Tank	The primary tank, sometimes referred to as the "shell," is the component of the waste tank that actually contains the liquid waste. The primary tank is contained within the secondary containment, if any.
Progeny	Decay products or descendants of specific radionuclides.
Pump Pit	Shielded reinforced concrete structures located below grade at the low points of transfer lines, contain pump tanks and are usually lined with stainless steel.
Pump Tank	All FTF pump pits house a pump tank with the pump pits providing secondary containment for pump tanks. The pump tanks have a nominal capacity of 8,000 gallons each. The pump tanks installed in FTF are all of the same basic size (8.5 feet tall, 12 feet in diameter).
PUREX Process	The Plutonium Uranium Extraction process used in the F-Canyon reprocessing facility to extract uranium and plutonium from aluminum-clad, depleted uranium targets which had been irradiated in the site's nuclear production reactors.

R	
Residual Radioactivity	Radioactivity in structures, materials, soils, groundwater and other media at a site remaining after closure.
Riser	The risers through the tank tops provide for access to the tank and annulus interiors. Risers are used primarily to provide for the installation of equipment such as pumps and cooling equipment; instrumentation such as level probes and leak detection; ventilation; and to provide access to the tank interior for sampling, depth measurement and inspection.
S	
Saltcake	Saltcake located in waste tanks consists of crystallized salts with interstitial void space and entrained soluble solids (assumed to be partially sludge solids).
Saltstone Production	A process in which low-activity salt solution is mixed with dry chemicals (cement, slag and fly ash) to form a homogeneous grout mixture.
Secondary Containment	Also referred to as an annulus, of a waste tank. The secondary containment surrounds the primary tank of Types I, II, III and IIIA waste tanks, providing a location for collection of any leakage from the primary tank.
Segregation	Separation of sand from binder as the result of impact and separation of water from grout as the result of gravity settling of the solids from the grout slurry.
Set time	Time after mixing at which the grout responds as a solid.
Shotcrete	Concrete conveyed through a hose and pneumatically projected at high velocity onto a surface. Shotcrete undergoes placement and compaction at the same time due to the force with which it is projected from the nozzle. Shotcrete was used in the construction of Type IV tanks.
Silica fume	Also known as microsilica, is a byproduct of the reduction of high-purity quartz with coke in electric arc furnaces in the production of silicon and ferrosilicon alloys. Silica fume is used as an addition in Portland cement concretes to improve properties. It has been found that silica fume improves compressive strength, bond strength and abrasion resistance. Addition of silica fume also reduces the permeability of concrete to chloride ions, which protects concrete's reinforcing steel from corrosion.
Slag	Slag was introduced into the design mixes which in addition to its hydraulic activity, also provides chemical reducing power to the mix. Slag has been shown to possess chemically reducing properties that are favorable for technetium reduction and for plutonium and selenium.
Solubility	Mixture of at least two liquid components or of at least one solid and a liquid component.
Source Term	The amount and type of radioactive material released into the environment.
Stabilized Contaminant	Grouted waste remaining in the waste tanks or ancillary equipment after system closure.
Stochastic	A probabilistic distribution of parameters.
Supernate	Liquid salt solution found above the sludge layer after settling of solids in waste tanks has occurred as a result of a liquid waste transfer to one of the waste processing facilities or receipt tanks. Also referred to, generally, as supernatant.

U	
Underliner Sump	An engineered feature that collects any leakage through the concrete or stainless steel liners beneath waste tanks.
Unit Weight	Weight of a unit volume, typically one cubic foot.
V	
Valve Boxes	Transfer valve boxes facilitate specific waste transfers that are conducted frequently. The valves are generally manual ball valves in removable jumpers with flush water connections on the transfer piping. The valve boxes provide containment of and access to the valves.
Vault	Term used to describe the underground concrete floor, walls and roof that enclose the steel primary liner of the waste tank.
Viscosity	Rheological quality of fluids describing the resistance to flow.
W	
Waste Characterization System	Computer based system designed to integrate historical information, current sample data, and physical properties of constituents to develop predictions of concentrations and inventory.
Working Slab	Concrete surface usually placed to create a level construction surface. This concrete is normally lower quality without reinforcement and is either broken up after or cracked during construction activities between the tanks, thus is not considered a barrier to vertical water migration.

APPENDIX A: LIQUID WASTE SYSTEM DESCRIPTION

Appendix Purpose

The purpose of this appendix is to provide a brief overall description of the SRS Liquid Waste System.

Appendix Contents

This Appendix briefly describes the history of the SRS underground radioactive waste storage tanks and their contents, and describes the methods used to treat and dispose of this waste.

Key Points

- SRS has 51 waste tanks within two tanks farms, FTF and HTF, which entered service between 1954 and 1986.
- There are four types of waste tanks, designated Types I, II, III/IIIA and IV.
- The 27 Type III/IIIA tanks meet current EPA requirements for full secondary containment and leak detection; the other 24 do not meet these requirements.
- As of March 2010, approximately 37,400,000 gallons of radioactive waste were stored in the waste tanks, most of this from separation of special nuclear materials and enriched uranium in the two SRS nuclear materials processing facilities known as F Canyon and H Canyon.
- The high-level waste fraction removed from the waste tanks (including the sludge waste) 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.
- Salt waste removed from the waste tanks is pretreated with the resultant low-volume, high-activity fraction being sent to DWPF and the high-volume, low-activity fraction being treated and disposed of as a non-hazardous, cementitious waste form (i.e., saltstone) in the Saltstone Disposal Facility (SDF).

A.1 Background

The SRS is an approximately 310 square mile site located in the state of South Carolina and bordering the Savannah River. Since it became fully operational in 1954, it has produced nuclear material for national defense, research, medical, and space programs. The separation of nuclear material from irradiated targets and fuels resulted in the generation of large quantities of radioactive liquid waste, which is currently stored onsite in large underground radioactive waste storage tanks.

Most of the waste tank inventory currently stored at SRS is a complex mixture of chemical and radioactive waste generated during the separation of special nuclear materials and enriched uranium from irradiated targets and spent fuel using the PUREX process in F Canyon and the HM process in H Canyon. 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 in this liquid radioactive waste is due, in part, to the fact that waste streams from the first cycle (high-heat) and second cycle (low-heat) extractions from each canyon were typically stored in separate waste tanks to better manage waste heat generation.

When these acidic streams were pH-adjusted with caustic to form a high alkaline solution and transferred to a waste tank in one of the two tank farms, the resultant precipitate settled into four characteristic sludge consistencies. Typically, this sludge waste can still be found in the waste tanks where it was originally deposited beginning in 1954. Historically, new waste receipts into the tank farms 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 waste 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 sludge than the corresponding LAW sludge. 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 sludge components.

The soluble portions of the first and second 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. Beginning in 1964, large evaporator systems were used to volume reduce the liquid salt waste to form two distinct waste types: concentrated supernate and saltcake. 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 FTF and HTF. More recently, with transfers of sludge slurries to sludge washing tanks, removal of saltcake to support salt waste pretreatment, 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 through the life of the program. [SRR-LWP-2009-00001]

A.2 Liquid Waste System

The Liquid Waste System is the integrated series of facilities at SRS that safely manage the storage of existing waste inventory in the 49 active waste tanks, support the transfer and waste reduction of this waste, perform the pretreatment of waste in preparation for eventual waste disposal, and provides for the permanent disposal of the high-volume, low-activity decontaminated salt solution, as well as other smaller low-level waste liquid streams, in the SDF. Key facets of the Liquid Waste System are briefly described in the text that follows.

A.2.1 Waste Tank Storage

The SRS has a total of 51 waste tanks located within two tank farms, FTF (22 waste tanks) and HTF (29 waste tanks). These waste tanks were placed into operation between 1954 and 1986. There are four distinct waste tank designs, Types I through IV. Type III/IIIA waste tanks are the newest waste tanks and were placed into operation between 1969 and 1986. There are a total of 27 Type III/IIIA waste tanks. Figure A.2-1 shows Type III waste tanks during construction. These waste tanks meet current EPA requirements for full secondary containment and leak detection. The remaining 24 waste tanks do not meet EPA requirements for secondary containment. The twelve Type I waste tanks are the oldest waste tanks and were constructed in the early 1950s. The four Type II waste tanks were constructed between 1955 and 1956. There are eight Type IV waste tanks that were constructed between 1958 and 1962. Two of these Type IV waste tanks, Tanks 17 and 20, were removed from service and filled with grout in 1997 under SCDHEC approved General Closure Plan. [PIT-MISC-0002, PIT-MISC-0004]

Figure A.2-1: Waste Tanks Under Construction

Thirteen SRS waste tanks have a history of leakage. These waste tanks include Type I, Type II and Type IV designs. No Type III or IIIA waste tanks have developed leak sites. Five of the 13 waste tanks with a history of leakage are located in FTF (Tank 1, Tank 5, Tank 6, Tank 19 and Tank 20). [SRR-STI-2010-00283] Sufficient waste has been removed from these waste tanks such that there are currently no active leak sites. [SRR-LWP-2009-00001] Only once has waste leaking from a primary waste tank reached the environment. This event occurred in September 1960 and was associated with Tank 16 in HTF. [DPSOX-5954] In all other cases, leakage from a primary waste tank has been confined within the secondary containment annuli and has not reached the surrounding soils.

Although outside the scope of this determination, waste did reach the soil from a spill associated with Tank 8. In 1961, Tank 8 was inadvertently over-filled and waste escaped along the junction area where a transfer line entered Tank 8. The majority of the liquid was contained within the secondary encasement associated with the transfer lines for Tanks 1 through 8 in FTF, but an estimated 1,500 gallons spilled into the surrounding soil. [DPSPU 75-11-8]

A.2.2 Waste Tank Space Management

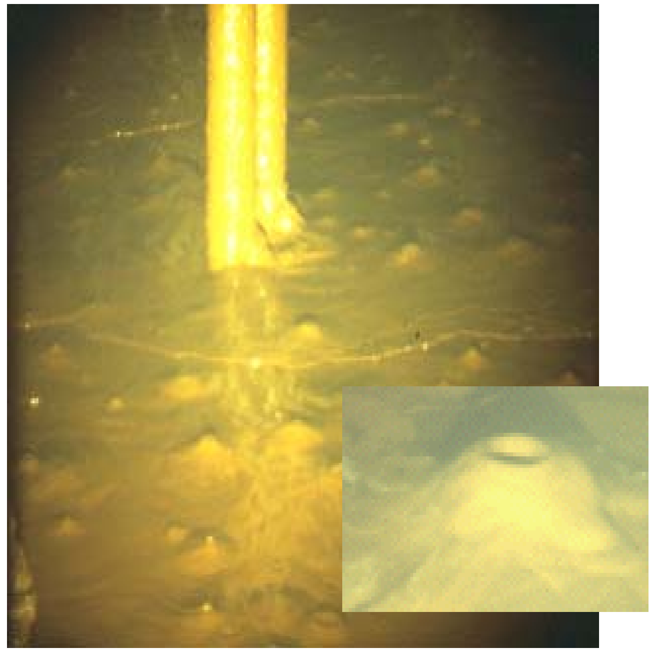
To make better use of available waste tank storage capacity, incoming liquid waste is evaporated to reduce its volume. Since 1951, the tank farms have received over 140,000,000 gallons of liquid waste, of which over 100,000,000 gallons have been evaporated. As of March 31, 2010, approximately 37,400,000 gallons of liquid radioactive waste was stored in 49 active waste tanks. Projected available waste 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/IIIA waste tank space, the waste tanks that have full secondary containment systems, has been filled that normal operations, waste removal, and waste processing operations cannot continue. A portion of the Type III/IIIA waste tank space must be reserved as contingency space should a new waste tank leak be realized. [SRR-LWP-2010-00040]

Waste receipts and transfers are normal tank farm activities as the tank farms receive new waste from the H-Canyon legacy material stabilization program, 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 sufficient salt waste has been treated and disposed of in SDF. Three evaporator systems are currently operating at SRS, the 2H, 3H, and 2F systems. Evaporator operations are currently impacted due to limited salt waste storage space. [SRR-LWP-2009-00001]

Figure A.2-2: Supernate (Top) and Saltcake (Bottom)

As of March 31, 2010, approximately 358,000,000 curies of radioactivity were stored in the SRS tank farms. 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 either crystallizes as salts or remains as a concentrated supernate solution (Figure A.2-2). The resulting crystalline solids are commonly referred to as saltcake. [SRR-LWP-2010-00040]

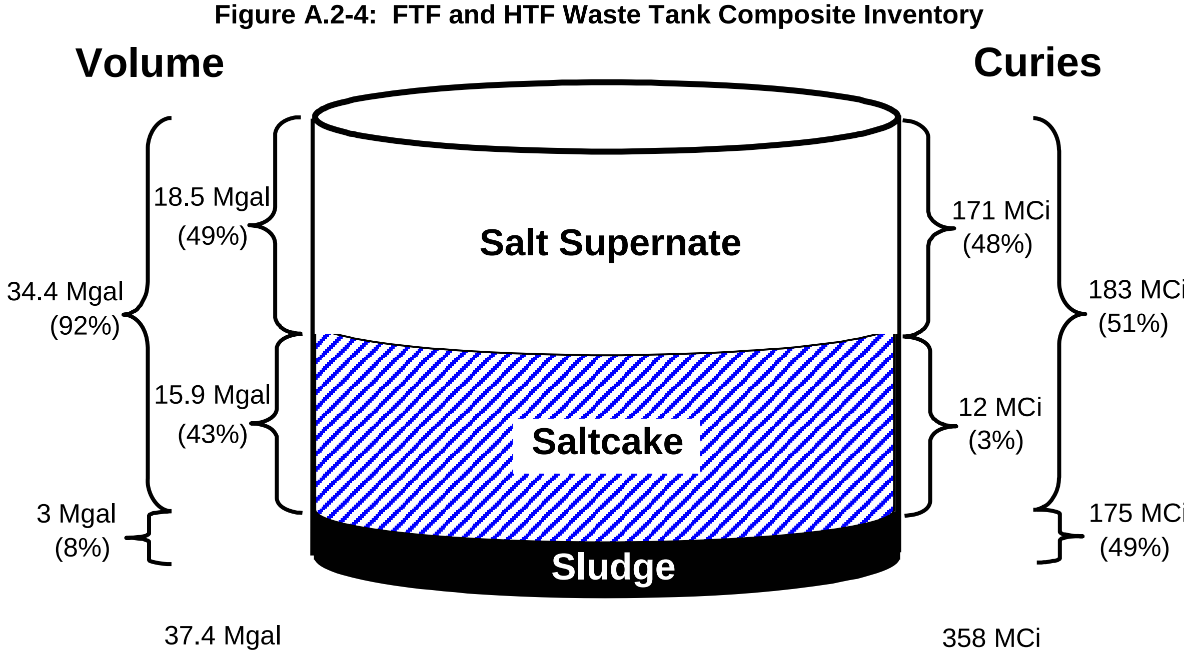
Figure A.2-3: Sludge Component of Radioactive Waste



The sludge component (Figure A.2-3) of the radioactive waste represents approximately 3,000,000 gallons (8% of total) of waste volume but contains approximately 175,000,000 curies (49% of total) of the radioactivity. The salt waste makes up the remaining 34,400,000 gallons (92% of total) of waste and contains approximately 183,000,000 curies (51% of total) of the radioactivity. Of that salt waste, the supernate accounts for approximately 18,500,000 gallons and 171,000,000 curies of the 183,000,000 curies total salt waste related activity. The saltcake accounts for approximately 15,900,000 gallons and 12,000,000 curies of the remaining salt waste. The sludge contains the majority of the long-lived (half-life > 30-years) radionuclides (e.g., actinides) and strontium. Figure A.2-4 shows the breakdown of this waste. [SRR-LWP-2010-00040]

Radioactive waste volumes and radioactivity inventories reported herein are based on the WCS database, which includes the chemical and radionuclide inventories on a tank-by-tank basis. The 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 waste tanks, waste sample analyses, and influent receipts such as H-Canyon waste and DWPF recycle. Volumes and curies referenced in this appendix are current as of March 2010. [SRR-LWP-2010-00040]

Figure A.2-4: FTF and HTF Waste Tank Composite Inventory



Inventory values as of March 31, 2010
[SRR-LWP-2010-00040]

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. 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. [SRR-LWP-2009-00001] 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.

A.3 Safe Disposal of the Waste

In the 1980s, waste removal operations were initiated in the SRS tank farms. Ultimately, the goal is remove and treat the waste, safely closing the waste tanks and disposing of the waste in one of two final waste forms: glass, which will contain greater than 99% of the radioactivity, and saltstone, which will contain the vast amount of volume but less than one percent of radioactivity. Both the salt waste and the sludge waste must be uniquely treated to prepare these two waste forms for disposal. 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. This separation will be accomplished in SWPF. However, until the startup of the SWPF, additional salt treatment processes, known as Interim Salt Processing, will be used to accomplish this separation on a limited amount of the lower activity salt waste.

A.3.1 Salt Processing

A final DOE technology selection for salt processing was completed and a Salt Processing EIS ROD was issued in October 2001. [DOE/EIS-0303 ROD] The ROD designated Caustic Side Solvent Extraction (CSSX) as the preferred alternative to be used to separate cesium from the salt waste. Based on the current *Liquid Waste System Plan*, DOE anticipates using a two-phase, three-part process to treat salt waste. [SRR-LWP-2009-00001]

During the initial Interim Salt Processing phase, relatively small volumes of the lower activity will be treated using the following two salt waste treatment processes:

- **Deliquification, Dissolution and Adjustment (DDA):** For salt waste in Tank 41 as of June 9, 2003, the treatment of deliquification (i.e., extracting the interstitial liquid) is sufficient to produce a salt waste that meets the Saltstone Production Facility (SPF) WAC. Deliquification is an effective decontamination process because the primary radionuclide in salt is Cs-137, which is highly soluble. To accomplish the process, the saltcake is first deliquified by draining and pumping. The deliquified saltcake is dissolved by adding water and pumping out the salt solution. The resulting salt solution is given time to allow additional insoluble solids to settle prior to being sent to the SPF feed tank. If necessary, the salt solution may be aggregated with other Tank Farm waste to adjust batch chemistry for processing at SPF.
- **Actinide Removal Process (ARP)/ Modular CSSX Unit (MCU):** For salt waste in selected waste tanks (e.g., Tank 25), further decontamination is performed. Salt waste from these waste tanks first will be sent to ARP. In ARP, monosodium titanate is added to the waste as a finely divided solid. Actinides are sorbed on the monosodium titanate and then filtered out of the liquid to produce a stream that is sent to MCU. The salt solution is further treated to reduce the concentration of Cs-137 using the CSSX process.

Interim Salt Processing will be utilized pending the construction and start-up of the SWPF, the second and more robust phase of salt processing. The SWPF is a large, high-capacity facility that incorporates both the ARP and CSSX processes in a full-scale shielded facility capable of handling and effectively decontaminating salt waste with high levels of radioactivity.

In addition, DOE is exploring the viability of augmenting salt processing capabilities using rotary microfilters and ion exchange columns installed in Type IIIA waste tank risers. The salt solution would be struck with monosodium titanate similar to the actinide removal processes associated with Interim Salt Processing and SWPF, and the insoluble solids and monosodium titanate would be filtered using rotary microfiltratrion technology. This clarified salt solution would then pass through ion exchange column(s) designed to target the removal of cesium.

Figure A.3-1: Vitrification Canister Prior to Use

A.3.2 Sludge Processing

Sludge is “washed” to reduce the amount of 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 SRS waste tanks will be blended into separate sludge “batches” to be processed and fed to DWPF for vitrification.

Final processing for the washed sludge and the high-activity fraction of the salt waste occurs at DWPF. This waste includes monosodium titanate sludge from ARP or SWPF, the cesium strip effluent from MCU or SWPF, and the washed sludge slurry. 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 (Figure A.3-1). As the filled canisters cool, the molten glass solidifies (Figure A.3-2), 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). A low-level recycle waste stream from DWPF is returned to the tank farms. DWPF has been fully operational since 1996. [SRR-LWP-2009-00001]

Figure A.3-2: Sample of Vitrified Radioactive Glass

A.3.3 Saltstone: On-Site Disposal of Low-Level Waste

The Saltstone Facility, located in Z Area, consists of two facility segments: SPF and SDF. The SPF is permitted as a wastewater treatment facility per SCDHEC Regulations. SPF receives and treats the decontaminated salt solution to produce saltstone by mixing the low-level waste liquid stream with cementitious materials (cement, flyash, and slag). A slurry of the components is pumped into the disposal cells located in SDF, where the saltstone grout solidifies into a monolithic, non-hazardous, solid low-level waste form called saltstone. SDF is permitted as an Industrial Solid Waste Landfill Facility (ISWLF) site. [SRR-LWP-2009-00001]

The SDF will be comprised of many large disposal units. Each of the disposal units will be filled with saltstone. The saltstone itself provides primary containment of the waste, and the walls, floor, and roof of the disposal units provide additional engineered barriers.

The current active disposal unit (Vault 4) dimensions are approximately 200 feet wide, 600 feet long and 30 feet high. The disposal unit is divided into two units which are 200 feet wide and 300 feet long, with a 3-inch separation gap between the units. Each unit is further divided into six cells, with each cell measuring approximately 100 feet x 100 feet. Thus, Vault 4 is comprised of 12 cells each approximately 100 feet x 100 feet. The original SDF disposal unit (Vault 1) dimensions are approximately 100 feet wide, 600 feet long and 27 feet high. The disposal unit is divided into two units which are 100 feet x 300 feet with a 3-inch separation gap between units. Each unit is further divided into three cells, with each cell measuring approximately 100 feet x 100 feet. Thus, Vault 1 is comprised of six cells each approximately 100 feet x 100 feet. [SRS-REG-2007-00002]

Future disposal cells are planned to be nominally 150 feet diameter by 22 feet high each and will be designed in compliance with provisions contained in the *Consent Order of Dismissal in Natural Resources Defense Council, et al. v. South Carolina Department of Health and Environmental Controls, et al. (South Carolina Administrative Law Court, August 7, 2007)*. The first of the future disposal units, comprised of Disposal Cells 2A and 2B, is currently under construction in the SDF (Figure A.3-3). [C-CG-Z-00030]

Figure A.3-3: Saltstone Facility

Closure operations will begin near the end of the active disposal period in the SDF, i.e., after most or all the disposal units have been constructed and filled. Backfill of native soil will be placed around the disposal units. The present closure concept includes an erosion barrier, two drainage layers along with backfill and a vegetative cover. [WSRC-STI-2008-00244]

Construction of the Saltstone Facility and the first two disposal units was completed between February 1986 and July 1988. The Saltstone Facility started radioactive operations June 12, 1990. Future disposal cells will be constructed on an as-needed basis in coordination with salt processing production rates. [SRR-LWP-2009-00001]