Technical Evaluation Report

For F-Area Tank Farm Facility, Savannah River Site, South Carolina

Final Report

U.S. Nuclear Regulatory Commission
Office of Federal and State Materials
and Environmental Management Programs
Washington, DC 20555-0001

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Acronyms and Abbreviations

ac	acre(s)
ADMP	Advanced Design Mixer Pump
ALARA	As Low As Reasonably Achievable
Bq	becquerels
CFR	Code of Federal Regulations
CSR	Chemical Sludge Removal
Ci	curie(s)
cm	centimeter(s)
CZ	contaminated zone
DOE	United States Department of Energy
DSA	Documented Safety Analysis
DWPF	Defense Waste Processing Facility
ECC	Enhanced Chemical Cleaning
Eh	Measure of reduction (or oxidation) potential
EIS	Environmental Impact Statement
EPA	United States Environmental Protection Agency
FEPs	Features, Events, and Processes
FDB	F-Tank Farm Diversion Box
FFA	Federal Facility Agreement
FGR	Federal Guidance Reports
FMB	Fourmile Branch
ft	feet
FTF	F-Tank Farm
FTF PA	F-Tank Farm Performance Assessment
g	gram(s)
gal	gallon(s)
GCL	Geosynthetic Clay Liner
GCU	Gordon confining unit
GSA	General Separations Area
HA	Hazard Analysis
HDPE	High Density Polyethylene
HLW	High Level Waste
hr	hour(s)
HRR	Highly Radioactive Radionuclide
HTF	H-Tank Farm
ICRP	International Commission on Radiological Protection
ISCM	Integrated Site Conceptual Model
in	inch(es)
K_ds	Distribution Coefficients
kg	kilogram(s)
km	kilometer(s)
lb	pound(s)
l	liter(s)
LLW	Low Level Waste
m	meter(s)
MCCs	Moisture Characteristic Curve

MEP	Maximum Extent Practical
mi	mile(s)
mil	Mils
mol	mole(s)
mrem	millirem(s)
MSL	Mean Sea Level
mSv	milliSievert(s)
NCRP	National Council on Radiation Protection and Measurements
NDAA	Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005
Ni	Nickel
NRC	United States Nuclear Regulatory Commission
PA	Performance Assessment
pH	Measure of acidity or alkalinity of a solution
PMP	Probably Maximum Precipitation
psi	pounds per square inch
Pu	Plutonium
RAI	Requests for Additional Information
RMEP	Removal to the Maximum Extent Practical
RPP	Radiation Protection Program
s	second(s)
SA	Special Analysis
SCDHEC	South Carolina Department of Health and Environmental Control
SEE	Systems Engineering Evaluation
SMP	Submersible Mixer Pump
SOF	Sum of Fractions
SRNL	Savannah River National Laboratory
SRS	Savannah River Site
Sv	Sieverts
Tc	Technetium
TCCZ	tan clay confining zone
TEDE	Total Effective Dose Equivalent
TER	Technical Evaluation Report
U	Uranium
UCL	Upper Confidence Limit
USGS	US Geological Survey
UTR	Upper Three Runs
UTR-LZ	Upper Three Runs- Lower Zone
UTR-UZ	Upper Three Runs- Upper Zone
WCS	Waste Characterization System
WIR	Waste Incidental to Reprocessing
WMC	Waste Mixing Chamber
yr	year(s)

EXECUTIVE SUMMARY

On September 30, 2010, the U.S. Department of Energy (DOE) submitted the "Draft Basis for Section 3116 Determination for Closure of F-Tank Farm (FTF) at the Savannah River Site" (basis document) to the U.S. Nuclear Regulatory Commission (NRC) for review, as required by Section 3116 of the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA). Section 3116 of the NDAA requires DOE to consult with NRC when determining whether certain wastes associated with spent fuel reprocessing are not high-level wastes (HLW). The purpose of DOE's draft basis document is to demonstrate that stabilized residuals in the 20 remaining underground radioactive waste storage tanks in the FTF as well as related ancillary facilities, are not high-level waste and therefore, meet the criteria in NDAA Section 3116.

Consistent with the requirements in Section 3116 of the NDAA, DOE consulted with NRC on the draft basis document. NRC has conducted a review of the draft basis document and offers recommendations to DOE in this Technical Evaluation Report (TER) for closure of the 20 remaining underground radioactive waste storage tanks in the FTF. In addition, NRC staff offered specific recommendations for Tanks 18 and 19 given the near term closure efforts associated with these tanks. Tanks 18 and 19 are also the only cleaned tanks for which a final inventory has been developed. NRC's review results and recommendations are being provided to DOE for consideration only and are not intended to represent any regulatory authority related to DOE's waste determination (WD) activities. DOE is solely responsible for determining whether the waste streams addressed in the draft basis are not HLW and therefore, satisfy the requirements in Section 3116 of the NDAA.

Once a waste determination has been made, the NDAA requires NRC, in coordination with the State of South Carolina, to monitor DOE disposal actions as a means of assessing DOE's compliance with the performance objectives in Title 10, Code of Federal Regulations, Part 61 (10 CFR Part 61), Subpart C. In addition to NRC and the State of South Carolina, DOE will also monitor activities associated with the closure of FTF to ensure compliance with all statutory requirements in the Atomic Energy Act of 1954, as amended, as well as other applicable laws and regulations.

Upon finalization of the draft waste determination, DOE will proceed with cleaning and closure of the FTF tanks. FTF waste tanks are closed in accordance with requirements in the Savannah River Site's (SRS) Federal Facility Agreement (FFA), a formal agreement between DOE, the U.S. Environmental Protection Agency (EPA) and the South Carolina Department of Health and Environmental Control (SCDHEC). One purpose of this agreement is to establish a schedule by which SRS waste tanks that do not meet secondary containment standards are removed from service. At this time, the FFA schedule calls for the closure of two HLW tanks by 2012.

The FTF and F-Canyon comprise the F area of the SRS. The FTF contains 22 below grade carbon steel and reinforced concrete HLW tanks, which store liquid radioactive waste generated from the chemical separations facility in the F-Canyon. The FTF tanks consist of three basic tank types (I, III, and IV), as described in Section 1.1 herein. Type I is the smallest and oldest of the tank types. Constructed in the early 1950s, these tanks are 23 m (75 ft) in diameter and 7.5 m (24.5 ft) in height with a storage capacity of 2,850 m^3 (750,000 gal). The Type III tanks are

the youngest FTF tanks. At a height of 11 m (34.5 ft), a width of 26 m (85 ft) and a storage capacity of about two times that of the Type I tanks 4,940 m^3 (1,300,000 gal), these tanks were added to the FTF during an eleven year period (1969-1980). Type IV tanks are the largest of the tank types. Designed and sited in the late 1950s, these tanks stand at a height of 11 m (34.5 ft), a width of 26 m (85 ft) and a 4,940 m^3 (1,300,000 gal) storage capacity.

To date, six of the 22 FTF tanks have been cleaned. This includes Tanks 17 and 20, which were closed with approval from SCDHEC prior to the implementation of the NDAA. As previously stated, both a WD and approval from SCDHEC are required prior to the closure of the remaining FTF tanks. In addition to removing the waste, FTF closure also includes cleaning and stabilizing the tank system components (including tanks, vaults, piping, structures, and ancillary equipment).

Initially, DOE employs mechanical techniques (primarily agitation, and pumping) to remove bulk waste. After mechanical cleaning operations, a combination of mechanical and chemical cleaning techniques are employed to clean tanks and annuli. For tanks that have been cleaned, DOE reports that in excess of 99 percent of the waste volume has been removed. The remaining activity in the cleaned tanks is approximately one percent of the activity estimated by DOE prior to significant cleaning activities. DOE estimates that over 99 percent of the total inventory based on a starting point of maximum operational historical inventory is expected to be removed prior to closure. However, DOE does not consider this estimate to be a removal goal.

The NDAA contains three criteria for determining that waste is not HLW. Criterion 1 states that the waste does not require permanent isolation in a deep geologic repository for spent fuel or HLW. Typically all HLW requires deep geological disposal. Criterion 1 recognizes that waste can be disposed of consistent with its risk, not only considering the origin of the waste. If there are no characteristics of the waste that would require geological isolation and Criterion 2 and 3 can be met, then according to the NDAA, the waste is not HLW.

However, Criterion 1 allows DOE to consider that waste may require disposal in a deep geologic repository even though the other criteria of the NDAA can be met. Under certain circumstances (e.g., unique radiological properties of the waste, proliferation concerns), geologic disposal is warranted to protect public health and safety and the environment even if other criteria are satisfied. Because there appears to be no special properties of the waste and there are no proliferation concerns that would necessitate deep geologic disposal, NRC staff notes that NDAA Criterion 1 can be met for all tanks. That is, the cleaned tanks do not require exhumation and disposal in a geologic repository.

Criterion 2 of the NDAA states that the waste has had highly radioactive radionuclides (HRRs) removed to the maximum extent practical (MEP). To assess conformance with Criterion 2, NRC staff assessed DOE's estimated waste inventory, identification of HRRs, selection of treatment technology, and demonstration of removal to the MEP including the costs and benefits of additional radionuclide removal.

NRC staff's key review results related to Criterion 2 are as follows:

- In general, DOE's approach to developing inventories for cleaned Tanks 18 and 19 is reasonable, although DOE's approach to quantifying uncertainty could be improved.

- DOE's approach to developing inventories for tanks that have not been cleaned is reasonable and tends to bias inventory estimates high. NRC staff will continue to monitor DOE's efforts towards reducing the FTF source term as it pertains to the Criterion 3 evaluation, including optimization associated with as low as is reasonably achievable criteria.

- DOE's process for identification of HRRs is reasonable and should continue to be evaluated as tank farm closure progresses.

- DOE has a program in place to identify, evaluate, and implement cleaning technologies to remove HRRs to the MEP.

- With regard to tanks that have not yet been cleaned, DOE has a process in place to demonstrate removal of HRRs to the MEP, but this process could benefit from additional detail.

NRC staff's key recommendations related to Criterion 2 are as follows:

- NRC staff recommends that DOE explore methods to improve the process by which residual waste volumes and associated uncertainty are estimated.

- NRC staff recommends that DOE continue to evaluate its HRR list as additional information becomes available, to the extent that the list of HRRs is used to inform decisions such as the selection of radionuclides characterized in residual waste, selection of treatment technologies, and screening of radionuclides for the purpose of Performance Assessment (PA) calculations.

- As practical, NRC staff recommends that DOE continue to participate in technology exchanges and continuously evaluate new cleaning technologies as they become available, rather than defaulting to previously selected technologies, or relying on previous evaluations for technology selection.

- NRC staff recommends that DOE consider how it might better assess and optimize the effectiveness of selected technologies (e.g., obtain better baseline information).

- NRC staff recommends that DOE provide more emphasis on removal of HRRs in its technology selection process and provide a clearer linkage between the Criterion 2 evaluation and the PA results, including consideration of long-term risks associated with the FTF facility.

- NRC staff recommends that DOE provide additional detail on the methods to be used to demonstrate removal to the MEP to ensure consistent (non-arbitrary) application of the criterion.

- Using the cost-benefit analysis performed for Tanks 18 and 19 as an example, NRC staff recommends that DOE perform a more rigorous cost-benefit analysis that includes consideration of the long-term benefits associated with additional radionuclide removal, to demonstrate removal to the MEP for FTF tanks that will be cleaned in the future.

Considering additional information provided by DOE in its requests for additional information (RAI) responses, Tank 18 now appears to be the single largest risk driver for the FTF facility with Tank 18 scheduled for closure per the FFA by December 2012. Given its risk-significance, the NRC staff thinks that DOE should more fully evaluate the practicality of additional radionuclide removal from Tank 18 and explore options for delaying final closure (i.e., grouting) of Tank 18[1] for the reasons listed below. It is important to note that the risk associated with a short delay in the grouting of Tank 18 on the order of a few years is not expected to be significant given ongoing operation and maintenance of the FTF and the fact that a large portion of the residual tank waste has been removed; however, a decision to delay of the grouting of Tank 18 should consider any associated short term risks.

- Insufficient information was provided to the NRC staff related to the costs and benefits of additional radionuclide removal and other factors influencing the decision regarding practicality of additional HRR removal from Tank 18. The NRC staff recommends that DOE provide additional information or perform additional analysis to support the Criterion 2 demonstration for Tank 18.

- Significant technical uncertainties exist with respect to DOE's ability to meet the performance objectives in 10 CFR Part 61, Subpart C, that the NRC staff thinks can be addressed in the near-term (e.g., solubility studies). Permanent closure activities such as grouting of the waste tank may make it more difficult for DOE to evaluate or reduce the risks associated with this waste tank in the future, if risk reduction is deemed necessary pending results of future research. Additionally, the results of the near term studies could reduce the extent to which other uncertainties will need to be addressed to support Criterion 3 of the NDAA for tank farm closure.

- A delay in Tank 18 grouting could provide additional time for alternative technologies to be developed (e.g., the improved Mantis design that is anticipated to be used on the H-Tank Farm, Type IV tanks), that could result in greater removal of HRRs from Tank 18, if additional HRR removal is deemed practical.

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[1] Although the information provided for Tank 19 under Criterion 2 is similar to that provided for Tank 18, given the lower inventory and risk associated with Tank 19, NRC staff thinks that final closure of Tank 19 can proceed as planned.

Criterion 3 of the NDAA states that waste will be disposed of in compliance with 10 CFR Part 61, Subpart C, performance objectives. Subpart C provides for site-stability and sets requirements for protection of the public, the inadvertent intruder, and individuals during operations. It is important to note that NRC staff is not making a conclusion on the ability of the FTF to meet the requirements of the performance objectives in 10 CFR Part 61, Subpart C due to uncertainty in the final inventories for the remaining tanks. To assess conformance with Criterion 3, NRC staff evaluated: (i) waste classification for tank system components; (ii) PA results (including infiltration, near-field release, far-field transport, dose methodology, and exposure assessment); (iii) inadvertent intruder analysis; (iv) radiation protection program (RPP) for individuals during operations; and (v) stability of the disposal facility after closure. The NDAA also requires disposal of waste in accordance with state-approved closure plans or state-issued permits.

NRC staff's key review results related to Criterion 3 are as follows:

- Due to lack of transparency and traceability of DOE's PA documentation, it is difficult to determine whether features, events, and processes (FEPs) are comprehensively evaluated in DOE's PA.

- DOE's approach to developing inventories for FTF sources is reasonable and tends to err on the side of conservatism for tanks that have yet to be cleaned.

- Major degradation processes are evaluated in DOE's cementitious material and steel liner modeling analyses; however, assumptions regarding long-term performance of cementitious materials and liners are highly uncertain and do not appear to be fully supported by site-specific evidence and other observations.

- Although DOE's overall approach to modeling waste release is reasonable, DOE assumptions regarding solubility limiting phases, solubility limits, and chemical transition times are particularly risk-significant and have not been confirmed through waste characterization and experimentation.

- Additional characterization, experimentation, or modeling may be needed to reduce uncertainty in assignment of risk-significant parameters such as basemat and natural system K_ds (distribution coefficients).

- DOE's PA modeling results indicate potential excessive dispersion in the near-field and far-field models.

- Far-field model construction and calibration are generally acceptable; however, the calibration process could be improved and made more transparent in future PA updates.

- NRC staff is convinced that large voids do not currently exist in the subsurface along FTF flow paths to the 100 m (330 ft) point of compliance. However, calcareous zones that have undergone dissolution may still represent high permeability pathways that could have a significant effect on contaminant flow and transport. Additional information

could be collected during the monitoring period to support DOE's modeling treatment of the calcareous zones in the lower portion of the UTR aquifer.

- DOE's dose methodology approach is well supported and adequate for the purposes of demonstrating compliance with the performance objectives in 10 CFR Part 61, Subpart C.

- NRC staff provides several recommendations in this TER where additional model support would be needed prior to tank closure to provide reasonable assurance that closure activities would comply with the requirements in 10 CFR 61.41.

- DOE developed reasonable exposure scenarios to evaluate the inadvertent intruder and demonstrated that performance objectives in 10 CFR 61.42 could be met. However, due to overlap in the groundwater analyses used to support the 10 CFR 61.41 and 10 CFR 61.42 evaluations, compliance with 10 CFR 61.42 is tied to resolution of technical issues associated with the 10 CFR 61.41 analysis.

- NRC staff noted a few instances where biosphere parameters are not fully supported as detailed in Section 4.2.17.

- DOE's effort to study uncertainty in modeling predictions and identify important model sensitivities is commendable.

- Improvements can be made to DOE's parameter distribution assignments, hybrid modeling approach, benchmarking process, and evaluation and interpretation of probabilistic modeling results as detailed in Section 4.2.19 of this TER.

- DOE can demonstrate compliance with protection of individuals during operations (10 CFR 61.43).

- NRC staff thinks that additional information is needed to evaluate site-stability under 10 CFR 61.44 as detailed in Section 4.3.4.

NRC staff's primary recommendation related to Criterion 3 is as follows:

- NRC staff recommends that DOE conduct waste release experiments to increase support for key modeling assumptions related to: (i) the evolution of pH and Eh in the grouted tank system over time; (ii) identification of HRR association with solid phases comprising the residual wastes; and (iii) expected solubility of HRRs under a range of environmental or service conditions that the residual wastes in the contaminated zone are expected to be exposed to over time. Implementation of this recommendation is deemed crucial for NRC staff to have reasonable assurance that the performance objectives in 10 CFR Part 61, Subpart C can be met. Given the risk-significance of Tank 18 to the overall PA and the short timeline for closure of this tank, DOE should initiate discussions with NRC staff regarding implementation of this recommendation for Tank 18 as soon as practical. Experiments to address this recommendation should be conducted prior to final closure of this single tank. Results of the Tank 18 residual waste

experiments will be evaluated by NRC staff to determine the need for additional data collection, experiments, modeling, etc. for Tank 18, as well as other FTF tanks. Additional information regarding NRC staff's recommendations in this area, including details on the suggested implementation of other recommendations listed below will be provided in NRC staff's plan for monitoring the FTF later in FY2012.

Additional NRC staff recommendations related to Criterion 3 include the following:

- NRC recommends that DOE perform a systematic scenario analysis process in which FEPs are identified, screened, and dispositioned using transparent and traceable documentation of the FEPs considered, the screening arguments, and how FEPs are implemented in the models to support future waste determination efforts.

- NRC supports DOE's commitment to sample each tank following waste retrieval activities and will follow-up with DOE on sampling and analysis of cleaned tanks during the monitoring period.

- DOE should consider how it might improve far-field model calibration and transparency in future updates to its PA.

- DOE should continue to evaluate the appropriateness of selected transport parameters (e.g., dispersivities and K_ds, particularly for calcareous zones) and selection of sorption models (see discussion in Section 4.2.9.4 on Pu transport) during the monitoring period.

- DOE should consider additional data collection related to calcareous zone outcrop locations and tracer tests to provide further support for the adequacy of its modeling treatment of the Upper Three Runs-Lower Zone (UTR-LZ) aquifer.

- Finally, DOE should evaluate the need for additional vertical or horizontal mesh refinement to ensure that contaminant plumes are not artificially dispersed over the volume of the cells in the far-field model and that time discretization is adequate.

For a broader and more detailed discussion of DOE SRS' waste determination and PA approach and NRC staff's evaluation, please see the appropriate Sections of the TER.

In summary, statements and recommendations contained herein are based on NRC staff review of DOEs Draft Section 3116 Waste Determination dated September 30, 2010, DOE responses to NRC staff's requests for additional information, supporting references, and information provided during meetings and teleconferences (summaries of which are a matter of public record) between DOE and NRC in preparing this TER. If, in the future, DOE determines it is necessary to revise its assumptions, analysis, design, or waste management approach and those changes are important to meeting the criteria of the NDAA, DOE should consult once again with NRC. Note that NRC is providing consultation to DOE as required by the NDAA, and NRC is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is HLW. This NRC staff assessment is a site-specific evaluation and is not a precedent for any future decisions regarding non-HLW or incidental waste determinations at SRS or other DOE sites.

1. INTRODUCTION

1.1 Facility and Site Description

The Savannah River Site (SRS) is a 780 km^2 (300 mi^2) site, located in the upper coastal plain of South Carolina, adjacent to the Savannah River. The site is approximately 12 miles south of Aiken, South Carolina, and 15 miles southeast of Augusta, Georgia. It is owned by the Department of Energy (DOE). Operation at the site began in 1951. The primary use for the site was the production of nuclear material for national defense. Ancillary operations included applications for research, medical, and space programs.

Between 1954 and 1986, DOE generated significant quantities of radioactive waste from the reprocessing of spent nuclear fuel. The radioactive waste was also generated from the production of targets for nuclear weapons and production of material for space missions. This waste was managed by storage in underground tanks throughout the site. These locations were designated as tank farms.

DOE has 51 underground waste storage tanks located at two tank farms. F-Tank Farm (FTF) contains 22 tanks and H-Tank Farm houses the remaining 29 tanks. DOE employed a variety of tank designs at the site. The tank designs varied in capacity, containment strategies, and internal support features. Twenty-seven tanks were designed and built with full secondary containment and leak detection systems. These tanks meet the EPA requirements for underground storage tanks. The remaining 24 tanks do not have full secondary containment and do not meet EPA requirements. Twelve tanks without secondary containment have leakage histories, but sufficient waste has been removed from these such that there are currently no active leak sites. Two tanks in the FTF, which did not have secondary containment, have been closed and grouted. Of the 49 remaining operational tanks, 29 are in H-Tank Farm and 20 are in FTF (12 of the 20 FTF tanks are non-compliant).

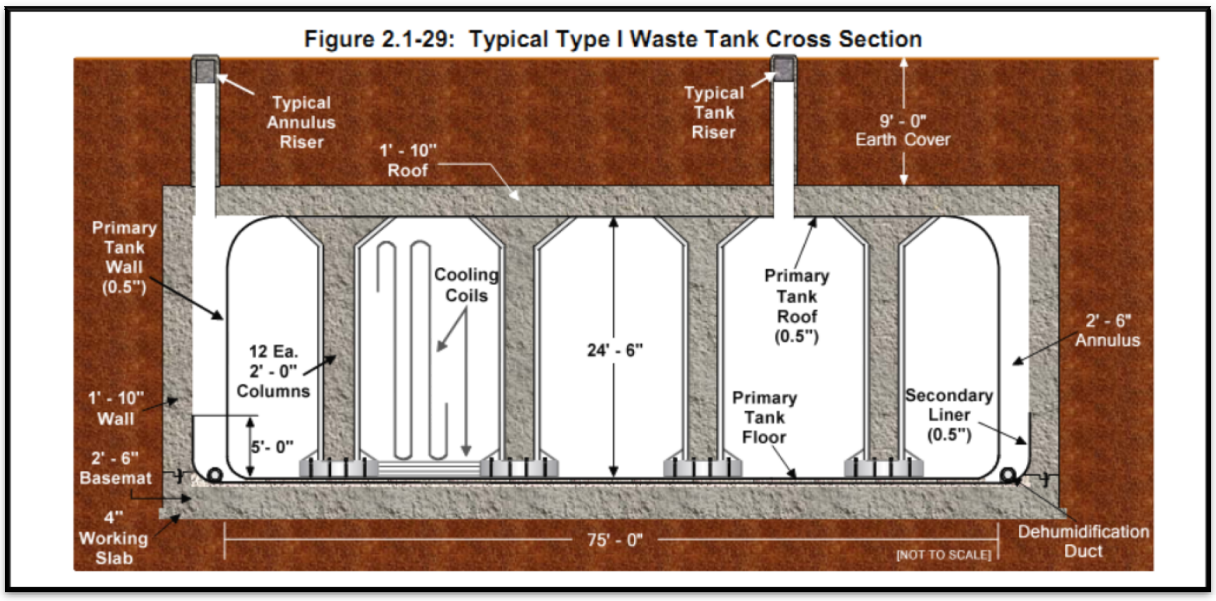
The FTF occupies approximately 0.08 km^2 (20 acres) in the F-Area within the General Separations Area (GSA) near the center of the SRS. The FTF contains 22 carbon steel waste tanks of three different basic designs (see Figures 1-1, 1-2, and 1-3), eight with a nominal capacity of 2,850 m^3 (750,000 gal) per tank and 14 with a nominal capacity of 4,940m^3 (1,300,000 gal) per tank. Most of the waste in these tanks originated in the SRS F Canyon facility, which recovered nuclear material produced in the site's nuclear production reactors. DOE's approach for treating and disposing of FTF waste tanks, ancillary equipment, and its residual radioactivity is evaluated by NRC staff in this TER.

1.1.1 Land and Water Use

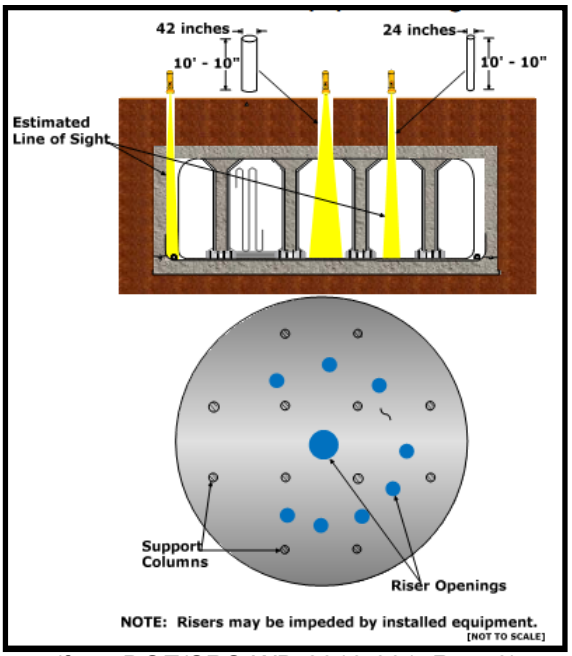
About 90 percent of SRS's land area consists of natural forests and managed pine plantations. Approximately 10 percent of the total land area is developed or used for industrial facilities. Land use within the GSA is classified as heavy nuclear industrial. Land within 8 km (5 mi) of the FTF is currently used either for industrial purposes or is forested. Pine plantations managed for timber production by the U.S. Forest Service occupy regions surrounding the F-Area. The SRS End State Vision (PIT-MISC-0089) is based on the following assumptions about land ownership and use: (i) the entire site will be owned and controlled by the Federal Government in perpetuity; (ii) the property will be used only for industrial purposes; (iii) site boundaries will

remain unchanged, and (iv) residential use will not be allowed onsite. Because the SRS is controlled, the public does not have unrestricted access to any site facility. The SRS is 94 percent forested, with the remainder used as aquatic wildlife habitats or developed for industrial purposes.

Figure 1-1 Typical Type I Tank Cross Section (a) and Plan View Map of Access Ports (b)



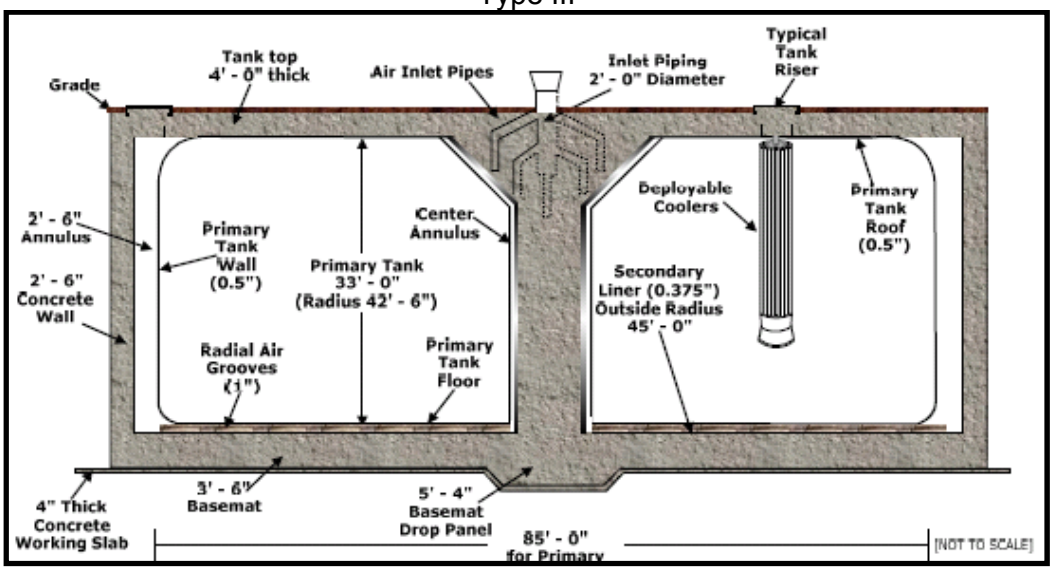
(from DOE/SRS-WD-2010-001, Rev. 0)



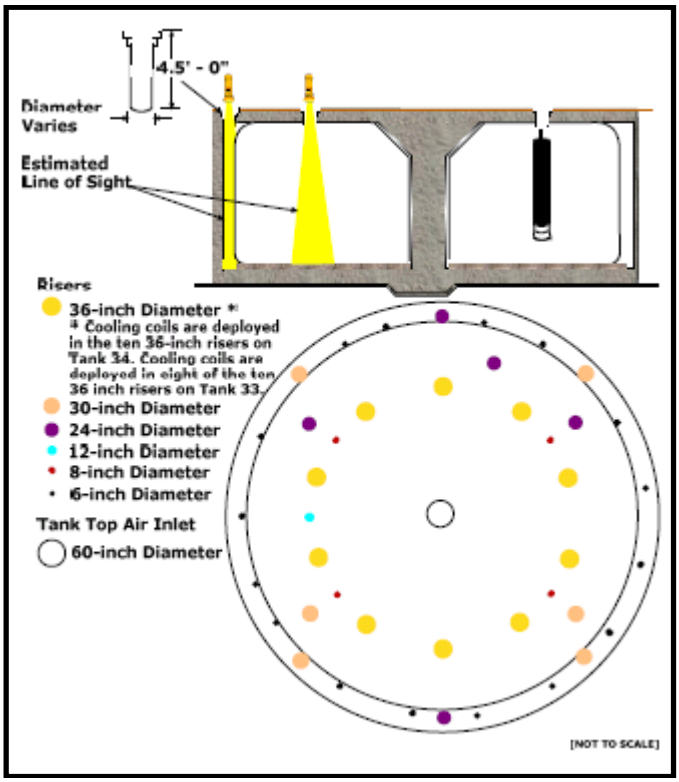
(from DOE/SRS-WD-2010-001, Rev. 0)

Figure 1-2 Typical Type III/IIIA Tank Cross Sections (a and c) and Plan View Map of Access Ports (b and d)

Type III



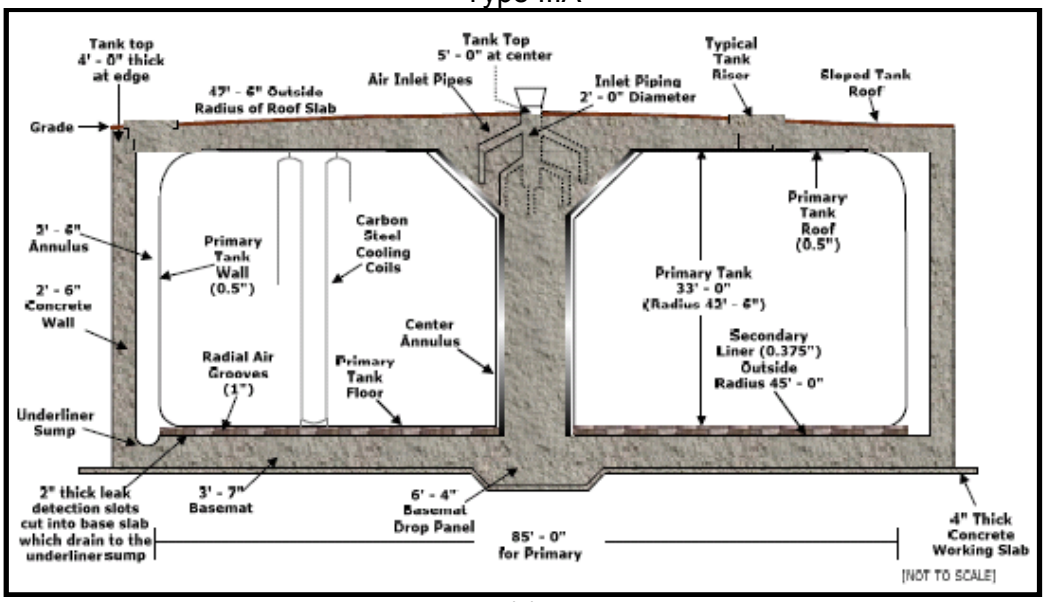
(a)



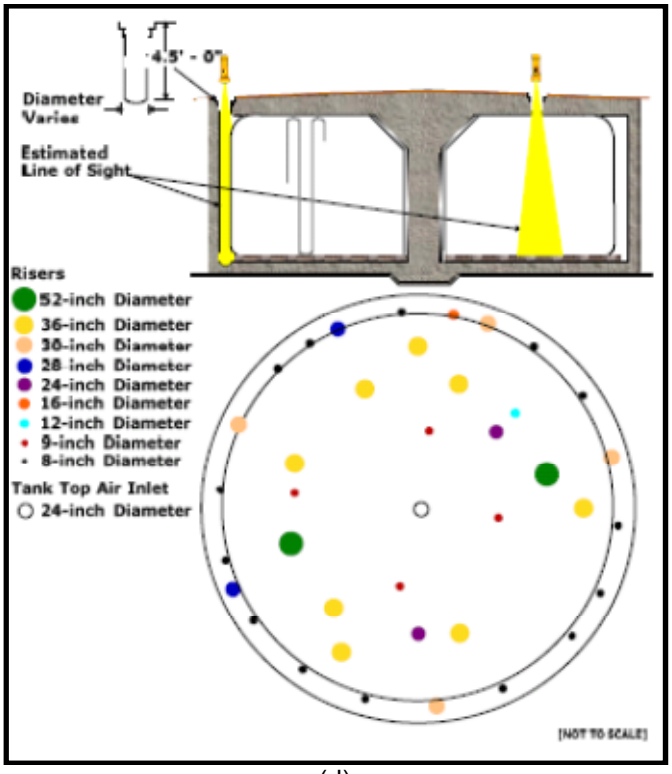
(b)

Figure 1-2 (cont)

Type IIIA

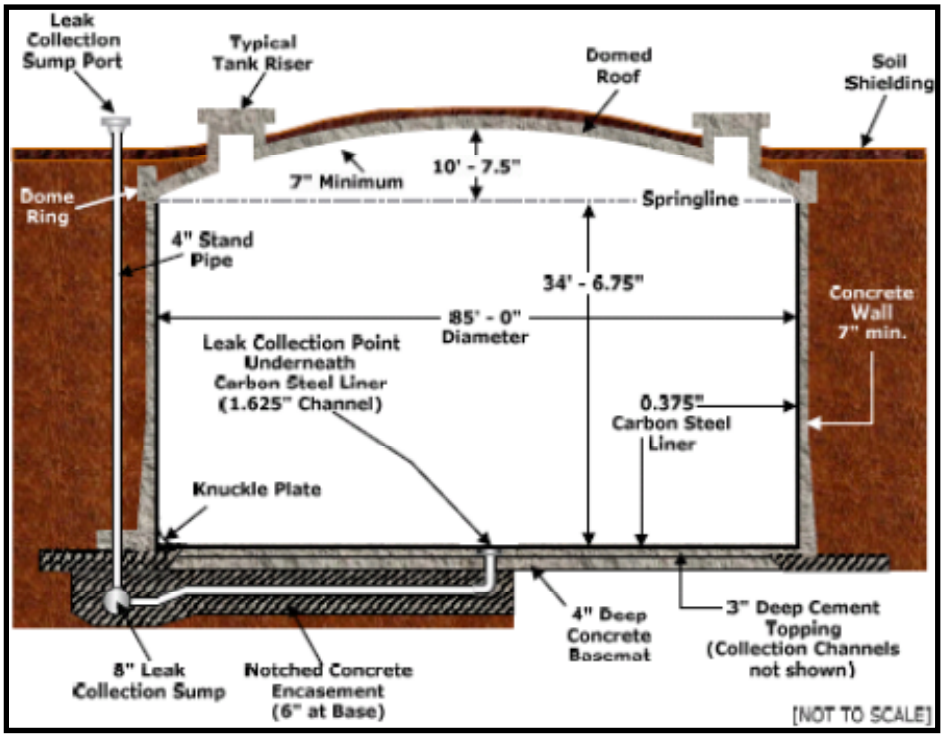


(c)

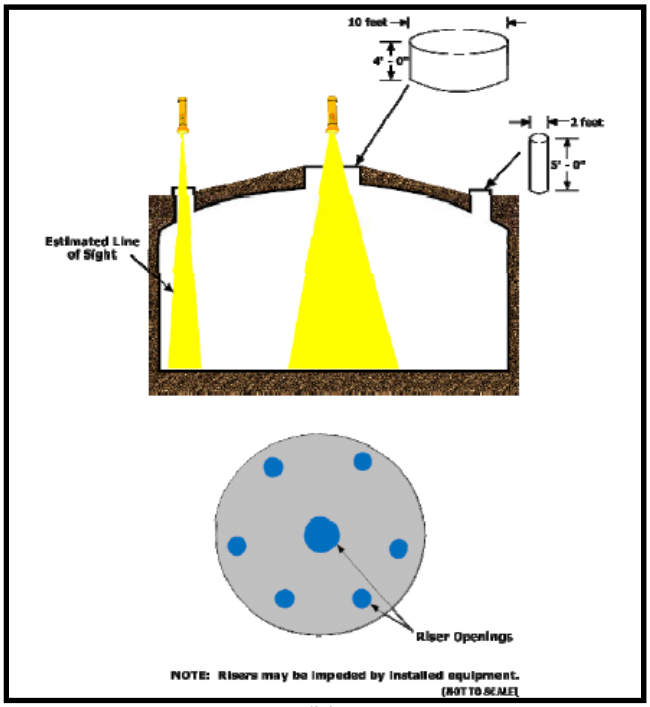


(d)

Figure 1-3 Typical Type IV Tank Cross Section (a) and Plan View Map of Access Ports (b)



(a)



(b)

The Savannah River bounds the SRS to the southwest for approximately 32 km (20 mi) and the site is 257 river kilometers (160 river miles) from the Atlantic Ocean. Five upstream reservoirs (Lakes Jocassee, Keowee, Hartwell, Richard B. Russell, and Clarks Hill/Strom Thurmond) are tapped as needed to supplement downstream river flow. Savannah River flow averages approximately 295 m^3/s (10,400 ft^3/s) at the SRS. The major Savannah River tributaries on the SRS are Upper Three Runs (UTR) (a clear, cold, spring-fed stream), Fourmile Branch (FMB), Pen Branch, Steel Creek, and Lower Three Runs. These tributaries, which drain most of the SRS to the Savannah River, are not commercial water sources. These streams originate on the Aiken Plateau in the Coastal Plain and descend 15 to 61 m (50 to 200 ft) before discharging to the Savannah River.

Downstream of the SRS, the Savannah River supplies domestic water and is used for commercial and sport fishing, boating, and other recreational activities. The Savannah River floodplain supports an extensive swamp that covers approximately 40 km^2 (15 mi^2) of the SRS; a natural levee separates this swamp from the river. The SRS has widespread wetlands, most of which are associated with floodplains, streams, or impoundments. Approximately 200 Carolina bay isolated wetlands are dispersed throughout the uplands of the SRS. The Savannah River and streams on the SRS are classified by the South Carolina Department of Health and Environmental Control (SCDHEC) as freshwaters, defined as surface water suitable for primary and secondary contact recreation, drinking water after treatment, fishing, survival and propagation of an indigenous aquatic community of fauna and flora, and industrial and agricultural uses. Prominent surface water features at the SRS include Par Pond and L Lake (Figure 1). Par Pond and L Lake are former reactor cooling water impoundments that cover approximately 11 and 4 km^2 (2,700 and 1,000 ac) respectively. Neither water body lies within watersheds associated with the FTF.

UTR, its unnamed tributaries, Crouch Branch, and FMB and its tributaries are perennial streams near the FTF. U.S. Geological Survey (USGS) gauging stations monitor flow in UTR and FMB near F-Area, and water temperature and quality are measured monthly. The SRS has National Pollutant Discharge Elimination System permits and an associated monitoring plan for discharge of non-radioactive liquid effluents to surface waters. Continuous monitoring of water flowing through streams occurs downstream of several process areas to detect and quantify levels of radioactivity in effluents transported to the Savannah River.

1.1.2 Terrestrial and Aquatic Biota

Approximately 200 Carolina bays on SRS exhibit a range of plant communities from herbaceous marsh to forested wetland. The swamp forest near the Savannah River 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 continually flooded are dominated by second-growth bald cypress and water tupelo. Nearer the F-Area, the FMB and UTR seepline areas are located in a bottomland hardwood forest community. The forest canopy is dominated by sweetgum, red maple, and red bay; sweet bay is common. The understory includes an 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 FMB are tag alder, willow, sweetgum, and wax myrtle. The smooth purple coneflower is an endangered plant species occurring on the SRS. (DOE-EIS-0303).

SRS supports abundant terrestrial and semi-aquatic wildlife. Animal populations include forest dwelling and urban wildlife, several commercially and recreationally important species, and a few threatened and endangered species. Fauna common to the SRS site includes 55 mammalian species (e.g., white-tailed deer, feral hog, raccoon, beaver, otter, gray fox, opossum, bobcat, gray squirrel, fox squirrel); macroinvertibrates and fish (observed in UTR, e.g., shiners, sunfish, pirate perch, madtoms, darters); 255 bird species, including waterfowl and wading birds (e.g., eastern cottontail, mourning dove, northern bobwhite, eastern wild turkey), 44 amphibian species and 59 reptile species (i.e., salamanders, frogs and toads, crocodiles, turtles, lizards, and snakes). (WSRC-TR-2005-00201).

The FTF is located within a densely developed, industrialized area of the SRS, providing habitat for only those animal species typically classified as urban wildlife. Species commonly encountered include the Southern toad, green anole, rat snake, rock dove, European starling, house mouse, opossum, and feral cats and dogs. Grasses and landscaped areas also provide marginal terrestrial wildlife habitat. Ground-foraging bird species (e.g., American robin, killdeer, and mourning dove) and small mammals (e.g., cotton mouse, cotton rat, and Eastern cottontail) are sometimes present on lawns and within landscaped areas.

Loblolly pine trees are assumed to invade the FTF engineered closure caps after the 100-yr period of institutional control, and their root systems are predicted to penetrate and thereby affect the durability and functionality of the geosynthetic clay liner (GCL). DOE indicates that the erosion barrier of their engineered closure cap design will preclude intrusion of burrowing animals; thus, this degradation mechanism is not considered for performance modeling purposes.

1.1.3 Local Meteorology and Climatology

Climate at SRS is humid and subtropical with relatively short, mild winters and long, warm summers. Summer winds are light. Summer high temperatures exceed 32 °C (90 °F) on more than half of all days on average. Scattered afternoon and evening thunderstorms are common. The Bermuda high that dominates summer climates diminishes during autumn as continental air masses become more prevalent, resulting in lower humidity and more moderate temperatures. Low pressure systems migrate through the region during winter months, resulting in a pattern that alternates between warm, moist, subtropical air from the Gulf of Mexico region and cool, dry arctic air. The Appalachian Mountains moderate the extremely cold temperatures associated with Arctic air masses. Less than one-third of winter days have minimum temperatures below freezing on average, and days with temperatures below −7 °C (20 °F) are infrequent. Measurable snowfall occurs once every two years on average. Windy conditions and tornadoes are most prevalent during the spring when temperatures are mild and humidity is relatively low.

1.1.4 Hydrology and Hydrogeology

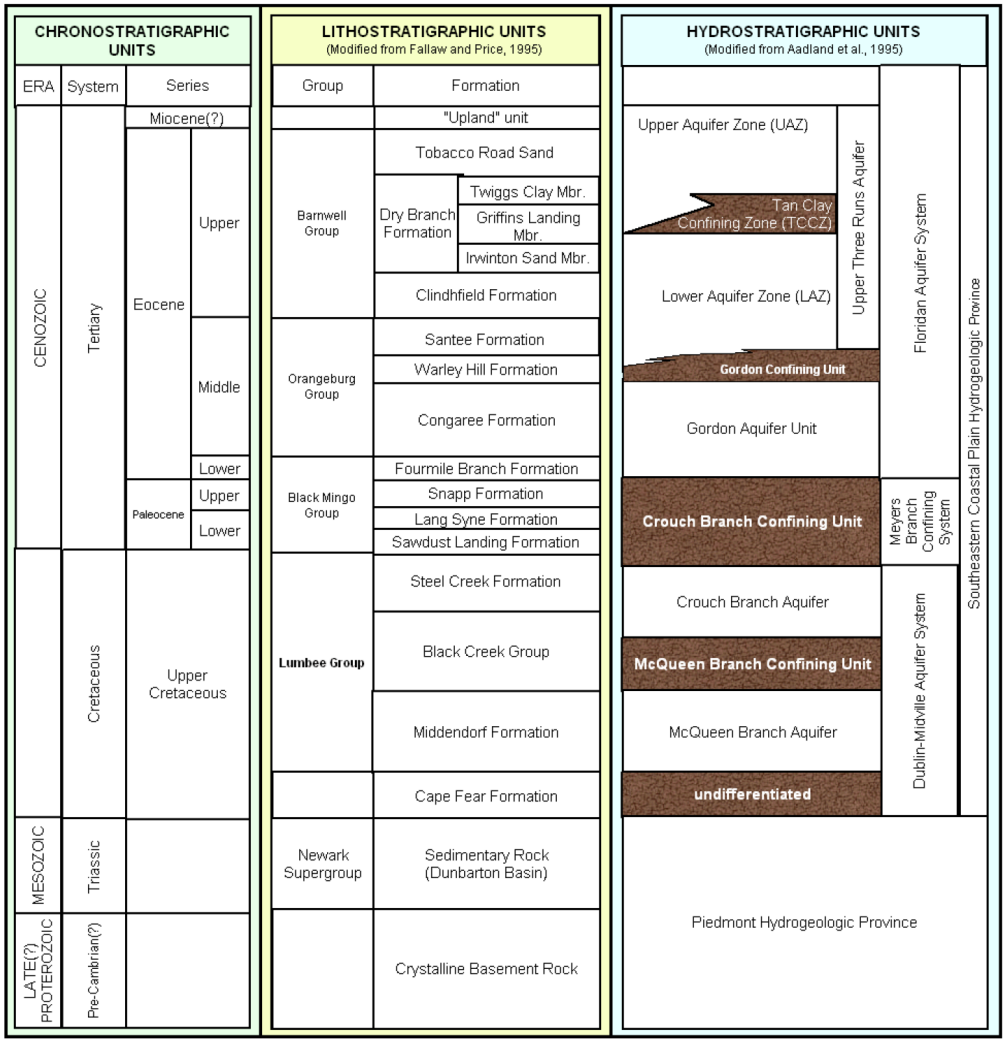
The SRS lies on the Atlantic Coastal Plain, above a southeast-dipping wedge of unconsolidated and semi consolidated sediments forming the Upper Coastal Plain sedimentary sequence. Sediments range in age from Late Cretaceous to Recent and include sands, clays, limestones and gravels. This sedimentary sequence ranges in thickness at the SRS from 213 m (700 ft) at the northwestern boundary to 430 m (1,410 ft) at the southeastern boundary.

The Coastal Plain sedimentary sequence forms a series of aquifers and confining or semi-confining units. Aquifer systems below the SRS include the Floridian and Dublin–Midville. The Upper Coastal Plain sediments lie above and fill-in the Dunbarton basin, a Triassic–Jurassic Rift basin filled with lithified terrigenous and lacustrine sediments; below the Dunbarton basin lies crystalline bedrock of metamorphosed sedimentary and igneous rock that may range in age from Precambrian to late Paleozoic (see Figure 1-4). UTR, a large, long, blackwater stream, drains an area greater than 500 km^2 (195 mi^2). The stream is approximately 40 km (25 mi) long, and has its lower 27 km (17 mi) within Site boundaries. UTR receives more water from groundwater sources than other SRS streams and it flows approximately 3 km (2 mi) to the north of the FTF. The UTR floodplain ranges in width from 0.4 to 1.6 km (0.25 to 1 mi) and has a steep southeastern side and a gently sloping northwestern side. The 100 yr, 1,000 yr and 10,000 yr flood plain for UTR Basin near F-Area is approximately 42.1, 42.7, and 43.6 m (138, 140 and 143 ft) above mean sea level (MSL) (WSRC-TR-99-00369), which is approximately 25.6 m (84 ft) below the lowest waste tank bottoms based on the elevation provided in DOE PA, Rev. 1, Table 3.2-1 (DOE, 2010). The lowest elevation of the lower foundation layer at the bottom of the side slope of the engineered closure cap is approximately 82.3 m (270 ft) above MSL, which is approximately 38.7 m (127 ft) above the 10,000 yr flood plain level (WSRC-STI-2007-00184).

FMB, a smaller perennial blackwater stream, has its headwaters near the center of the SRS, drains a 57 km^2 (22 mi^2) area (including much of F-Area), and traverses 24 km (15 mi) before discharging at the Site boundary to the Savannah River or Beaver Dam Creek. The V-shaped valley of FMB has a floodplain ≤305 m (≤1000 ft) in width with steep to gently sloping sides. FMB flows approximately 1.6 km (1 mi) to the south of the FTF, and is the nearest surface water of significance. Flow within SRS streams ranges from 0.2 m^3/s (8 ft^3/s) in small tributaries to 7 m^3/s (245 ft^3/s) in UTR. The 100,000 yr flood plan for FMB Basin near F-Area is 59 m (194.1 ft) above MSL, which is approximately 10 m (33 ft) below the lowest waste tank bottoms based on the elevation provided in DOE PA, Rev. 1, Table 3.2-1 (SRR-REG-2007-00002, Rev. 1) according to (WSRC-TR-99-00369). Because the FTF is not located within a flood plain region and has adequate topographical relief for runoff, DOE does not consider flooding to be an issue of concern.

The Floridian Aquifer System is of primary importance for estimating the ultimate fate and transport of any residual FTF contaminants. At the SRS, the Floridian Aquifer System is composed of the upper and lower zones of the UTR aquifer above, the Gordon aquifer below, and their associated confining units (the Tan Clay (semi) confining zone within UTR aquifer and the Gordon confining unit (GCU) separating the UTR aquifer from the Gordon aquifer). Groundwater flow in the UTR aquifer is predominantly horizontal with only a small downward component. The saturated thickness of the Upper Three Runs-Upper Zone (UTR-UZ) water-table aquifer is 5 m on average (16 ft), and the thickness of the UTR-LZ aquifer is approximately 20 m (66 ft) (DOE, 2010). Groundwater divides are located between surface water drainages; the FTF sits on a groundwater divide located between FMB and UTR streams (Figure 1-5). DOE expects that contaminants will ultimately discharge to one or both streams, depending on source location.

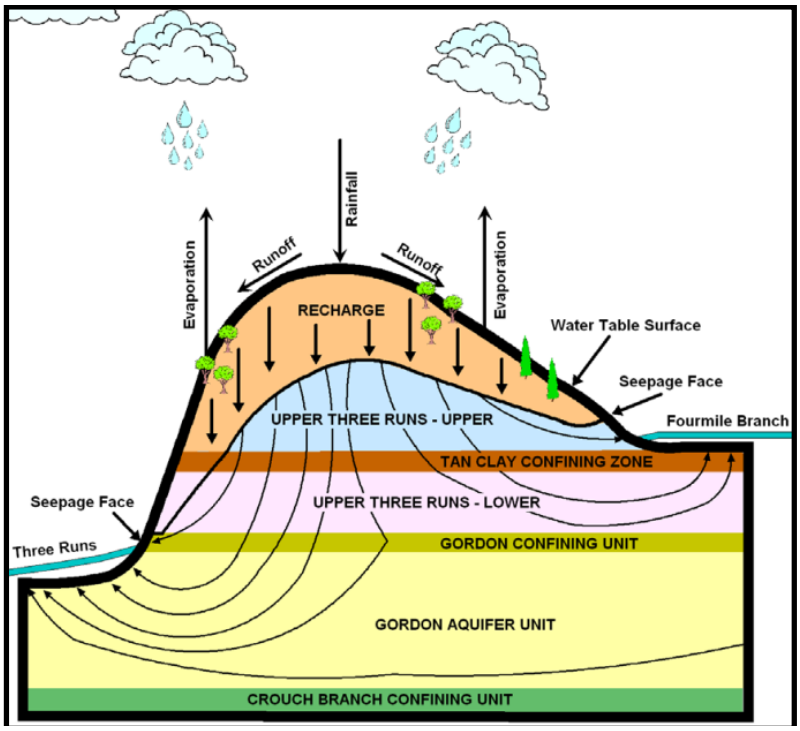
Figure 1-4 Chronostratigraphic, Lithostratigraphic, and Hydrostratigraphic Units at SRS



The vadose zone at the FTF is composed of the middle to late-Miocene-age “Upland Unit”—an informal name for the high elevation Upper Coastal Plain sediments of the Aiken Plateau in southwestern South Carolina. The lower surface of the “Upland Unit” is irregular due to erosion of the underlying formations. The “Upland Unit” represents a complex depositional environment having seen both fluvial deposition and occasional transitional marine influences. The resulting hydrogeological system is complex, ranging from very transmissive aquifers to their poorly transmissive confining units. The material composition of the “Upland Unit” is highly variable, consisting of coarse- to fine-grained sand, pebbly sand with lenses, and beds of sandy clay and clay. Vertically upward through the unit sorting of grains generally becomes poorer, clay beds generally become more abundant and thicker, and sands generally become more argillaceous and indurated. The vadose zone thickness between the UTR water table and the concrete slabs underlying the tanks is approximately 0.3 to 6.1 m (1 to 20 ft).

F-Area soils may contain as much as 20 to 40 percent clay. The predominant general soil association at the F-Area is the Fuquay–Blanton–Dothan soil. Soils in this industrial area are classified as Udorthents, which consist of a well-drained soil type that formed as the spoil or refuse from excavations and major construction operations. The major steps in constructing tank groupings consisted of digging an excavation and stockpiling the excavated soil, constructing the tanks, and backfilling around the tanks with the previously stockpiled soil. Udorthents are most commonly associated with well drained upland soils. A few small, poorly drained areas with spoil are also included. Clayey Udorthents have demonstrated good retention for most radionuclides.

Figure 1-5 Hydrogeological Conceptual Model Near FTF



1.1.5 Geology—Soft Zones, Seismology, and Volcanology

At the GSA, much of the sediment wedge consists of porous media that is not thought to support significant preferential flow. NRC staff noted observations of soft zones in a portion of the UTR-LZ aquifer within the McBean Member (shallow marine shoreface/shelf facies) of the Santee Formation due to carbonate dissolution. These soft zones were discovered in the 1950s by the U.S. Army Corps of Engineers, while drilling geotechnical boreholes to ascertain site-stability (U.S. Army Corps of Engineers, 1952). FTF Tanks 1–8 are located above soft zones in the UTR-LZ (WSRC-TR-2007-00283). Similar soft zones are located below Tanks 25–28 and 44–47 (WSRC-TR-2007-00283). Soft zones were also found in the subsurface west of Tanks 17 and 19 within borehole DH-5, southeast of Tank 33, within borehole FSEPB6, and near F Canyon within borehole FB1 (WSRC-TR-2007-00283). Soft zones found within exploratory boreholes beneath tank locations were filled with grout to provide waste tank foundation support. Dissolution of calcareous material (i.e., calcarenitic and shelly limestone, marl,

calcareous quartz sand, and sandy shell hash; Siple, 1967; Evans, 1995) by meteoric water under vadose conditions (DOE, 2011, RAI-FF-1 response) within the units that now comprise the UTR-LZ aquifer is thought to have caused localized ground subsidence and formed Carolina bays at the land surface, although not all soft zones at depth coincide with surficial bays (Siple, 1967). Dense, well-cemented, low hydraulic conductivity micritic limestone at the base of the McBean Member within much of the GSA is not of significant concern because it simply acts to strengthen the integrity of the underlying GCU (Evans, 1995). Soft zones composed of highly porous, high hydraulic conductivity shelly limestone and coarse-grained shell hash that occur (where present) in the center of the McBean Member (Evans, 1995) are of more concern. Where the soft zones are present in the UTR-LZ aquifer, they likely cause preferential flow that dominates the flow field (Evans, 1995).

Seismicity in the southeastern U.S. has been recorded for 300 years. Regionally, the record is dominated by the Charleston, South Carolina, earthquake on August 31, 1886. Pre-network, mostly qualitative seismic data were collected from 1698 to 1974, after which earthquakes began being instrumentally recorded. The most recent earthquake within an 80 km (50 mi) radius of the SRS was the magnitude 3.2, August 8, 1993, Couchton earthquake. This event did no damage to the SRS. Three recorded earthquakes with epicentral locations within the SRS have occurred dated: (i) June 9, 1985 (magnitude 2.6); (ii) August 5, 1988 (magnitude 2.0); and (iii) May 17, 1997 (magnitude 2.3). Strong motion accelerometers were not triggered as a result of these earthquakes. Tinker Creek fault, which is associated with activity of the Coastal Plain sediments, could subject F-Area to seismic activity. SRS is not located within a region of active plate tectonics characterized by volcanism, thus, volcanology is not an issue of concern.

1.2 FTF Closure Strategy

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. Therefore, DOE did not consider Tanks 17 and 20 to be within the scope of the draft FTF Section 3116 basis document. Nevertheless, DOE gained valuable experience from closing Tanks 17 and 20. The lessons learned from this experience are discussed in the draft basis document. Additionally, the PA for the FTF at the Savannah River Site (SRS-REG-2007-00002), takes into account radionuclide residuals from all FTF waste tanks, including Tanks 17 and 20.

Tank cleaning involves stepwise processes of bulk waste removal, sludge removal, and heel removal using a series of mechanical and chemical techniques. Tank access is gained through existing or newly introduced risers that allow access to internal tank surfaces. Bulk waste removal is accomplished largely through mechanical techniques including agitation and pumping. Heel and residual waste removal is accomplished through a combination of chemical and mechanical techniques. Residual waste in tank annuli (Type I, III, and IIIA tanks) will be removed using mechanical and wall crawler devices with washing capabilities. DOE will continue efforts to sample and characterize waste remaining in the tanks after cleaning. Further, they will continue and refine mapping techniques that allow accurate estimates of residual waste volume. Cooling coils in tanks will be flushed and grouted to reduce voids. Ancillary equipment will be isolated and removed as appropriate. Ancillary equipment left in place will be cleaned and some ancillary equipment will be stabilized. An engineered closure cap will be installed over the entire FTF area following the closure of the tanks and ancillary

equipment. While detailed design considerations will evolve with time, DOE now plans to install a cap with multiple layers to manage water infiltration and biointrusion.

The current Federal Facility Agreement (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. WSRC-OS-94-42, "Federal Facility Agreement for the Savannah River Site Administrative Document Number 89-05-FF Effective Date: August 16, 1993. DOE addresses the closure of the remaining FTF tanks (Type III and IIIA) and ancillary structures in the SRS Liquid Waste System Plan (SRR-LWP-2009-00001).

1.3 NRC Review Approach

NRC technical staff has conducted a detailed review of both Rev. 0 (DOE 2008) and Rev. 1 (DOE 2010) of the PA as well as DOE's draft basis document. The review has been supplemented by confirmatory computations of various results using analytical models (e.g., PORFLOW, Goldsim, HELP). NRC review process resulted in a number of requests for additional information (RAIs). Topics addressed in the RAIs are discussed elsewhere herein. RAIs were presented to DOE in December 2010. A series of technical exchanges by teleconference took place in January and February 2011. The teleconferences provided DOE an opportunity to better understand the purpose and intent of NRC's RAIs. DOE responded to NRC's RAIs in late May and June 2011. As part of the process for completing this TER, NRC staff again arranged a series of teleconferences to get a better understanding of DOE responses and proposed subsequent actions.

Prior to NRC's review of the draft basis document in support of DOE's Section 3116 determination for the FTF facility, the staff supported DOE in its development of the PA for the FTF. The purpose of the PA is to assess the performance of radioactive material over an extended period of time. In support of this effort, NRC, SCDHEC and EPA participated in several technical meetings, which were led by DOE, to ensure that all parties understood all underlying assumptions, parameter values and modeling results that DOE intended to use in its assessment. Summaries for these meetings can be found on DOE's website at http://www.em.doe.gov/stakepages/wmdi_swd.aspx?PAGEID=WMDI. As a result of these meetings, DOE issued a preliminary FTF PA for NRC, EPA and SCDHEC review and comment. DOE followed the premliminary FTF PA with a final draft of the FTF PA.

In July 2010, DOE, NRC, and representatives from SCDHEC and EPA participated in a public scoping meeting (see ADAMS ML102000163) in Aiken, South Carolina where members of the public were invited to provide their comments on the FTF WD. This meeting focused on the assumptions and analyses that would be used for the development of the draft FTF basis document (otherwise referred to as a waste determination or WD). Considering feedback received at this public scoping meeting, DOE drafted the FTF basis document for the Section 3116 determination, which can be found on DOE's webpage, cited above, along with the FTF WD meeting summary.

1.4 Prior Reviews

Prior to this review, NRC staff assessed several draft basis documents (otherwise referred to as waste determinations or WDs) by evaluating the methodology DOE used to determine if residual tank waste had been properly classified as incidental. The following are summaries illustrating NRC's involvement in these reviews:

SRS HLW Tank Closure: Classification Of Residual Waste As Incidental (1999)

In December 1999, after reviewing DOE's tank closure methodology to determine if residual waste (after cleaning) could be classified as incidental waste in HLW tanks at the Savannah River site, NRC staff determined that the waste could be considered incidental waste in keeping with the criteria approved by the Commission in SRM SECY 93-391 "Denial of PRM 60-4 – Petition for Rulemaking from the States of Washington and Oregon Regarding Classification of Radioactive Waste at Hanford", dated February 16, 1993. These criteria were communicated to DOE by letter dated March 2, 1993. DOE proposed the use of these criteria in its tank closure methodology to make its waste determination.

For this assessment, the staff determined that two of the three criteria for incidental waste classification of HLW tanks were satisfied. The requirements in Criterion 2, which call for waste in a solid form to not exceed the limits for Class C LLW, were not met. While many of the HLW waste tanks would meet this requirement through concentration averaging with water washing and bulk waste removal, there were still other tanks that would only meet the Criterion 2 requirements by a more stringent means such as oxalic acid cleaning to ensure that the residual waste did not exceed the Class C limitations. For this reason, DOE relied on an alternative classification consideration, in lieu of Criterion 2, that is similar to the classification considerations contained in 10 CFR 61.58 which acknowledges the acceptance of an alternate means of classifying waste if the Commission finds reasonable assurance that the waste still meets the performance objectives of 10 CFR Part 61. The staff concluded that this was an acceptable means of meeting the concentration limitations for Class C waste as set forth in 61.55 and encouraged DOE to develop a site-specific alternative to Criterion 2, notwithstanding the requirements set forth in Criterions 1 and 3.

The staff also reviewed the closure plan, regulatory basis, and grouting procedure that DOE used for Tanks 17 and 20 and determined that the filled tanks were expected to provide a stable waste form.

Technical Evaluation Report for Draft Waste Determination for Salt Waste Disposal (2005)

In support of the WD for salt waste disposal at SRS, NRC, in response to a request from DOE, conducted a review to determine the validity of the methodology used by DOE to determine that the tank waste was WIR. This methodology was used by DOE in its draft basis document for salt waste disposal in accordance with the requirements in the NDAA. "The draft waste determination addressed salt waste that DOE proposed to remove from the HLW tank farms, treat through various processes, and dispose of on site in the Saltstone Disposal Facility (SDF)" (see ML053010225, U.S. Nuclear Regulatory Commission Technical Evaluation Report for DOE's SRS Draft Section 3116 WD for Salt Waste Disposal, dated December 2005). NRC staff also evaluated how DOE would address salt waste that needed to be removed from the HLW

tank farms, treated through various processes, and disposed of on site in the Saltstone Disposal Facility. In this TER, the staff presented information on DOE's salt waste processing strategy, review criteria, and the agency's review approach, as well as NRC's analysis and conclusions with respect to whether there is reasonable assurance that DOE's proposed approach can meet certain requirements of the NDAA for determining that the tank waste is not HLW.

Based on the information provided in the draft waste determination, NRC staff concluded that there is reasonable assurance that the applicable criteria of the NDAA can be met, provided certain assumptions made in DOE's analyses are verified via monitoring. NRC staff identified factors important to assessing compliance with 10 CFR 61, Subpart C, including improvements in future modeling and the associated support to justify the predictions of the modeling. The staff also emphasized to DOE the importance of verifying that the assumptions made in assessing whether the performance objectives of 10 CFR 61, Subpart C, could be met. The staff's concern was that some of the assumptions made in the analysis, if incorrect, could lead to noncompliance with the performance objectives. NRC staff went on to share its plan to observe actions taken by DOE to fulfill its requirements as part of its responsibilities under the NDAA, along with the State of South Carolina.

At this time, NRC staff is also in the process of reviewing the revised saltstone PA, which was issued in 2009. The staff anticipates completion of this review and issuance of the associated technical evaluation report in November 2011.

NRC Staff Review of the PA for the FTF at the Savannah River Site (2009)

In December 2005, DOE submitted the *Draft Section 3116 Determination for Closure of Tank 19 and Tank 18 at the Savannah River Site* for NRC review per the requirements in the NDAA. As a result of the staff's review, several issues were identified in the PA, which prompted the staff to seek clarification from DOE through the issuance of RAIs (see ADAMS ML060800295, Letter from Flanders to Gilbertson, dated March 31, 2006).

Two years later, DOE requested that NRC stop its review and consultation on the *Draft Section 3116 Determination for Closure of Tank 19 and Tank 18 at the Savannah River Site* that was submitted for review in 2005 (see ADAMS ML080090405, Letter from Marcinowski to Camper, dated December 20, 2007). DOE cited its decision to pursue a promising new technology for potentially removing additional waste from the tanks as its reason for requesting that NRC not move forward with its review at that time. Additionally, DOE went on to inform NRC that it would develop a new PA for the FTF prior to closure of Tanks 18 and 19 at SRS. In this letter, DOE also notified NRC that while it would not formally respond to the 50 open NRC RAIs that were submitted in March of 2006; it would, however, consider the RAIs in developing the new basis document, as appropriate.

2. CRITERION 1

The waste does not require permanent isolation in a deep geologic repository for spent fuel or HLW (NDAA).

2.1 Tank Waste Disposal

Criterion 1 allows High Level Waste (HLW) to be disposed of in accordance with its risk and not solely on the basis of its origin.

Criterion 1 acknowledges that waste may require disposal in a geologic repository even though the two other criteria of the NDAA may be met. Consideration could be given to those circumstances under which geologic disposal is warranted to protect public health and safety and the environment. For example, unique radiological characteristics of waste or non-proliferation concerns for particular types of materials might require disposal in a geologic repository.

2.2 NRC Evaluation (Criterion 1)

Provided that DOE can meet the remaining NDAA criteria (Criteria 2 and 3), there is no indication that other considerations would warrant disposal of the waste in geologic repository because there appears to be no special properties of the waste and there are no proliferation concerns. The NRC staff believes that DOE can meet Criterion 1.

3. CRITERION 2

The waste has had HRRs removed to the MEP.

NRC staff evaluated this criterion by analyzing DOE's: (i) methodology for developing radionuclide inventories for the tanks, and auxiliary equipment; (ii) process for identifying HRRs; (iii) selection of waste treatment technology; and (iv) demonstration of removal to the MEP, including analysis of the costs and benefits of additional radionuclide removal. For the purpose of reviewing DOE waste determinations, NRC guidance defines highly radioactice radionuclides (HRRs) as those that contribute most significantly to risk to the public, workers, and the environment (NRC, 2007).

3.1 Waste Inventory

This section describes DOE's approach to derive inventories for the tanks (including tank walls and in-tank components) and ancillary equipment. The inventory in the tanks: (i) must be developed to demonstrate that the waste has been processed to remove HRRs to the MEP (see Section 3.7), (ii) determines whether the waste is greater than Class C (Section 4.1), and (iii) is used to develop the source term in the PA (see Section 4.2.8.3). DOE's approach includes the use of Waste Characterization System (WCS) data, sampling results, and analytical methods (e.g., continuously stirred tank reactor model for residual remaining in transfer lines after flushing). This section briefly discusses uncertainty with respect to inventory estimates (e.g., uncertainty due to limited sampling of variable waste streams, measurement error, use of process knowledge, volume estimates, and density measurements).

Two types of inventories for the tanks are developed for use in the FTF PA including: (i) projected inventories for all tanks, including those that have not yet been cleaned, and (ii) final inventories for tanks that have been cleaned and sampled. For those tanks that have not been cleaned, this section describes DOE's basis for assumptions regarding the residual volume of waste expected to remain in the tanks following waste retrieval operations, as well as information about any available sampling data. This section details the approach used to estimate final inventories for tanks that have been cleaned (e.g., Tanks 18 and 19) including information about development of a sampling and analysis plan, methods used to determine residual waste volumes, and consideration of uncertainty in inventory estimates.

3.1.1 Projected Inventories for Tanks

DOE attempted to develop inventories that were both bounding and reasonable for waste tanks following operational closure. For those components projected to have insignificant impact on dose, DOE characterized its calculated inventories as considerably conservative, while inventories for components expected to more significantly affect dose were considered by DOE to be more reasonable.

An initial screening approach (described in CBU-PIT-2005-00228) narrowed the list of radionuclides to support characterization efforts and development of an initial inventory for use in the FTF PA modeling. This approach resulted in the screening of an initial list of 849 radioisotopes down to 159 using information on the physical properties of each radioisotope such as half-life and decay mechanism, waste production and processing information, and screening factors for ground disposal of radionuclides developed in NCRP-123. Appendix A of SRR-CWDA-2009-00045 summarizes the screening approach.

DOE used the following steps to develop an initial list of radionuclides:

1. DOE included isotopes from the four decay series: actinium, neptunium, thorium, and uranium, because FTF waste contains the first member of each series.

2. DOE included other isotopes known to be in HLW sludge.

The following screening criteria were then used to eliminate radionuclides from the initial list:

3. DOE eliminated isotopes with low risk for which there is no dose conversion factor.

4. DOE eliminated isotopes based on the NCRP-123 screening methodology for ground disposal.

5. DOE eliminated those isotopes that would result in a hypothetical exposure to a member of the public of 0.04 mSv/yr (4 mrem/yr) or less (including contributions of daughter products), assuming a large inventory of 3.7×10^{15} Bq (1.0×10^{6} Ci) and using the screening factors in NCRP-123.

6. DOE eliminated those isotopes that would result in a hypothetical exposure to a member of the public of 0.04 mSv.yr (4 mrem/yr) or less (including the contributions of daughter

products), assuming a large mass of 454.2 kg (1,000 lbs) and using the screening factors in NCRP-123.

7. DOE eliminated radionuclides that, due to their physical properties, are not be expected to be found in the waste (e.g., waste that is present as a gas and released in the reactor during fuel processing).

8. DOE eliminated radionuclides that would not be present at the time of closure (or institutional control period) based on the age of the waste and time to closure. This analysis considered decay and in-growth to determine whether a radionuclide should be eliminated. For those radionuclides that decayed rapidly to a long lived radionuclide, only the long-lived radionuclide is listed and tracked.

As stated above, 159 radionuclides remained following execution of the screening steps listed above. Of the list of 159 isotopes following this initial screening, DOE eliminated another 89 isotopes from the initial inventory list because of their decay behavior (e.g., due to rapid in-growth following the assumed closure or institutional control period or due to limited in-growth after the closure or institutional control period relative to the compliance period of 10,000 years). Table 4.2-6 of the revised PA lists the reasons for eliminating additional radionuclides from the initial inventory list.

Following the second screening, 70 radionuclides remained. Four additional radionuclides (Cf-251, Cf-252, Ra-228, and Th-232) were removed based on special analysis (SA) that indicated they were not present in FTF waste. DOE eliminated 12 additional radionuclides from the list of radionuclides because of their short half-lives of less than 5 years. These 12 radionuclides did not warrant separate transport modeling given their short half-lives, although their contributions to dose might be considered if they are daughter products of radionuclides that remained on the initial radionuclide inventory list for PA modeling. The final list of 54 radionuclides, along with the 18 non-radiological contaminants is reproduced in Table 3-1.

Table 3-1 Contaminants with an Initial Inventory (Including Non-Radiological Contaminants)

Ac-227	Cl-36	Eu-154	Ni-63	Pu-242	Th-229
Ag	Cm-243	F	NO2	Pu-244	Th-230
Al-26	Cm-244	Fe	NO3	Ra-226	U
Am-241	Cm-245	H-3	Np-237	Sb	U-232
Am-242m	Cm-247	Hg	Pa-231	Sb-126	U-233
Am-243	Cm-248	I-129	Pb	Sb-126m	U-234
As	Co-60	K-40	Pd-107	Se	U-235
Ba	Cr	Mn	Pt-193	Se-79	U-236
Ba-137m	Cs-135	Nb-93m	Pu-238	Sm-151	U-238
C-14	Cs-137	Nb-94	Pu-239	Sn-126	Y-90
Cd	Cu	Ni	Pu-240	Sr-90	Zn
Cf-249	Eu-152	Ni-59	Pu-241	Tc-99	Zr-93

DOE developed the initial list of radionuclides using the screening approach described above which took into account activity, half-life, and other considerations. DOE then used a step-by-step approach to developing tank inventories for use in performing modeling for the Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) starting with the inventory derived from Rev. 0 (SRS-REG-2007-00002, Rev. 0). In most cases, adjustments to the inventory based on tank type or radionuclide were made.

The steps for developing tank inventories included the following:

1. Inventory Starting Point: DOE used the radionuclide inventories developed for the Rev. 0 FTF PA (SRS-REG-2007-00002, Rev. 0) as a starting point. The Waste Characterization System or WCS (see Section 3.3.2.3 of the revised PA) was used to develop the Rev. 0 inventories. WCS is an electronic information system that tracks waste tank data, including projected radionuclides and non-radionuclide inventories, based on samples, analyses, process history, composition studies, and theoretical relationships. The WCS tracked 36 radionuclides, and the initial inventories for those radionuclides not historically tracked within the WCS were determined by special analyses including fission yield ratios and decay equations. DOE assumed that approximately 0.2 cm (0.06 in) of residuals would remain for tanks that have not yet been cleaned. This volume of residue was multiplied by the concentrations in the WCS to obtain an inventory quantity. DOE estimated the actual inventory in tanks that had been cleaned (i.e., Type IV tanks) based on sampling and characterization data.

2. Inventory Adjustments: The waste tanks were binned according to tank type (Type I, III/IIIA, and IV). Within each bin, the inventories were adjusted as follows:

    a. For Type I and IIIA tanks, the maximum *concentration* associated with an individual waste tank bin was applied to the other tanks within the bin to account for uncertainty surrounding future operations and movement of material within the FTF.

    b. For Type I and IIIA tanks, because of uncertainties related to future waste removal (e.g., unknowns about the effectiveness of tank cleaning technologies), the individual waste tank inventories were increased by one order of magnitude (e.g., the maximum concentration for the waste tank bin from “a” above or the 0.2 cm (0.06 in) of residual waste assumed to remain in the tanks following waste retrieval or a combination of both was assumed to be an order of magnitude higher). For Type I and IIIA tanks, DOE assumed that the inventory uncertainty fits a uniform distribution, with the adjusted reasonably bounding estimate as the maximum and the minimum set two orders of magnitude below the estimate.

    c. DOE made some small adjustments to Type III tanks, Tanks 33 and 34. However, these adjustments were not as large as those for Type I or Type IIIA tanks because DOE assumes that the cooling coils will be removed and enable cleaning the tanks with oxalic acid to 0.2 cm (0.06 in) of residual waste.

    d. For Type IV tanks, DOE revised Tank 18 and 19 inventories to account for an increased level of uncertainty surrounding the residual inventories remaining after waste removal. Specifically, based on preliminary sample results, DOE increased the estimates for Am-

243, Cs-135 and Np-237 for Tank 18 only. For Tanks 18 and 19, DOE assumed that the inventory uncertainty fits a uniform distribution from half of the Rev. 1 estimate up to twice that estimate. Type IV tanks, Tanks 17 and 20, were left unchanged from the Rev. 0 (SRS-REG-2007-00002, Rev. 0) estimates, which were based on measurements of residual solids developed during tank closure activities. DOE assumed that the inventory uncertainty for Tanks 17 and 20 fits a normal distribution, with the estimate set as the mean and the standard deviation set at 0.5.

3. Inventories for each specific tank are listed in DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1). For all tank types, for those radionuclides that were detected, but had an individual waste tank inventory less than 3.7×10^{10} Bq (1 Ci), DOE typically adjusted the inventory up to 3.7×10^{10} Bq (1 Ci) if not found to be risk-significant. Note that for other risk-significant radionuclides that were not detected, DOE set the inventories to the minimum detection limit of 3.7×10^{7} Bq (1.0×10^{-3} Ci).

4. DOE decreased the inventories of Ra-226 and Th-230 inventories from those in Rev. 0 (SRS-REG-2007-00002, Rev. 0) to better reflect the age of the waste (i.e., adjusted downward to the detection limit of 3.7×10^{7} Bq [1.0×10^{-3} Ci]).

SRR-CWDA-2009-00045 provides additional detail on the approach used to develop the inventory. Table 3.3-2 of Rev. 1of the PA lists the inventory on the assumed date of closure (September 30, 2020) for each FTF tank (SRS-REG-2007-00002, Rev. 1). Table 3–2 summarizes the initial estimated inventory of the tanks. Inventories for Tank 1 and Tank 25 are provided for Type I tanks and Type IIIA tanks, respectively, instead of listing each tank individually because the inventories among the tanks within these tank types are identical for the large majority of radionuclides. Note that for certain radionuclides, the inventories differ for tanks even within the same tank type. The Rev. 1 PA lists inventories for each specific tank (SRS-REG-2007-00002, Rev. 1).

Table 3-2 FTF Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) Estimated Radionuclide Inventory Decayed to 9/30/2020 (Ci)

Radionuclide	Tank 1 (I)	Tank 17 (IV)	Tank 18 (IV)	Tank 19 (IV)	Tank 20 (IV)	Tank 25 (IIIA)	Tank 33 (III)	Tank 34 (III)
Ac227	$1.0x10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Al26	$1.0x10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$
Am241	$6.0x10^{2}$	$8.4x10^{0}$	$8.2X10^{1}$	$2.3X10^{0}$	$1.6X10^{0}$	$1.0X10^{0}$	$6.3X10^{1}$	$1.6X10^{3}$
Am242m	$1.0x10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$
Am243	$1.4x10^{0}$	NE	$1.0X10^{-1}$	$1.0X10^{-1}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Ba137m	$8.7X10^{3}$	$1.0x10^{1}$	$9.1X10^{3}$	$6.2X10^{3}$	$2.3X10^{1}$	$4.9X10^{3}$	$9.0X10^{2}$	$3.7X10^{3}$
C14	$1.0X10^{0}$	$3.1x10^{-3}$	$1.0X10^{0}$	$1.0X10^{0}$	$6.6X10^{-4}$	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$
Cf249	$1.0X10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$
Cl36	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Cm243	$1.0X10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$
Cm244	$1.2X10^{2}$	$2.4x10^{-4}$	$1.0X10^{2}$	$1.0X10^{0}$	NE	$1.0X10^{-3}$	$1.0X10^{0}$	$1.0X10^{0}$
Cm245	$1.0X10^{0}$	$4.4x10^{-10}$	$1.0X10^{0}$	$1.0X10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$

Cm247	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Cm248	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Co60	$1.8X10^{1}$	$3.4x10^{-2}$	$1.0X10^{0}$	$1.0X10^{0}$	$4.8X10^{-3}$	$1.0X10^{0}$	$1.7X10^{1}$	$4.7X10^{1}$
Cs135	$1.0X10^{0}$	$1.7x10^{-4}$	$1.0X10^{0}$	$1.0X10^{0}$	$3.6X10^{-5}$	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$
Cs137	$9.2X10^{3}$	$1.1x10^{1}$	$9.7X10^{3}$	$6.5X10^{3}$	$2.4X10^{1}$	$5.2X10^{3}$	$9.5X10^{2}$	$3.9X10^{3}$
Eu152	$1.9X10^{1}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	NE	$1.0X10^{0}$	$3.5X10^{0}$	$1.3X10^{1}$
Eu154	$1.3X10^{2}$	2.3x10-^{2}	$3.2X10^{0}$	$1.0X10^{0}$	$1.6X10^{-1}$	$2.9X10^{0}$	$4.3X10^{1}$	$1.4X10^{2}$
H3	$1.0X10^{0}$	$5.8x10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$
I129	$1.0X10^{-3}$	$1.3x10^{-6}$	$1.0X10^{-3}$	$1.0X10^{-3}$	$2.6X10^{-7}$	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
K40	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Nb93m	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Nb94	$1.0X10^{0}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{0}$	$1.0X10^{0}$
Ni59	$6.3X10^{0}$	$1.8x10^{-1}$	$1.0X10^{0}$	$1.0X10^{0}$	$3.9X10^{-2}$	$1.0X10^{0}$	$1.0X10^{0}$	$1.8X10^{0}$
Ni63	$4.9X10^{2}$	NE	$8.2X10^{1}$	$1.4X10^{1}$	NE	$2.4X10^{1}$	$3.8X10^{1}$	$1.6X10^{2}$
Np237	$2.3X10^{-1}$	$1.4x10^{-2}$	$2.4X10^{-1}$	$2.2X10^{-3}$	$7.2X10^{-4}$	$1.0X10^{-3}$	$2.5X10^{-2}$	$6.8X10^{-2}$
Pa231	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Pd107	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Pt193	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{0}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Pu238	$1.4X10^{2}$	$5.4x10^{1}$	$7.0X10^{1}$	$4.4X10^{0}$	$6.1X10^{0}$	$1.2X10^{2}$	$3.6X10^{1}$	$1.0X10^{0}$
Pu239	$3.2X10^{1}$	$1.5x10^{1}$	$1.6X10^{2}$	$6.4X10^{0}$	$8.5X10^{-1}$	$2.2X10^{1}$	$2.2X10^{1}$	$1.4X10^{1}$
Pu240	$7.2X10^{0}$	$3.4x10^{0}$	$4.9X10^{1}$	$2.3X10^{0}$	$1.8X10^{-1}$	$4.8X10^{0}$	$3.9X10^{0}$	$3.2X10^{0}$
Pu241	$3.2X10^{1}$	$9.3x10^{1}$	$1.3X10^{2}$	$4.6X10^{0}$	$1.6X10^{1}$	$5.4X10^{1}$	$5.5X10^{1}$	$3.1X10^{1}$
Pu242	$1.0X10^{0}$	$5.3x10^{-3}$	$1.0X10^{0}$	$1.0X10^{0}$	$1.6X10^{-3}$	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$
Pu244	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Ra226	$1.0X10^{-3}$	NE	$1.9X10^{-3}$	$1.1X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
Sb126	$9.4X10^{-2}$	$4.0x10^{-3}$	$2.3X10^{-2}$	$3.6X10^{-3}$	$8.3X10^{-4}$	$3.3X10^{-4}$	$7.7X10^{-2}$	$3.4X10^{-1}$
Sb126m	$6.7X10^{-1}$	$2.8x10^{-2}$	$1.6X10^{-1}$	$2.6X10^{-2}$	$5.9X10^{-3}$	$2.4X10^{-3}$	$5.5X10^{-1}$	$2.4X10^{0}$
Se79	$4.5X10^{0}$	$1.6x10^{-2}$	$1.0X10^{0}$	$1.0X10^{0}$	$3.2X10^{-3}$	$1.0X10^{0}$	$1.0X10^{0}$	$1.3X10^{0}$
Sm151	$1.2X10^{4}$	NE	$4.6X10^{1}$	1.0X100	NE	$7.1X10^{1}$	$9.3X10^{2}$	$4.0X10^{3}$
Sn126	$6.7X10^{-1}$	$2.8x10^{-2}$	$1.6X10^{-1}$	$2.6X10^{-2}$	$5.9X10^{-3}$	$2.4X10^{-3}$	$5.5X10^{-1}$	$2.4X10^{0}$
Sr90	$1.3X10^{5}$	$6.6x10^{1}$	$1.1X10^{3}$	$5.2X10^{0}$	2.3X101	$1.0X10^{3}$	$1.4X10^{4}$	$5.5X10^{4}$
Tc99	$7.9X10^{1}$	$9.0x10^{-1}$	$1.0X10^{0}$	$1.4X10^{0}$	$8.5X10^{-1}$	$1.0X10^{0}$	$5.1X10^{0}$	$2.2X10^{1}$
Th229	$2.4X10^{-1}$	NE	$2.6X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$2.6X10^{-2}$	$7.1X10^{-2}$
Th230	$1.0X10^{-3}$	NE	$1.9X10^{-3}$	$1.1X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$
U232	$1.0X10^{0}$	$3.7x10^{-5}$	$1.0X10^{0}$	$1.0X10^{0}$	$8.0X10^{-6}$	$1.0X10^{0}$	$1.0X10^{0}$	$1.0X10^{0}$
U233	$2.3X10^{-1}$	NE	$1.1X10^{0}$	$1.9X10^{-1}$	NE	$1.0X10^{-3}$	$2.5X10^{-2}$	$6.8X10^{-2}$
U234	$1.7X10^{-1}$	NE	$3.8X10^{-1}$	$1.1X10^{-2}$	NE	$2.6X10^{-2}$	$7.9X10^{-2}$	$8.8X10^{-2}$
U235	$5.8X10^{-3}$	$3.0x10^{-4}$	$8.4X10^{-3}$	$2.6X10^{-4}$	$1.9X10^{-5}$	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.2X10^{-3}$
U236	$1.0X10^{0}$	NE	$1.0X10^{0}$	$1.0X10^{0}$	$2.7X10^{-5}$	$1.0X10^{0}$	$1.0X10^{0}$	1.0X100
U238	$1.7X10^{-1}$	$6.4x10^{-3}$	$2.2X10^{-1}$	$8.7X10^{-3}$	$5.6X10^{-4}$	$2.6X10^{-2}$	$7.9X10^{-2}$	$8.8X10^{-2}$
Y90	$1.3X10^{5}$	$6.6x10^{1}$	$1.1X10^{3}$	5.2X100	$2.3X10^{1}$	$1.0X10^{3}$	$1.4X10^{4}$	$5.5X10^{4}$
Zr93	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	NE	$1.0X10^{-3}$	$1.0X10^{-3}$	$1.0X10^{-3}$

NE=the inventory for radionuclide was not estimated

After each tank is cleaned, the amount of residual material remaining in the tanks will be determined using visual inspection techniques to estimate volumes and sampling to estimate concentrations as already has been done with Tanks 18 and 19. The inventory in residual

liquids after cleaning is expected to be negligible based on experience with mechanical cleaning of Tanks 17 and 20.

No separate inventory is estimated for the annulus, failed cooling coils, tank walls, cooling coils and columns as this inventory is expected to be insignificant compared to the tank floor inventory. Only Tanks 1, 5, and 6 contain annulus contamination in small volumes in the form of dried salts that are expected to be removed during preparation for closure. Cooling coils represent a small percentage of the in-tank volume and will be flushed to minimize residual waste holdup. In-tank surfaces are not expected to contain significant deposits based on sludge mapping inspections of Tanks 5 and 6 performed to date. Cooling coils, as well as tank walls and support columns, will be rinsed with oxalic acid, with the exception of Tanks 18 and 19. Tanks 18 and 19 contained significant quantities of zeolite. Oxalic acid is expected to be effective at cleaning internal surfaces, but does not react well with zeolite. Therefore, the radiological inventory associated with corrosion products on the walls of Tanks 18 and 19 was estimated and is included in the radiological inventory for these two waste tanks (SRR-CWDA-2009-00045).

3.1.2 Tanks 18 and 19 Final Characterization

3.1.2.1 Sampling and Analysis of Tank Floors

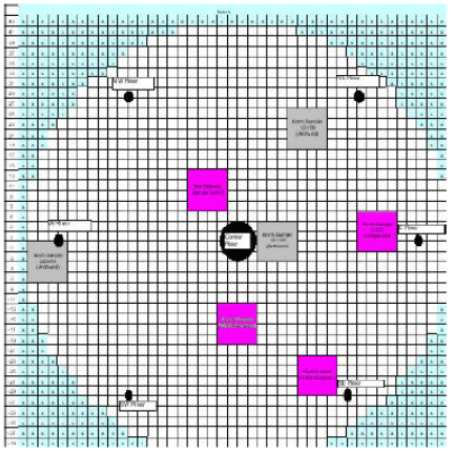
It is important to note that the inventory estimates in the Rev. 0 (SRS-REG-2007-00002, Rev. 0) and Rev. 1 PAs (SRS-REG-2007-00002, Rev. 1) for Tanks 18 and 19, are different than the final inventory estimates provided in response to NRC comment (DOE, 2011), as are the uncertainty estimates. The final inventory for Tanks 18 and 19 are used for a SA to evaluate the impact on the PA results of the revised inventory estimates. Comparisons between the Rev. 1 and SA inventories are provided below, as is a discussion on the overall approach to developing the final inventories for Tanks 18 and 19.

In 2009, DOE sampled Tanks 18 and 19 according to a plan that had been developed during the final clean-up process that used the Mantis rover to mobilize material and transfer it to the Waste Mix Chamber in Tank 7 where the waste was ground and collected. The original sampling plan called for two in-process samples from each of four tank quadrants: northwest, northeast, southwest, and southeast. Three samples from each quadrant were collected and sent to be analyzed. However, this sampling did not produce sufficient quantities to support analytical determinations in all four quadrants of Tanks 18 and 19. Therefore, DOE revised the plan to separately characterize material in two hemispheres (north and south regions) of Tanks 18 and 19 (SRNL-STI-2009-00782, SRNL-STI-2009-00779). Figure 3-1 shows the 2009 Mantis rollover locations for Tank 18 and Tank 19. Tank 18 had 7 rollover samples sites and Tank 19 had 4 sites.

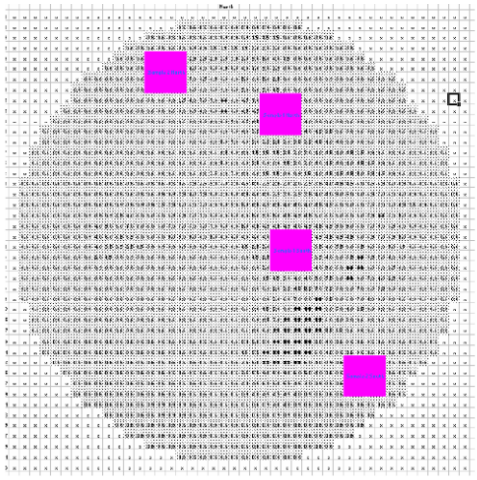
Along with the in-process Mantis samples, a floor scrape sample was collected for quality assurance in 2009 from the North region in Tank 18 and from the South region in Tank 19, which was shown to be different based on analytical uncertainty. No floor scrape samples were taken for the south region in Tank 18 or from the north region in Tank 19 in 2009 (SRNL-STI-2009-00782, SRNL-STI-2009-00779).

Figure 3-1 2009 Sample Locations for Tanks 18 and 19

Tank 18 (SRNL-STI-2009-00782)



Tank 19 SRNL-STI-2009-00779



Note: Squares denote sampling locations.

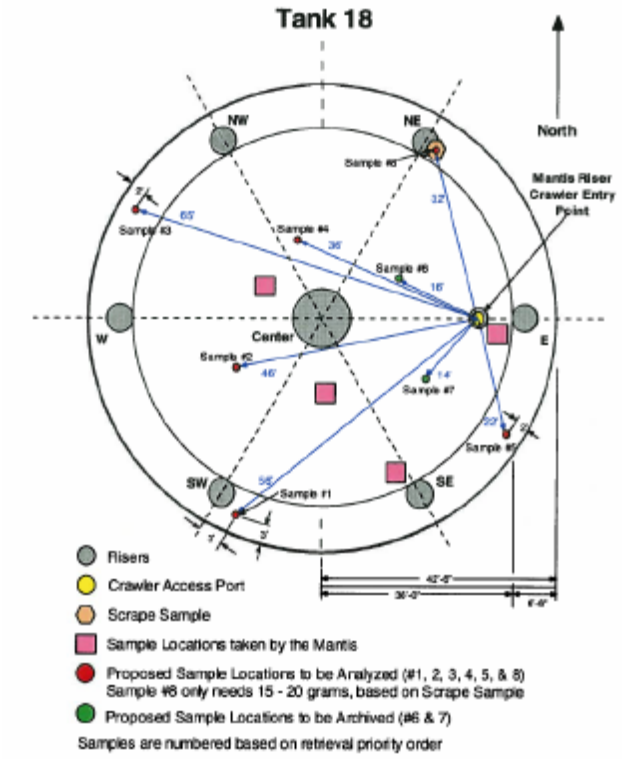
All of the samples from 2009 are classified as being from either the north region or the south region. The quantity of the analytes in either region is calculated separately. The average concentration is calculated for each hemisphere, and for the entire tank, and the 95 percent Upper Confidence Limit (UCL) is also calculated. Based on these sample analyses, waste tank floor residual material in Tank 18 is assumed to include two distinct regions, the southwest area of the waste tank and the remaining waste tank area. Additional floor scrape samples were taken to reduce the uncertainty in the results. The number of additional floor scrape samples was determined based on the reduction in uncertainty that would have resulted from each additional sample. When an additional sample has insignificant impact on the uncertainty, no more samples are needed. This method resulted in obtaining eight additional samples from each tank, six of which were analyzed and two of which were archived.

The location of the samples was based on partitioning the tank into an inner ring and outer ring and 60 degree segments. The goal was to have a total of four samples from each hemisphere, with two samples in each ring per hemisphere. Therefore, in 2010, eight additional floor scrape samples were collected from specific locations within each tank with a radiation hardened robot. Two additional samples were collected and archived (but not analyzed). Figure 3-2 shows the location of samples for Tanks 18 and 19.

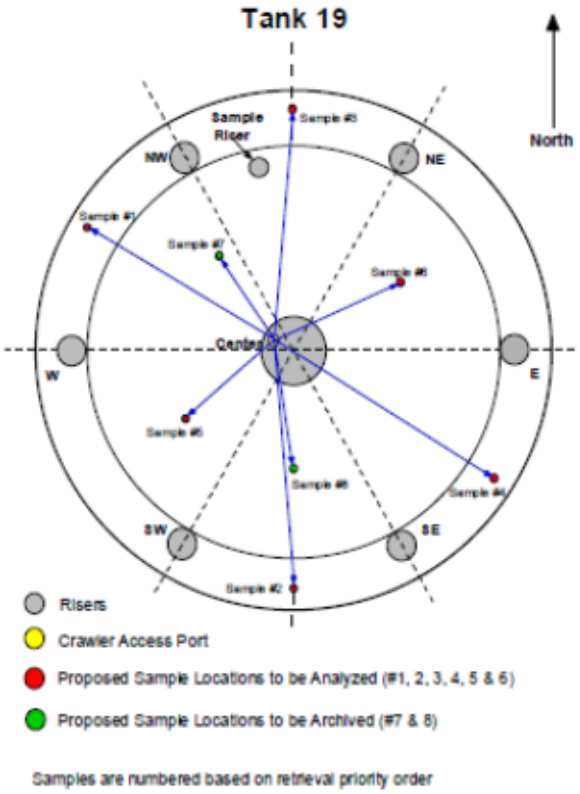
Each sample was divided into three aliquots and characterized. DOE provided references SRNL-STI-2010-00386, SRNL-STI-2010-00439, which detail the analytical procedures, and quality assurance and control measures used for analyzing the floor samples. DOE found it challenging to meet the detection limits of 1.85 Bq/g (5.0×10^{-11} Ci/g) for certain radionuclides (I-129, Ra-226, Ac-227 and Pa-231). In addition, analysis of Cl-36, K-40, Pd-107, Pt-193 and Al-26, Zr-93, Nb-94), required new or modified analytical methods to attain the detection limits (SRNL-STI-2010-00386). Statistical analysis of the 2010 sample results showed the material from each region (north and south) to be similar enough to be characterized as one population; DOE found that all samples fell within three standard deviations of the mean (SRNL-STI-2010-00401 and SRNL-STI-2010-00400).

Figure 3-2 Tanks 18 and 19 2010 Floor Sample Locations

Tank 18 (SRNL-STI-2010-00401)



Tank 19 (SRNL-STI-2010-00400)



3.1.2.2 Sampling and Analysis of Tank Walls

SRNL-STI-2009-00802, “Characterization of Tank 18F Wall and Scale Samples” (March 2010) and SRNL-STI-2009-00799 “Characterization of Tank 19F Wall and Scale Samples” (March 2010) provide characterization details for the wall samples. The walls of the Tank 18 were determined to have two classes of residual material: (i) material build-up, called “scale,” adhered to the wall in specific locations; and (ii) general corrosion film assumed to cover the entire wall surface. Two samples of corrosion were taken (one from the upper wall and one from the lower wall), as well as one sample of scale. Only a subset of radionuclides was analyzed in the wall samples because of analytical limitations measuring very low levels. The scale samples proved to be similar in concentration to the floor samples, so the floor sample statistics were used in conjunction with the wall volume estimates to estimate the wall inventory. Because there were only two corrosion samples, the 95th percentile was very large. Therefore the Reasonably Conservative Estimate was set to be the same as the Best Estimate. Because only a subset of radionuclides was analyzed, the remaining radionuclide concentrations were determined using a surrogate ratio to U-238 in the floor samples.

3.1.2.3 Volume on Tank Floors and Walls

DOE determined the residual volume on the tank floors through material mapping, which uses photographic images to estimate the relative depth of waste in relation to known objects within the tank (e.g., lifting plates and Mantis wheels). The floor of a Type IV waste tank has 69, 0.3 m (1 ft) by 0.3 m (1 ft) by 1.3 cm (0.5 in) lifting plates arrayed in a grid pattern across the floor. Each plate has a 0.6 cm (0.5 in) weld bead affixing it to the floor and the remnant of where the lifting rod attached to the top of the plate during tank construction. These lifting plates were used as landmarks for material mapping). In some areas where there were no lifting plates, or they could not be seen, the Mantis wheels were used as a relative object in the photograph for comparison.

Following cleaning, DOE took more than 140 and 175 high definition photographs of Tanks 18 and 19, respectively. However, the Mantis, which was used as a reference object to determine residual waste thickness, was unable to reach certain areas of the tanks because the wheels lost traction as a result of material build-up inside the Mantis' wheels and as a result of the drag weight of hose and tether system. Because of lack of relative objects for comparison and the inability of the Mantis to reach the western regions of Tank 18, DOE approximated the volume in the western region of Tank 18 using a mean of other areas that were visually similar. This approach was also applied for areas in Tank 19 where no relative objects were available. From visual inspection of photographs from Tank 18, DOE determined the volume of material on the stiffener bands and on the tank walls to be 0.03 m^3 (8 gal) and 0.42 m^3 (110 gal) respectively. DOE did not detect discernable material on the stiffening bands or walls of Tank 19. In total, DOE estimated that 15.2 m^3 (4,000 gal) remained in Tank 18 and 7.6 m^3 l (2,000 gal) in Tank 19 (see U-ESR-F-00041 and U-ESR-F-00042).

To determine the uncertainty in the residual volume, DOE mapped 10 areas scattered across each tank and predicted a best estimate, a low value and a high value for each area. Although the same team re-mapped these 10 areas using the same approach, the team did not have the benefit of the original estimate. The team was asked to provide a low, best, and high estimate. The best estimate total for these 10 areas was within 3 percent of the original mapping volume for the same areas. The high volume was 35 percent more than the original volume and the low volume was 30 percent less than the original volume (U-ESR-F-00041, U-ESR-F-00042). Therefore, a Reasonably Conservative bounding value for the residual volume was set at 35 percent greater than the Best Estimate volume, or 20.1 m^3 (5,300 gal) for Tank 18 and 10.3 m^3 (2,700 gal) for Tank 19 (SRR-CWDA-2010-00117 and SRR-CWDA-2010-00118).

DOE determined the final inventories of the tanks by multiplying the concentration µCi/g in the samples by the volume estimate, and assuming a density of 1.1 g/mL. Uncertainty in the inventories was considered by using the 95^{th} percentile UCL as a best estimate for those constituents for which sample analyses provide measured results. DOE also estimated a reasonably conservative upper bounding inventory by adding two times the standard deviation to the average. In cases where the 95^{th} percentile UCL was greater than twice the standard deviation added to the mean, the Reasonably Conservative estimate was also assigned the 95^{th} percentile UCL (SRR-CWDA-2010-00117 and SRR-CWDA-2010-00118).

For those radionuclides that were not detected (i.e., the results only returned minimum detection limits), the lowest detection limit was used as a Best Estimate and the highest detection limit was used as the reasonably conservative estimate with the exception of Cl-36 and Ra-226, where the lower detection limit is applied for both the best estimate and the reasonably conservative estimate. The final inventories are provided in Table 3-3 in Section 3.2 of this TER.

3.1.3 Ancillary Equipment Inventory

The purpose of the ancillary equipment is to transfer waste and reduce waste volume through evaporation. The relative amount of residual radioactivity within these components is related to its service life, materials of construction, and radioactive material that came in contact with the components.

Ancillary equipment includes: (i) transfer lines; (ii) transfer line secondary containment, (iii) diversion boxes, (iv) valve boxes; (v) pump tanks, (v) pump pits; (vi) the catch tank; and (vii) the evaporator systems. The following provides a brief description of the equipment:

- The FTF has more than 13,716 m (45,000 ft) of transfer line, typically made of stainless steel and located below ground.
- A secondary pipe (transfer line secondary equipment) constructed of carbon steel, stainless steel or cement-asbestos surrounds most of the primary lines.
- Diversion boxes are shielded reinforced concrete structures containing transfer line nozzles that direct waste to a certain location. Most diversion boxes are located below grade and are lined with stainless steel or sealed with waterproofing compounds.
- Transfer valve boxes facilitate waste transfers, and are usually manual ball valves in removable jumpers with flush water connections on transfer lines.
- The three FTF pump tanks received waste transfers from the F-Canyon Facility.
- Pump pits, which house each pump tank, are reinforced concrete structures located below grade at low points of the transfer lines.
- The single FTF catch tank is designed to collect drainage from the diversion box and Type 1 tank transfer line encasements.
- The two evaporator systems were used to reduce the amount of liquid volume of radioactive waste resulting from nuclear processes.

DOE developed ancillary equipment inventories for the transfer lines, pump and catch tanks, and evaporators and concentrate transfer system.

Secondary containments for transfer lines and pumps, as well as diversion and valve boxes were assumed to have insignificant inventories such that the inventory of the other ancillary

equipment bounded the impacts of these components. No leakage of waste from primary core pipe into secondary containment has been identified.

Dry sludge concentrations were used to estimate the inventory for ancillary equipment. DOE indicated that this approach is conservative because supernate, as opposed to dry sludge, was the primary waste transferred through the lines, and actinides and long-lived radionuclides are more highly concentrated in the sludge than in the supernate. While the supernate is more highly concentrated in short-lived radionuclides than the sludge, DOE expects that the short-lived fission products are not a primary risk driver for the inadvertent intruder because the intrusion is assumed to occur at 100 years. DOE reduced the sludge concentrations by 20 percent to estimate the ancillary equipment inventory. Because 16.7 weight percent of solids is a constraint on waste transfers, the use of a slightly higher percentage of the dry sludge concentrations is considered by DOE to be reasonable. DOE calculated a weighted average of tank concentrations that contributed to the inventory for any particular transfer line segment.

DOE made analytical calculations to determine the amount of residual radioactivity remaining in transfer lines. These calculations considered three different processes: 1) Diffusion into the metal; 2) Residue in the oxide film; and 3) Residue left behind after a transfer and flushes.

For the first process, DOE evaluated diffusion into the metal using Fick's second law. For the second process, DOE assumed that the protective oxide film on the steel is composed of 20 weight percent sludge. For the third process, DOE assumed that waste transfers occur (with waste concentrations equivalent to 20 weight percent of the measured sludge concentrations) and that the transfer lines are flushed with a volume of water three times the line volume following the transfer as indicated by normal operating procedure. DOE used an analytical approach that included a continuously stirred tank reactor to calculate the residual inventory following flushing. This third mechanism for waste buildup in the transfer lines led to the highest estimated inventory—the other two processes led to insignificant residue inventory (less than one percent of the total inventory from all three processes). Table 3.3-11 of Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) lists the estimated inventories of the residual contaminants after flushing three times.

The pump and catch tank inventory assumes that 0.2 cm (0.06 in) of residual material will remain in these vessels after rinsing and flushing. Tank 8 inventory was assumed as a starting inventory for the FTF catch tank because Tank 8 inadvertently overflowed into the catch tank. No other tanks were associated with significant events with the FTF catch tank. Table 3.3-16 of the Rev. 1 PA lists the inventory for the pump tanks and the FTF catch tank.

DOE estimated the inventory of the evaporators and concentrate transfer system (CTS) based on field characterization data. CBU-LTS-2004-00078 discusses characterization that has been completed. One of the evaporators, 242-F, has already undergone cleaning, along with the CTS. The second evaporator system (242-16F) is expected to meet the same level of decontamination as 242-F. Additional cleaning of the CTS is planned before closure, but DOE assumed no additional waste retrieval in estimating the impacts from the CTS in the PA calculations.

3.2 NRC Evaluation of Waste Inventory

The TER discusses inventory in this section with respect to NRC's evaluation of compliance with Criterion 2. Section 4.2.7 of the TER discusses NRC staff's evaluation of inventory as it pertains specifically to the Criterion 3 evaluation. It is important to note subtle differences in NRCs approach to evaluating DOE's inventory when considering Criterion 2 versus Criterion 3. For example, acknowledging that the inventory for FTF tanks and auxiliary components that have not been cleaned cannot be known at this time, an estimate of cleaning effectiveness of the final inventory must necessarily be made for the purposes of demonstrating removal to the MEP (Criterion 2). It would follow that the same process would be used to develop the final inventory for the purpose of estimating potential long-term risks from the disposal facility under Criterion 3 or at least some relationship between the two inventories would exist.

Because the inventory is uncertain, DOE could elect to develop an inventory based on the best available information. However, when the NDAA criteria are easily met or results of the PA are relatively insensitive to the inventory (e.g., because the costs of additional removal are relatively high or because most key radionuclides are solubility limited), there is a tendency to develop "conservative" estimates of the inventory in the PA that are easy to defend and achieve. However, determining when an estimate is conservative may not always be straightforward.

For example, DOE may elect to develop an inventory that is biased high based on low removal estimates. While it is a relatively simple matter to demonstrate that over-estimating the inventory for the Criterion 3 evaluation is clearly more conservative (does not impact the dose results or biases the dose results high), it is not as easy to determine whether it is more conservative to over- or under-estimate the inventory for the purposes of the Criterion 2 evaluation for those tanks and components that have yet to be cleaned. For example, pessimistic estimates of cleaning technology effectiveness may suggest the relative difficulty of removing additional radioactivity from the tanks and auxiliary components, thereby potentially increasing the relative costs associated with additional radionuclide removal or providing DOE with a reason to terminate cleaning operations earlier, rather than later. In this sense, higher residual inventory estimates (or lower cleaning technology effectiveness) when considering the practicality of additional radionuclide removal for a particular technology may tend to bias the Criterion 2 evaluation in a non-conservative direction. On the other hand, if higher residual inventories are assumed with no presumption or optimistic assumptions regarding cleaning effectiveness, one might conclude that the potential benefits associated with additional radionuclide removal may be higher than they actually are, necessitating the consideration of additional cleaning technologies and waste retrieval that may not be cost beneficial. Thus, selection of a "conservative" approach to developing FTF inventories for the purpose of performing a Criterion 2 evaluation or setting removal goals is not always clear.

For tanks that have not yet been cleaned, DOE attempts to over-estimate inventories, which is conservative for the purposes of the Criterion 3 evaluation (see Section 4.2.7). However, DOE does not provide explicit removal goals as part of the Criterion 2 evaluation. Instead, DOE indicates that clean-up will proceed until it is no longer practical to do so. Although DOE Rev. 0 (SRS-REG-2007-00002, Rev. 0) modeling assumption that 0.16 cm (0.06 in) will remain in the tanks post waste retrieval could be considered a de facto removal goal, DOE provides an adjustment factor of an order of magnitude to account for "uncertainties" in future waste retrieval providing some latitude with respect to the expected removal efficiencies in the Rev. 1 PA

(SRS-REG-2007-00002, Rev. 1). With respect to tanks that have been cleaned and characterized (Tanks 18 and 19), DOE provides limited information regarding technology removal effectiveness due to complicating factors such as uncertainty in the baseline inventory or difficulty with selecting appropriate effectiveness measures such as volume reduction[2]. For the purpose of the Criterion 2 evaluation, NRC will focus on the approach DOE used to develop the final inventory estimates for Tanks 18 and 19. This approach will serve as a model for characterization of other FTF tanks following cleaning operations in the future, rather than evaluation of removal goals or estimates described above, although some comparisons between projections and final estimates are made.

3.2.1 NRC Evaluation of Projected Inventories for Tanks

NRC commented on DOE's Rev. 0 PA (SRS-REG-2007-00002, Rev. 0) and expressed concerns with (i) significant under-predictions of key radionuclide concentrations based on the WCS (e.g., Tc-99, Cs-137, Am-241), (ii) potentially overly optimistic assumptions regarding the residual waste volumes for tanks yet to be cleaned with oxalic acid of 0.2 cm (0.06 in), and (iii) lack of basis for the assumed 75 percent removal estimates for Tanks 18 and 19.

DOE responded to NRC's concerns on the Rev. 0 PA (SRS-REG-2007-00002, Rev. 0) with respect to tank inventory development. DOE used a generally more conservative approach to developing inventories in the Rev. 1 PA (SRS-REG-2007-00002, Rev. 1). The inventory was updated using a 10 step approach discussed in the PA and summarized in the text above. To address the concern of under-predicting concentrations, DOE applied the maximum concentrations for each tank type, and assumed a factor of 10 increase in the inventory for Type I and IIIA tanks.

It is important to note, that in applying the maximum concentration of a single tank to all tanks of the same type, DOE conservatively assumed that all eight of the Type I tanks in FTF contain 2.9×10^{12} Bq (79 Ci) of Tc-99. In fact, preliminary analysis of sampling results for Tanks 5 and 6, the first two FTF Type I tanks that have undergone residual heel removal campaigns, suggests that the residual inventory of Tc-99 will be significantly less than 2.9×10^{12} Bq (79 Ci) (i.e., by as much as two orders of magnitude, see RAI-UA-3 response [DOE, 2011] that cites data in SRNL-STI-2009-00492 and SRNL-STI-2009-00493). NRC staff agrees that the inventory of Tc-99 for Type I tanks is likely significantly lower than assumed in the Rev. 1 PA.

The uncertainty analysis in the Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) includes revised estimates of inventory uncertainty based on tank type and radionuclide (see Table 5.6-3 in the PA, SRS-REG-2007-00002, Rev. 1, and SRR-CWDA-2009-00045 for more information on how these distributions were developed). DOE assumes a uniform distribution that ranges from a factor of 100 times less to a factor of 10 times greater than inventory depending on tank type and radionuclide. DOE's approach to developing parameter distributions for the FTF inventory

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[l]In the case of chemical cleaning, volume estimates may not be the most appropriate measure of cleaning effectiveness in the case of preferential removal of waste components or due to the formation of precipitates in the waste sludge.

for tanks that have yet to be cleaned is biased towards lower values but is considered reasonable given the conservatisms of the base case (deterministic) values.

NRC staff has reviewed the approach used to estimate the inventories of those tanks that have not yet been cleaned and agrees that the approach generally biases the inventory estimates high. Because DOE has not stated quantitative removal goals, the main use of projected inventories is to predict doses for Criterion 3. Additionally, because final inventories have not been provided for Type I, III, and IIIA tanks (only Type IV final inventories have been provided), the Criterion 2 determination cannot be made for Type I, III, and IIIA tanks. However, it is helpful to describe DOE's approach to developing the inventory in this section of the TER so that DOE's projections can be compared to final characterization values for Tanks 18 and 19 and to other tanks as they are cleaned in the future. Regarding tanks with zeolites (Tanks 7, 25, and 27), DOE indicates its commitment to addressing zeolite uncertainty through waste retrieval and sampling in its comment responses on the Rev. 0 PA (SRR-CWDA-2009-00054). NRC staff will follow-up on this commitment during the monitoring period.

3.2.2 NRC Evaluation of Characterization of Tanks 18 and 19

The uncertainty analysis in DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) assumes a uniform distribution with a minimum of 0.5 and maximum of 2 multipliers for Tanks 18 and 19. Because final sampling data from Tanks 18 and 19 were not available in time for use in the Rev. 1 PA (SRS-REG-2007-00002, Rev. 1), NRC staff was concerned that the assignment of a factor of 2 multiplier for the upper and lower bounds of the Tank 18 and 19 inventories in the probabilistic analysis might not be sufficient to capture the true uncertainty in the inventories for these tanks (see CC-IN-1, (Camper, 2010)).

In comment CC-IN-1 (Camper, 2010) on the Rev. 1 PA, NRC staff also requested a breakdown in the Tanks 18 and 19 inventory uncertainty and a comparison of projected to final inventory data to ensure that the assumed PA inventory did not under-predict the dose. Table 3-3 shows NRC staff's comparison of the projected and final inventories; the risk-significant radionuclides that were underestimated in the original projections are highlighted in red. Other radionuclides that were underestimated are highlighted in yellow. The sampling results show that for Tank 18 plutonium was underestimated, which is important to determining the long-term risk in the PA.

Table 3-3 Comparison of Final Characterization with Projected Inventories for Tanks 18 and 19

Constituent	Units	Tank 18 Sample Best Estimate (decayed to 2020)	Tank 18 Projected Estimate PA Rev. 1 (decayed to 2020)	Factor Difference	Tank 19 Sample Best Estimate (decayed to 2020)	Tank 19 Projected Estimate PA Rev. 1 (decayed to 2020)	Factor Difference
Ac-227	Ci	$1.5x10^{-4}$	**$1.0x10^{-3}$**		$9.6x10^{-6}$	**$1.0x10^{-3}$**	
Al-26	Ci	$1.9x10^{-4}$	**$1.0x10^{0}$**		$3.8x10^{-5}$	**$1.0x10^{0}$**	
Am-241	Ci	$1.6x10^{2}$	**$8.2x10^{1}$**	$2.0x10^{0}$	$2.6x10^{0}$	**$2.3x10^{0}$**	

Constituent	Units	Tank 18 Sample Best Estimate (decayed to 2020)	Tank 18 Projected Estimate PA Rev. 1 (decayed to 2020)	Factor Difference	Tank 19 Sample Best Estimate (decayed to 2020)	Tank 19 Projected Estimate PA Rev. 1 (decayed to 2020)	Factor Difference
Am-242m	Ci	$3.8x10^{-2}$	**$1.0x10^{0}$**		$2.5x10^{-4}$	**$1.0x10^{0}$**	
Am-243	Ci	$2.3x10^{0}$	**$1.0x10^{-1}$**	$2.3x10^{1}$	$6.8x10^{-3}$	**$1.0x10^{-1}$**	$1.0x10^{0}$
Ba-137m	Ci	$8.7x10^{3}$	**$9.1x10^{3}$**		$4.0x10^{3}$	**$6.2x10^{3}$**	
C-14	Ci	$9.0x10^{-1}$	**$1.0x10^{0}$**		$4.1x10^{0}$	**$1.0x10^{0}$**	$4.0x10^{0}$
Cf-249	Ci	$2.3x10^{-3}$	**$1.0x10^{0}$**		$5.2x10^{-4}$	**$1.0x10^{0}$**	
Cl-36	Ci	$2.8x10^{-4}$	**$1.0x10^{-3}$**		$9.1x10^{-5}$	**$1.0x10^{-3}$**	
Cm-243	Ci	$1.8x10^{-2}$	**$1.0x10^{0}$**		$1.7x10^{-3}$	**$1.0x10^{0}$**	
Cm-244	Ci	$9.8x10^{1}$	**$1.0x10^{2}$**		$2.7x10^{-1}$	**$1.0x10^{0}$**	
Cm-245	Ci	$1.2x10^{-2}$	**$1.0x10^{0}$**		$1.6x10^{-3}$	**$1.0x10^{0}$**	
Cm-247	Ci	$2.1x10^{-6}$	**$1.0x10^{-3}$**		$1.3x10^{-6}$	**$1.0x10^{-3}$**	
Cm-248	Ci	$9.5x10^{-5}$	**$1.0x10^{-3}$**		$5.8x10^{-5}$	**$1.0x10^{-3}$**	
Co-60	Ci	$3.2x10^{-1}$	**$1.0x10^{0}$**		$1.2x10^{-2}$	**$1.0x10^{0}$**	
Cs-135	Ci	$3.0x10^{-2}$	**$1.0x10^{0}$**		$5.4x10^{-2}$	**$1.0x10^{0}$**	
Cs-137	Ci	$9.2x10^{3}$	**$9.7x10^{3}$**		$4.2x10^{3}$	**$6.5x10^{3}$**	
Eu-152	Ci	$4.7x10^{-3}$	**$1.0x10^{0}$**		$1.7x10^{-4}$	**$1.0x10^{0}$**	
Eu-154	Ci	$2.1x10^{-1}$	**$3.2x10^{0}$**		$3.8x10^{-3}$	**$1.0x10^{0}$**	
H-3	Ci	$8.0x10^{-3}$	**$1.0x10^{0}$**		$2.5x10^{-3}$	**$1.0x10^{0}$**	
I-129	Ci	$2.7x10^{-4}$	**$1.0x10^{-3}$**		$2.2x10^{-4}$	**$1.0x10^{-3}$**	
K-40	Ci	$1.6x10^{-2}$	**$1.0x10^{-3}$**	$1.6x10^{1}$	$1.0x10^{-3}$	**$1.0x10^{-3}$**	
Nb-93m	Ci	$8.6x10^{-2}$	**$1.0x10^{-3}$**	$8.6x10^{1}$	$1.8x10^{-2}$	**$1.0x10^{-3}$**	$1.8x10^{1}$
Nb-94	Ci	$5.5x10^{-4}$	**$1.0x10^{-3}$**		$1.0x10^{-4}$	**$1.0x10^{-3}$**	
Ni-59	Ci	$3.3x10^{-1}$	**$1.0x10^{0}$**		$3.5x10^{-4}$	**$1.0x10^{0}$**	
Ni-63	Ci	$1.6x10^{1}$	**$8.2x10^{1}$**		$1.3x10^{-2}$	**$1.4x10^{1}$**	
Np-237	Ci	$1.9x10^{-1}$	**$2.4x10^{-1}$**		$1.5x10^{-3}$	**$2.2x10^{-3}$**	
Pa-231	Ci	$4.6x10^{-2}$	**$1.0x10^{-3}$**	$4.6x10^{1}$	$6.9x10^{-5}$	**$1.0x10^{-3}$**	
Pd-107	Ci	$1.2x10^{-1}$	**$1.0x10^{-3}$**	$1.2x10^{2}$	$2.0x10^{-1}$	**$1.0x10^{-3}$**	$2.0x10^{2}$
Pt-193	Ci	$3.6x10^{-3}$	**$1.0x10^{-3}$**	$4.0x10^{0}$	$1.5x10^{-3}$	**$1.0x10^{0}$**	
Pu-238	Ci	$1.3x10^{3}$	**$7.0x10^{1}$**	$1.9x10^{1}$	$3.4x10^{0}$	**$4.4x10^{0}$**	
Pu-239	Ci	$2.8x10^{2}$	**$1.6x10^{2}$**	$1.8x10^{0}$	$4.0x10^{0}$	**$6.4x10^{0}$**	
Pu-240	Ci	$6.5x10^{1}$	**$4.9x10^{1}$**	$1.3x10^{0}$	$9.8x10^{-1}$	**$2.3x10^{0}$**	
Pu-241	Ci	$2.7x10^{2}$	**$1.3x10^{2}$**	$2.0x10^{0}$	$3.9x10^{0}$	**$4.6x10^{0}$**	
Pu-242	Ci	$2.7x10^{-2}$	**$1.0x10^{0}$**		$1.7x10^{-3}$	**$1.0x10^{0}$**	
Pu-244	Ci	$6.2x10^{-6}$	**$1.0x10^{-3}$**		$5.3x10^{-6}$	**$1.0x10^{-3}$**	

Constituent	Units	Tank 18 Sample Best Estimate (decayed to 2020)	Tank 18 Projected Estimate PA Rev. 1 (decayed to 2020)	Factor Difference	Tank 19 Sample Best Estimate (decayed to 2020)	Tank 19 Projected Estimate PA Rev. 1 (decayed to 2020)	Factor Difference
Ra-226	Ci	$3.4x10^{-3}$	**$1.9x10^{-3}$**	$1.8x10^{0}$	$4.1x10^{-3}$	**$1.1x10^{-3}$**	$4.0x10^{0}$
Sb-126	Ci	$1.8x10^{-3}$	**$2.3x10^{-2}$**		$4.7x10^{-4}$	**$3.6x10^{-3}$**	
Sb-126m	Ci	$1.3x10^{-2}$	**$1.6x10^{-1}$**		$3.3x10^{-3}$	**$2.6x10^{-2}$**	
Se-79	Ci	$4.8x10^{-4}$	**$1.0x10^{0}$**		$4.6x10^{-4}$	**$1.0x10^{0}$**	
Sm-151	Ci	$3.7x10^{1}$	**$4.6x10^{1}$**		$1.5x10^{-1}$	**$1.0x10^{0}$**	
Sn-126	Ci	$1.3x10^{-2}$	**$1.6x10^{-1}$**		$3.3x10^{-3}$	**$2.6x10^{-2}$**	
Sr-90	Ci	$2.5x10^{3}$	**$1.1x10^{3}$**	$2.0x10^{0}$	$6.9x10^{0}$	**$5.2x10^{0}$**	$1.3x10^{0}$
Tc-99	Ci	$9.0x10^{-1}$	**$1.0x10^{0}$**		$3.8x10^{-1}$	**$1.4x10^{0}$**	
Th-229	Ci	$8.9x10^{-4}$	**$2.6x10^{-3}$**		$2.0x10^{-4}$	**$1.0x10^{-3}$**	
Th-230	Ci	$2.1x10^{-3}$	**$1.9x10^{-3}$**	$1.1x10^{0}$	$1.1x10^{-4}$	**$1.1x10^{-3}$**	
U-232	Ci	$6.9x10^{-4}$	**$1.0x10^{0}$**		$9.5x10^{-5}$	**$1.0x10^{0}$**	
U-233	Ci	$4.0x10^{-2}$	**$1.1x10^{0}$**		$4.3x10^{-3}$	**$1.9x10^{-1}$**	
U-234	Ci	$3.1x10^{-1}$	**$3.8x10^{-1}$**		$4.8x10^{-3}$	**$1.1x10^{-2}$**	
U-235	Ci	$1.1x10^{-2}$	**$8.4x10^{-3}$**	$1.3x10^{0}$	$1.7x10^{-4}$	**$2.6x10^{-4}$**	
U-236	Ci	$1.2x10^{-2}$	**$1.0x10^{0}$**		$2.5x10^{-4}$	**$1.0x10^{0}$**	
U-238	Ci	$2.8x10^{-1}$	**$2.2x10^{-1}$**	$1.27x10^{0}$	$5.4x10^{-3}$	**$8.7x10^{-3}$**	
Y-90	Ci	$2.5x10^{3}$	**$1.1x10^{3}$**	$2.0x10^{0}$	$6.9x10^{0}$	**$5.2x10^{0}$**	
Zr-93	Ci	$8.6x10^{-2}$	**$1.0x10^{-3}$**	$8.6x10^{1}$	$1.8x10^{-2}$	**$1.0x10^{-3}$**	$1.8x10^{1}$

NRC staff also evaluated the final characterization efforts for Tanks 18 and 19, which serve as the basis for the inventory comparison above. NRC staff evaluated DOE's approach to: (i) estimating residual waste volume; (ii) residual waste sampling and analysis; as well as (iii) consideration of uncertainty. With respect to volume estimates, in total, DOE estimates that 15.2 m^3 (4,000 gal) remain in Tank 18 and 7.6 m^3 (2,000 gal) in Tank 19 through material mapping. The upper Reasonably Conservative Estimate on Tank 18 is considered to be 20.1 m^3 (5,300 gal). NRC staff notes that there is significant uncertainty in the material mapping approach. The material mapping approach is somewhat subjective, and the mapping team had to apply weights to depths of areas that look similar to areas that lacked relative objects for depth comparison. This estimation approach had to be applied to the southwest area of Tank 18, which has the largest residual mound, and has been characterized as different from the rest of the tank in previous documentation (WSRC-TR-2003-00449 and SRR-CWDA-2010-00117).

NRC staff reviewed the sampling and analysis plan with respect to the representativeness of the samples, detection limits, and consideration of analytical uncertainty (SRR-CWDA-2010-00117 and SRR-CWDA-2010-00118 provide an overview of the characterization and sampling locations; SRNL-STI-2009-00782 and SRNL-STI-2009-00779 describe the basis for the number of samples and the locations; SRNL-STI-2010-00386 and SRNL-STI-2010-00439 provide

details on the detection limits and methods used). In 2009, DOE recommended that the southwest outer ring of Tank 18 be sampled because the material in this region differed from the other material in the tank prior to the Mantis cleaning (SRNL-STI-2009-00782). In 2009, DOE used the Mantis to take samples from 7 sites on the floor of Tank 18. The 2010 samples showed that the floor of Tank 18 was relatively homogeneous. While the statistical analysis of the six sample sites did reflect homogeneity, DOE has not fully explained why this area is no longer different.

DOE determined the final sample size based on how the additional data reduced uncertainty: characterizing 5 additional samples from Tank 18 and 6 additional samples from Tank 19 reduced uncertainty by more than half and further samples showed diminishing effectiveness at reducing uncertainty (SRNL-STI-2009-00782, SRNL-STI-2009-00779). As described in SRNL-STI-2009-00782, to determine the number of additional floor scrape samples to be taken in 2010 from Tank 18, DOE established a one-sided upper 95 percent confidence limit for each radionuclide from the 2009 samples. Then, DOE calculated a hypothetical change in the UCL that assumed that another sample would be taken. So, in this regard, the 2009 sample variance was used to determine the number of 2010 samples.

NRC staff notes that when the floor scrape samples were compared to the 2009 in-process samples, they were not sufficiently similar for certain radionuclides (e.g.: Ag, Ba-137m, Cd, Cr, Cs-135, Cs-137, Mn, Pu-239, Tc-99 and U-232). DOE stated that the 2009 samples may have been affected by the residence time in the grinder (SRNL-STI-2010-00401). NRC staff did not find this rationale sufficient because it does not explain the mechanism or process within the grinder that would cause a change in concentration, nor did it explain why only some radionuclides would experience an impact.

For those radionuclides that were not detected above instrument detection levels, DOE assigned the lowest detection limit as the best estimate concentration and the highest limit for the reasonably conservative estimate. Based on factors such as interference from other radionuclides, the reported instrument or method minimum detection limit can differ between sample analyses. Using the highest detection limit for the best estimate would have been more conservative, but because these radionuclides were not present, or present in very small quantities, and the difference in the highest versus lowest detection limit was not great, this is a reasonable approach. For Cl-36 and Ra-226, DOE improved the measurement approaches to detect them at the target levels, which is the concentration that would yield the inventory assumed in the PA along with the assumed volume (SRR-CWDA-2010-00117). Because the difference between the high and low detection limits was significant, DOE believed it would be overly conservative to use the high detection limit for the reasonably conservative estimate and used one order of magnitude above the best estimate instead. This approach is reasonable because DOE had made extra effort to measure these radionuclides at lower levels.

DOE analyzed the impacts of the final inventories on the Rev. 1 PA results in a special analysis (SA) (SRR-CWDA-2010-00124). The SA covers each section of the PA, describing the specific impacts of the new inventory. The results of DOE's SA for Tanks 18 and 19 are covered in Section 3.7 of this TER. DOE accounted for uncertainty in the SA through a deterministic sensitivity analysis and a probabilistic analysis. The deterministic sensitivity analysis assumed the Reasonably Conservative inventory (which was twice the standard deviation added to the

mean). The probabilistic analysis involved developing inventory multipliers for each radionuclide, which is described in the following paragraph.

The inventory multipliers were developed in a separate GoldSim model that sampled distributions for the tank floor, lower and upper wall concentration; and tank floor and scale volume in over 10,000 realizations, producing an inventory for each realization. The distributions in the separate GoldSim model assumed that both concentration and volume were normally distributed. For the tank floor concentration, DOE fit a normal distribution to the mean and standard deviation of the six floor samples. For tank floor and scale volume, DOE fit a normal distribution using the Best Estimate as the mean and the Reasonably Conservative Estimate (30 percent higher) as the 95th percentile. NRC staff notes that artificially fitting a normal distribution for the volume may not properly take into account the variability of the depths of residual material on the tank floor. DOE also could have assessed the uncertainty by looking at the variability in the depths estimated through material mapping and developing distribution models based on that sample population (weighted for the area of each sample). This method could have resulted in an alternative estimate of the uncertainty of the residual volume. NRC staff notes that DOE did not treat tank wall volume as a variable in this separate simulation to determine the inventory multipliers.

DOE divided the simulated inventory by the Best Estimate to obtain an inventory multiplier for each of the 10,000 runs. The average and standard deviation of the 10,000 multipliers were applied within a normal distribution in the probabilistic analysis. This is opposed to the assumption of a uniform distribution for the multipliers using the Rev. 1 PA. DOE did not provide a basis for the assumption of a normal distribution. The overall result of using a normal distribution is that the uncertainty distributions in the probabilistic PA have shaper peaks and longer tails. For example, the projected inventory used in the PA for Pu-238 was 2.6 x 10^{12} bq (70 Ci) and the multipliers were 0.5 to 2, so the uncertainty analysis assumed uniform distribution from 1.3 x 10^{12} Bq (35 Ci) to 5.2 x 10^{12} Bq (140 Ci). The final characterization Best Estimate for Pu-238 was 4.8 x 10^{13} Bq (1,300 Ci) and the multipliers were 0.4 for average and 0.19 for standard deviation, so the uncertainty analysis assumed a normal distribution with mean of 2.0 x 10^{12} Bq (520 Ci) and 9.3 x 10^{12} Bq (250 Ci) standard deviation. For Np-237, the PA projected inventory was 9.0 x 10^{9} Bq (0.24 Ci) and the multipliers were 0.5 and 2, so the uncertainty analysis assumed a uniform distribution from 4.4 x 10^{9} Bq (0.12 Ci) to 1.8 x 10^{10} Bq (0.48 Ci). The final characterization Best Estimate for Np-237 for Tank 18 was 7.0 x 10^{9} Bq (0.19 Ci), and the multipliers were 0.64 and 0.35, so the uncertainty analysis assumed a normal distribution with a mean of 4.4 x 10^{9} Bq (0.12 Ci) and a 2.5 x 10^{10} Bq (0.07 Ci) standard deviation. NRC staff also notes that for many radionuclides, such as Np-237, the inventory is rarely sampled higher than the Best Estimate in the probabilistic evaluation. For Np-237, two standard deviations above the mean is a value of 0.14, while the Best Estimate is 0.19. NRC staff expects DOE to have more certainty of the true value when additional information is known regarding the inventory. However, it is important to keep in mind how the multipliers are derived and how they compare to the original uncertainty that was predicted over the course of cleaning efforts, as well as how the assumption of a normal distribution impacts the values that are being selected for the probabilistic analysis.

NRC staff also evaluated DOE's development of the tank wall inventory based on corrosion estimates and iron oxide sorption. NRC staff asked why DOE did not detect discernable material on the stiffening bands or walls of Tank 19 but found a film left on Tank 18 walls. DOE

explained that the Tank 19 wall residue had been removed during a second washing of the walls. A second washing was not carried out for the Tank 18 walls because the original overhead video was not of high enough resolution to show discernable residual material, and therefore, DOE did not believe a second washing would be necessary for Tank 18.

NRC staff notes that the Pu-238 concentrations in the upper wall, lower wall, and scale samples were higher than in the floor samples, which was not expected by DOE and was not consistent with the sample analysis results for the other radionuclides. For all other plutonium isotopes, the wall samples were similar to the floor samples. Although the amount in terms of volume on the walls is small in comparison to that on the tank floor; this is important to note because Pu-238 is included in the set of HRRs that are considered important to the long-term performance of the FTF (due to the impact of its progeny U-234, Th-230, and Ra-226 that grow-in over time). NRC staff does not agree that applying the floor concentrations for the scale material is conservative for those constituents that showed a higher concentration in the scale samples. For example, the average and 95th UCL for the floor samples for Tank 18 were 1.5 x 10^{5} Bq/g (3.99 x 10^{-6} Ci/g) and 1.8 x 10^{5} Bq/g (4.8 x 10^{-6} Ci/g) respectively (SRNL-STI-2010-00401), while the average for the scale sample was 5.6 x 10^{5} Bq/g (15.1 x 10^{-5} Ci/g) (SRNL-STI-2010-00525). The fact that DOE could not objectively explain these results might indicate greater uncertainty in the Pu-238 inventory measurements than that is currently estimated.

Overall, DOE has attempted to properly characterize the final inventory of the residual material in Tanks 18 and 19. DOE has applied methodical approaches to determine the remaining volume, the number of samples, the location of the samples, and has attempted to quantify the uncertainty associated with both concentration and volume on the tank floor and walls. However, NRC staff has remaining concerns with the approach, specifically with regard to quantification of volume uncertainty, the lack of explanation in the differences between the 2009 and 2010 sample results, the unexpected Pu-238 concentrations in the tank walls, and the assumptions surrounding development in inventory multipliers for the probabilistic analysis. NRC staff thinks that DOE could improve waste characterization, particularly with regard to determining the remaining volume in the portions of the tanks that could not be mapped due to lack of relative objects given the relative importance and known history of some of these areas. The application of weights based on similar regions for these portions is not transparent; it is not clear how similarities were determined given that the portions were not able to be mapped. DOE did not discuss whether alternative technologies are available for mapping.

3.2.3 NRC Evaluation of Ancillary Equipment Inventory

In reviewing the Rev. 0 PA (SRS-REG-2007-00002, Rev. 0), NRC staff asked DOE for additional information on the development of the ancillary equipment inventory using weighted estimates of activities in the tanks. DOE indicated that the estimate was based on best estimates from waste transfer data. NRC also questioned the appropriateness of the continuously stirred tank reactor model used to calculate the transfer line inventory. DOE indicated that the model is appropriate because of the high velocity of slurried sludge in the transfer lines (SRR-CWDA-2009-00054). DOE also argued that the concentrations are conservative due to the assumption regarding the transfer of sludge versus supernate. Due to the low risk-significance associated with the auxillary equipment given the inventories assumed by DOE, no further justification is required by DOE.

DOE also indicated in its comment responses that PA Section 8.2, "Further Work," discusses future characterization of the transfer lines (SRR-CWDA-2009-00054). Section 8.2 states that further work should be conducted to refine and confirm the assumed radionuclide inventories including refinement of potential waste estimates for areas that have not been sampled, such as the piping and other ancillary equipment. However, Section 8.2 does not make firm commitments to sampling of the ancillary equipment or transfer lines. NRC staff will follow-up on the need for auxiliary equipment characterization in the monitoring period.

NRC noted that the list in Table 3.3-11 of Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) has 60 radionuclides instead of the 54 radionuclides listed in the projected tank inventories. The 60 radionuclides match those that are listed in the Rev. 0 of the PA (SRS-REG-2007-00002, Rev. 0). NRC informed DOE that the ancillary equipment list of radionuclides was not updated from the Rev. 0 to Rev. 1 PA. DOE plans to explain the difference in the next PA revision.

3.3 Identification of Highly Radioactive Radionuclides

DOE performed an evaluation that considered the risk to workers, the public, and the environment to identify HRRs that, according to NDAA Section 3116(a)(2), must be removed to the MEP. DOE summarized the methodology for the evaluation in Section 5.1.1 of DOE/SRS-WD-2010-001. The methodology consists of a screening approach and assessment of whether the radionuclides may be important to demonstrating compliance with the performance objectives of 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," Subpart C. The screening approach, which is summarized in Section 3.1 of this TER, resulted in a list of 54 radionuclides for consideration in the assessment of compliance with the performance objectives in DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1). The remainder of this section will summarize the assessment of compliance with the performance objectives.

To determine which screened radionuclides were HRRs, DOE considered which radionuclides may be important to demonstrating compliance with the Part 61 Subpart C performance objectives because they contribute to the dose to members of the public, inadvertent intruders, and workers. Specifically, the assessment considered analyses performed for the FTF PA including: (i) doses estimated for the groundwater pathway at 100 m; (ii) doses estimated the airborne pathway at 100 m; and (iii) doses estimated for the inadvertent intruder; and (iv) results of uncertainty and sensitivity analyses. The evaluation resulted in 10 HRRs that are listed in Table 3-4. The methodology is summarized below.

DOE examined resulting doses from the ground water analysis in the FTF PA at 100 m over a 20,000 yr period for all FTF sources to identify potential HRRs. Those radionuclides, which in aggregate were not estimated to contribute more than 0.0125 mSv/yr (1.25 mrem/year) were screened from the list of 54 radionuclides and not considered HRRs. The evaluation identified nine radionuclides for consideration as HRRs: I-129, Cs-135, Ra-226, Th-229, U-233, U-234, Np-237, Pu-239, and Pu-240. Further, the analysis considered the projected inventories of these radionuclides at the time of closure. Radionuclides with insignificant inventories were removed from consideration if their parent was included for consideration as an HRR. Two radionuclides (Th-229 (progeny of U-233) and Ra-226 (progeny of both U-234 and Pu-238)) were removed for this reason while two parent radionuclides (Am-241 (parent of Np-237) and Pu-238 (parent of Ra-226)) were added to the list based on progeny ingrowth. Furthermore, DOE considered waste characterization sample data from Tanks 18 and 19 (SRR-CWDA-2010-

00117, SRR-CWDA-2010-00118) to remove Cs-135 and U-233 from consideration. These radionuclides were removed because their inventory in the samples was significantly lower than projected in the PA and, when considered in aggregate with previously screened radionuclides, they would not result in a dose greater than 0.0125 mSv/yr (1.25 mrem/year). The ground water analysis resulted in the inclusion of seven radionuclides as HRRs: I-129, U-234, Np-237, Pu-238, Pu-239, Pu-240, and Am-241.

Table 3-4 F-Area Tank Farm Highly Radioactive Radionuclides (adapted from Table 5.1-1, DOE/SRS-WD-2010-001, Rev. 0)

Radio-nuclide	Half-Life (yrs)	Table 1 of 10 CFR 61.55	Table 2 of 10 CFR 61.55	Ground-water Dose	Intruder Dose	Uncert-ainty in Dose	Worker Dose
Sr-90	2.9×10^{1}		X		X	X	X
Tc-99	2.1×10^{5}	X				X	
I-129	1.6×10^{7}			X			
Cs-137	3.0×10^{1}		X		X	X	X
U-234	2.5×10^{5}			X		X	
Np-237	2.1×10^{6}	X		X	X	X	
Pu-238	8.8×10^{1}	X		X			
Pu-239	2.4×10^{4}	X		X		X	
Pu-240	6.6×10^{3}	X		X		X	
Am-241	4.3×10^{2}	X		X	X	X	

DOE also examined resulting doses from the airborne pathway analysis in the FTF PA at 100 m to identify potential HRRs. The airborne pathway is not estimated to be a significant contributor to dose for members of the public and contributes, in aggregate, approximately 2.0×10^{-3} mSv/yr (0.2 mrem/year) dose. Therefore, no radionuclides were included as HRRs based on the airborne pathway.

Using the results from the FTF PA inadvertent intruder dose analysis, DOE identified potential HRRs. Those radionuclides that, in aggregate, would not contribute more than 0.25 mSv/yr (25 mrem/year) were not considered HRRs. The evaluation identified seven radionuclides for consideration as HRRs: Sr-90, Y-90, Cs-137, Ba-137m, Th-229, U-233, and Np-237. Further the analysis considered the projected inventories at the time of closure for those radionuclides. Radionuclides with insignificant inventories were removed from consideration if their parent was included for consideration as an HRR. One radionuclide — Th-229 (progeny of U-233) — was removed for this reason while one parent radionuclide was added to the list — Am-241 (parent of Np-237) — based on progeny ingrowth. Furthermore, DOE considered waste characterization sample data from Tanks 18 and 19 (SRR-CWDA-2010-00117, SRR-CWDA-2010-00118) to remove U-233 from consideration because its inventory in the samples was significantly lower than the PA projected, and when considered in aggregate with previously screened radionuclides would not result in a dose greater than 0.25 mSv/yr (25 mrem/year).

Additionally, DOE removed Y-90 and Ba-137m from further consideration as an HRR because they are typically associated with their parent radionuclides (i.e., Sr-90 and Cs-137, respectively) for human health protection due to their short half-lives. The inadvertent intruder analysis resulted in the inclusion of four radionuclides as HRRs: Sr-90, Cs-137, Np-237, and Am-241.

DOE also examined results of uncertainty and sensitivity analyses for protection of the public and an inadvertent intruder to identify potential HRRs. Using the results of the realizations with the highest peak doses for members of the public from uncertainty analysis for the FTF PA, DOE identified four significant radionuclides: Tc-99, Np-237, Pu-239, and Pu-240. Using the results of the sensitivity analysis, DOE identified Tc-99 and Pu-239 to be significant for protection of the public. Using the results of the deterministic sensitivity and barrier analysis for protection of the public, DOE identified Tc-99, U-234, Np-237, Pu-239, and Am-241. As a result of this assessment, Tc-99, U-234, Np-237, Pu-239, Pu-240, and Am-241 were included as HRRs. DOE also evaluated the results of sensitivity analyses for the inadvertent intrusion scenario leading to further consideration of five radionuclides as potential HRRs: Sr-90, Y-90, Cs-137, Ba-137m, and Am-241. DOE removed Y-90 and Ba-137m from further consideration as an HRR because they are typically associated with their parent radionuclides (i.e., Sr-90 and Cs-137, respectively) for human health protection due to their short half-lives. The assessment of sensitivities for the inadvertent intruder analysis resulted in the inclusion of the following radionuclides as HRRs: Sr-90, Cs-137, and Am-241.

3.4 NRC Evaluation of Highly Radioactive Radionuclides

The definition of “HRRs” used by DOE appears to be consistent with NRC’s understanding of the term. Specifically, NRC staff agrees with DOE that HRRs are those radionuclides that contribute most significantly to radiological risk to the public, workers, and the environment. Section 3.4 summarizes DOE’s approach to developing the list of HRRs. Based on the approach, DOE identified 10 radionuclides as HRRs. The HRRs are listed in Table 3-4.

NRC staff evaluated DOE’s selection of HRRs and notes that the combination of approaches used by DOE to identify HRRs in the context of the draft waste determination for SRS FTF is reasonable. However, as discussed in Sections 3.1 and 4, as DOE continues to evaluate assumptions and model support for the FTF PA and its inventory as a result of consultation and monitoring activities, DOE should concurrently re-evaluate its list of HRRs as new information that could significantly change the results of its HRR evaluation becomes available. For instance, DOE used the results of waste characterization for Tanks 18 and 19 to remove radionuclides (i.e., Cs-135, U-233, Th-229, Ra-226) from consideration as HRRs. There is an implicit assumption that these tanks will be representative of the residual inventory for the entire tank farm. DOE should assess, through future tank residual characterization, the validity of this assumption and the resulting decision to remove radionuclides from the list of HRRs.

3.5 Alternative Treatment Technologies

DOE has defined: (i) baseline technologies; and (ii) alternative technologies for cleaning of the tanks. The baseline technologies are proven to have met a set of program requirements and constraints. Because activities at SRS incorporated several different radioactive separations processes, each producing waste streams with different physical properties, the waste in the

FTF tanks has a wide range of fluid behavior. The varying fluid/sludge behavior combined with the high radiotoxicity of the waste, numerous interior interferences (e.g., cooling coils), and limited access openings means that technology effectiveness would vary across tanks. Therefore, the technologies explored by SRS workers involved variations of mixing, chemical dissolution, mechanical removal, and other remote robotic techniques

In selecting the baseline technologies DOE applied the following program requirements and constraints:

- Tank farm closure must comply with requirements in DOE Manual 435.1-1, and other applicable regulatory requirements.
- The schedule is contingent on budgetary and contractual constraints.
- Cleaning activities require significantly more volume than the waste being removed due to dilution and use of additional liquids, so activities must follow an integrated plan for managing tank storage space.
- Resulting waste streams and used equipment must have a disposal path.
- Cleaning activities should not compromise waste tank integrity.
- Equipment must withstand the local environmental conditions.
- Technologies must consider limitations of the transfer pipe system.
- Technologies must account for internal obstructions in certain tanks (e.g., cooling coils), as well as limited access to the interior (small riser access holes).
- Cleaning activities need to consider shared resources with other operational equipment (e.g., evaporators).
- Equipment must withstand high radiation fields and contamination.

DOE has defined the following technologies as part of the baseline for the various phases of waste removal:

Bulk Waste Removal:

- Jet Mixer Pumps: 3 or 4 submersible mixer pumps (SMP) or standard slurry pumps (SLP), which is the precursor to the SMP, are used to entrain and mix the surrounding fluid. The mechanical energy is supplied from pumps to dislodge and suspend settled sludge. Jet mixers are commonly used in large waste tanks where agitation with blade mixers is impractical. The SMP requires more liquid in the waste tank for operation and adds significant pump heat to the waste. The SLP, although less powerful, can operate at lower waste levels.

Mechanical Heel Removal:

- Mechanical Feed and Bleed: incorporates the same jet mixing used for bulk waste removal, but with an extended mixing time and can be used for tanks with or without cooling coils. If a mound exists, nozzles of the mixer pumps can be directed towards a mound and the mixer left in a stationary position for an extended time (indexing method).

- Robotic crawlers (e.g, Mantis): are equipped with an eductor that vacuums waste and sprays to dislodge sediment. Crawlers are used for tanks without cooling coils. Performance is dependent on the physical properties of the waste (results demonstrated that sticky, mud-like sludge like that in Tank 18 is more difficult to remove than the grainy sand-like sludge found in Tank 19).

Chemical Heel Removal:

- Oxalic Acid (OA): is added to the tank through downcomers or spray wash nozzles while SMPs or SLPs provide mixing. OA was chosen for its strength, effectiveness, reducing tendency, and because it was determined to be less corrosive, and readily available.

- Enhanced Chemical Cleaning (ECC): is similar to OA, but uses lower acid strengths to reduce formation of oxalates; the prototype waste tank for ECC will be Tank 8.

In 1996, SRS initiated an effort to investigate alternative technologies in conjunction with the TFA process. The team investigated a number of mechanical agitation techniques that led to prototype development including:

- Free jet flow agitators: agitate sludge fines to facilitate transfer. Use of a specific free jet flow agitator, called a Flygt mixer, began in Tank 19 in 1998. After several adaptations, three Flygt mixers were installed in Tank 19 for sludge and zeolite mixing. Several issues arose that decreased efficiency: blades had to be cut to accommodate risers; durability was less than anticipated: and zeolite material was more tenacious than surrogate “practice” material. DOE believes that its experience with this technology may be helpful if Flygt mixers are used in the future.

- Sluicing: a commercial fire monitor provides a directed jet sluicing spray to facilitate residual heel removal. A small control box, located above the tank can control pitch and direction of the water brush to “sweep” the tank floor. Material is swept toward the suction of a temporary transfer pump. The sluicing process tends to wash away fine particles, leaving behind large particles.

- Robotic Manipulator Arms: DOE developed a prototype arm for Tank 19 (Type IV), but never deployed the arm because of ineffectiveness or cost. These devices would be developed on a one-of-a- kind basis. So far development and deployment cost cannot be justified. Further, cooling coils and other obstacles that exist in Type I and III tanks would inhibit the maneuverability of robotic arms.

- Advanced Design Mixer Pump (ADMP): the ADMP is similar in appearance to the SLP: it is a long shaft, vertical, centrifugal mixer pump with two tangential nozzles. However, the vertical column is filled with gas, not liquid. It is 16.7 m (55 ft) long and cannot fit through 0.61 m (2 ft) diameter risers. It did, however, fit through Type IV tank center risers. Its theoretical cleaning radius is 15.2 m (50 ft). However, despite its horsepower, it underperformed in Tank 18. Its large size prevents it from being used for Type I tank remediation.

- Pulse Jet Mixer Agitation: the pulse jet air mixer employs a pair of air powered pulse jets. During the charge phase, vacuum is pulled on the charge vessel filling it with waste tank liquid. During the compression phase, the liquid is forced out under air pressure at a high volume. The time for one complete cycle is 1 to 2 minutes. As with the SLP, the assembly is rotated or oscillated using a turntable above the waste tank opening.

Continuing DOE efforts for further refinement and enhancement of technologies include:

- New mixer pumps for optimized agitation and multipurpose use
- Techniques for sludge removal directly beneath risers of waste tanks with cooling coils
- Monitoring progress at other sites
- Monitoring mixed acid flow sheet development
- Continuing the ECC process development
- Continuing small crawler vacuum cleaning system development (e.g., improved Mantis) based on lessons learned from Tank 18 and 19

3.6 NRC Review and Conclusions — Alternative Treatment Technologies

DOE employed a certain type of alternative analysis, referred to as a Systems Engineering Evaluation (SEE), for prior decisions and to select the current baseline technologies. Formal SEEs were conducted at the following time periods:

- Tank 19 Waste Removal (1998)
- Tank 18 Waste Removal (2001)
- Waste Removal Balance of Program (2003)
- Tanks 5 and 6 Heel Removal (2009)

The SEE performed in 1998, which selected the Flygt mixers for the Tank 19 heel removal selection, listed cost (1.25), schedule (1.5), testing (1), infrastructure (1.25), radcon (1.25), downstream impacts (1.25), effect on key resources (1), probability of success (1.5), reliability (1.25), and maturity (1) as constraints and respective weighting factors. Probability of success was defined as the likelihood to satisfy the program objective to design, build, and employ a heel removal campaign within a certain time frame (PIT-MISC-0040).

The SEE carried out in 2001 for Tank 18 heel removal ranked cost (0.32), effectiveness (0.28), complexity (0.25), and authorization basis impact (0.15), as criteria for the viable alternatives, which were screened based on those factors plus technology maturity, systems integration, reliability, and safety. Effectiveness was defined as the degree to which the technology would

perform its proposed function. The Flygt Mixers, which were deployed for Tank 19, were evaluated for Tank 18, but did not rank as high as the ADMP (WSRC-RP-2001-00024).

The Waste Removal Balance of Program SEE completed in 2003, which established much of the current baseline applied cost (0.257), effectiveness (0.177), reliability (0.165), technical maturity (0.166), complexity (0.087), reusability (0.075), integration (0.067), and radiological controls (0.057) as criteria for evaluating those alternatives considered viable (G-ESR-G-00051).

The SEE completed in 2009 for Tank 5 and 6 Heel Removal (SRR-CES-2009-00022) was not provided as a reference, but the references describing the history of waste removal in Tank 5 and 6 include a discussion of this SEE. MFB was chosen over the other mechanical or chemical alternatives because it scored higher in terms of effectiveness and maturity. Cost, schedule, and design impacts involved in installing an above ground transfer line, available tank space, and the ability to manage the additional waste water for beneficial reuse were cited as reasons for selecting the preferred option to supply water from a downcomer for Tank 5 (SRR-CWDA-2011-00033). For Tank 6, a recirculation loop was instead used to maximize mixing and transfer turnover rate and reduce the amount of wastewater added to the system (SRR-CWDA-2011-00005).

Since three of these studies took place prior to the passage of the NDAA in 2005, they would not be expected to apply the specific term HRR in decision criteria, but they could be expected to reasonably consider the effectiveness at reducing the long-term risk of key radionuclides. While DOE provides a comprehensive evaluation of available technologies, considering factors important to tank closure at the time of the 2003 report (G-ESR-G-00051), it does not contain information regarding the technology selection process used to support DOE's demonstration of compliance with objectives embodied in Criterion 2 of NDAA Section 3116. In RAI-MEP-3, NRC staff requested that DOE explain how its technology selection process is consistent with the NDAA criteria. NRC also requested DOE indicate how more recent information is considered in the technology selection process (e.g., technologies that have matured or been developed since issuance of the 2003 report). It is not clear to NRC staff that the 2009 SEE explicitly addresses the NDAA criteria.

Most of the technologies evaluated in the studies, with the exception of the chemical cleaning with OA, involve removing waste volume, as opposed to preferably removing certain radionuclides. (The OA cleaning for Tank 5 and Tank 6, discussed in Section 3.7.2.2, showed preferential removal of certain radionuclides, but the radionuclides that were preferentially removed were not consistent between the two tanks.) As can be seen with the criteria and weighting factors, DOE emphasized cost and effectiveness in all the evaluations. However, it is not clear that effectiveness was linked to removal of the HRRs because the studies do not discuss removal of key or important radionuclides.

To date, DOE has used mechanical technology (mixing and pump removal) for bulk waste removal; and a combination of chemical and mechanical methods to facilitate heel dislodgement and removal. In many cases, technologies have been improved to address physical, chemical, and system challenges. DOE claims that the current waste cleaning process removes more than 99 percent of the waste.

DOE uses a technology selection process to evaluate technologies for future cleaning of tanks, which is discussed in Section 3.7.1. The program includes evaluation of new technologies, but also allows for DOE to rely on a previously performed evaluation where conditions are similar. Also, DOE has indicated that future improvements are likely to be incremental because current processes are removing 99 percent of waste volume (V-ESR-G-00003). NRC staff encourages DOE to continue evaluating new technologies for future use as tank closure progresses.

The focus of the NRC's review is tank heel removal, annulus cleaning, and cleaning of ancillary equipment that will be stabilized in place as part of tank farm closure. NRC staff thinks that DOE can place more emphasis on the development of technologies, such as enhanced chemical cleaning, robotic, and jet spray technologies that will facilitate removal of low volume but high specific activity material. NRC acknowledges the inherent challenges in removing the last increments of material that add significantly to the residual source term. NRC staff is aware that the "cost-benefit" process that drives decisions to continue or terminate waste retrieval activities is driven by a variety of factors many of which are independent of the drivers associated with Criterion 3. At the same time, it is the position of NRC staff that technology selection should consider long-term risk of HRRs.

3.7 Removal to the Maximum Extent Practical (MEP)

In its basis document (DOE/SRS-WD-2010-001) and supplemental information (SRR-CWDA-2011-0005), DOE presents historical information on its waste retrieval activities. While Section 3116 does not specify numerical remedial objectives and does not require DOE to develop any such removal goals, DOE estimates that over 99 percent of the total activity inventory based on a starting point of maximum operational historical inventory is expected to be removed prior to closure. Cs-137 and Sr-90 alone account for approximately 99 percent of the HRR inventory in FTF. DOE anticipates that cleaning will essentially remove all of the soluble Cs-137, and based on waste removal activities to date, approximately 99 percent or more of the total waste volume (which would remove 99 percent of the Sr-90). The 99 percent removal efficiency is not intended to imply a correlation between 1 percent of the historical maximum and acceptable facility risk. Furthermore, individual tanks or structures may not achieve this level of removal. Instead of establishing specific goals such as volume, mass, or activity limits for the residuals in cleaned tanks, DOE will follow a process to determine when to terminate cleaning operations based on the practicality of additional removal, which is described in the following section.

3.7.1 Cleaning Process and Criteria for Termination of Cleaning Operations

DOE's approach consists of the following phases: initial technology selection, technology implementation, technology execution, technology effectiveness evaluation, and additional technology evaluation. In response to NRC staff comment CC-MEP-2, DOE provided details on the approach used for each of the tanks and the ancillary structures (DOE, 2011).

Each stage is briefly summarized below, and a figure from SRR-CWDA-2011-00091 (SRR, 2011) that illustrates DOE's approach is replicated in Figure 3-3.

1. During initial technology selection, DOE selects an optimal removal technology with an emphasis on the removal of HRRs.

2. In the technology implementation stage, DOE develops an Operating Plan that describes the cleaning process, anticipated end states, identifies specific metrics and data collection requirements for indicating effectiveness (e.g., radiation levels on transfer lines, solids concentration removed), and data collection requirements.

3. The technology execution phase continues until the removal is no longer effective.

4. DOE then formally evaluates the effectiveness of the technology and documents why the technology is no longer effective.

5. DOE then considers the practicality of continuing cleaning with upgraded systems or additional technologies.

The formality of the documentation associated with technology selection may vary because DOE may rely on a previously performed evaluation where conditions are similar. If DOE initiates a new technology evaluation (Step 4 in the approach above), DOE will employ a formal analysis based on weighted decision criteria similar to the following steps:

1. Identification of the communities of practice to be surveyed for viable technologies.

2. Identification of removal technologies (HRR-specific removal technologies as well as overall volume reduction technologies.)

3. Identification of criteria that will be used to compare the various removal technologies (i.e., removal capability, costs, technical maturity and complexity.)

4. Evaluation of technologies against the selected criteria.

Once it is no longer practical to continue its cleaning efforts, DOE will review the available information with the South Carolina Department of Health and Environmental Control (SCDHEC) and EPA and, if the three agencies (DOE, SCDHEC, EPA) concur, DOE will suspend waste removal activities. Then, DOE will perform final characterization of the tanks or ancillary structures, evaluate the costs and benefits of additional clean-up of utilizing the final inventory, and document the results in a removal report. The report will document the revised dose estimates using actual inventories from all waste tank systems that have been removed from service and knowledge gained will be used to adjust the estimated inventory of those waste tank systems that have not yet been removed from service. This approach has been completed for Tanks 18 and 19 and partially completed for Tanks 5 and 6, as described in the following section.

3.7.2 Removal to the MEP for Tanks That Have Undergone Heel Removal

To date, only Type IV Tanks 17 and 20 have undergone closure. Tanks 17 and 20 were cleaned, removed from service in 1997, and grouted prior to the passage of the NDAA and are therefore, not the subject of DOE's basis document (DOE/SRS-WD-2010-001), although the dose contributions from Tanks 17 and 20 are evaluated in DOE's PA. Tanks 18 and 19 have undergone heel removal campaigns and the final inventories have been characterized, but these tanks have yet to be grouted and closed as of the date of this TER. Tank 5 and 6 have

also undergone heel removal campaigns, but have not been finally characterized as of the date of this TER.

3.7.2.1 Tanks 18 and 19

Figures 3-4 and 3-5 show the timelines for cleaning activities associated with Tanks 18 and 19. Following bulk waste removal in Tank 19, DOE selected the Flygt Mixers as the best available technology for heel removal activities (PIT-MISC-0040). Heel removal activities using Flygt Mixers were carried out in 2001 for Tank 19. DOE selected a different technology for Tank 18 (ADMP) and in 2003, used this technology for heel removal (WSRC-RP-2001-00024). Waste retrieval with these selected technologies proceeded until it was no longer practical to continue using these technologies.

Figure 3-3 DOE Approach for Documenting Radionuclide Removal to the MEP (SRR-CWDA-2011-00091)

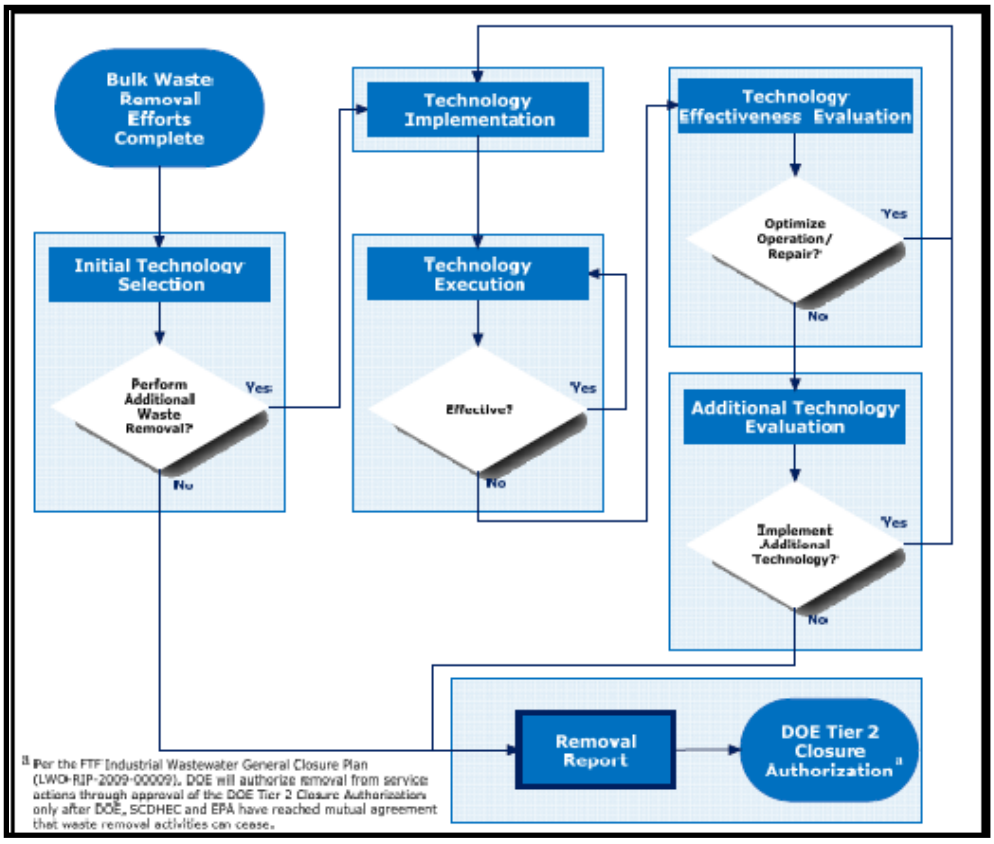


Figure 3-4 Tank 18 Historical Timeline

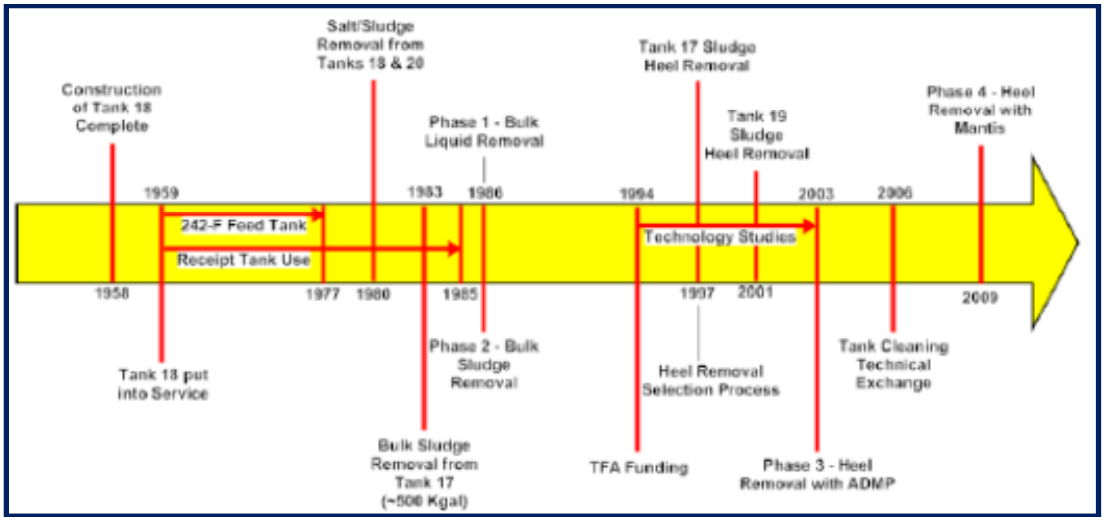
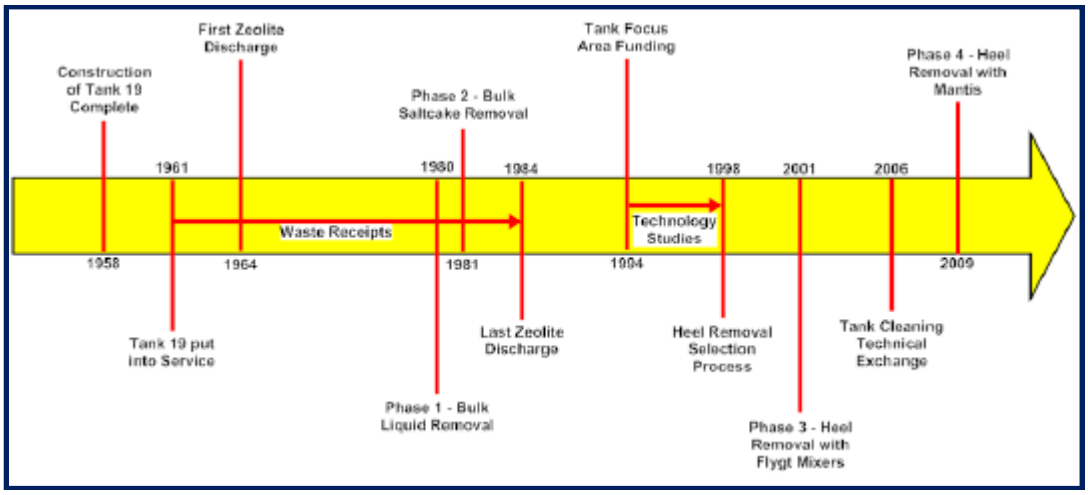


Figure 3-5 Tank 19 Historical Timeline



Three Flygt Mixers were used in Tank 19 to agitate sludge fines promoting transfer to an adjacent waste tank. However, several issues arose with the Mixers: (i) the diameter of the blade had to be reduced to fit in the tank riser reducing its hydraulic power; (ii) the mixers did not exhibit the long-term durability that had been observed in commercial applications; and (iii) the zeolite layer was more difficult to break up than the material used during testing. However, some lessons learned were carried forward for other technologies (e.g., the oscillating slew gear and the floor-mounted foot support that were used with the later SMP designs).

The ADMP used in Tank 18 was like the Standard Slurry Pump (SLP), but it used gas instead of liquid, had a higher horsepower, and a greater theoretical cleaning radius (50 ft). However, it also did not perform as expected. The principal difficulty was removal of a mound near the southwest riser that had formed because, during Phase 1 and 2, no SMP was installed in the

southwest riser, and the remaining three mixer pumps were limited in their ranges. This left the southwest area of the tank with "difficult to remove" undisturbed sludge. DOE did not install an SMP in the riser closest to this mound because the evaporator feed jet and gravity drain line in was fed through this riser. There was also a smaller mound remaining in the north region of the Tank 18 (SRR-CWDA-2011-00091, (SRR, 2011)).

In 2006, an alternative vacuum technology, the Mantis technology, was identified and found practical for use in Tanks 18 and 19 (SRR-CWDA-2011-00091). The Mantis technology was first used to clean Tank 19, and subsequently used in Tank 18. The Mantis technology was made more effective during implementation and development by adjusting sprays, attempting various vacuum patterns, using the hose-cable bundle to drag the solids, and by turning off the sprays when possible. The predicted operational time to clean the estimated solids was approximately 125 hours for Tank 19 and 36 hours for Tank 18. On several occasions, the Mantis became ineffective during waste retrieval due to failure of one or more of its components. In those cases, DOE evaluated the costs of repairing the Mantis versus benefits of additional removal and determined that it was practical to make repairs and continue the heel removal operations. Following additional removal operations and after a total of approximately 556 operating hours in Tank 19 and 459 operating hours in Tank 18, each Mantis became ineffective to the point that DOE decided that it was no longer practical to replace or repair each Mantis.

The volume reduction estimates by DOE as a result of cleaning are reproduced in Table 3-5, Figure 3-6 and Figure 3-7. As indicated in the footnote to Table 3-5, DOE thinks that the inventory remaining in Tank 18 after Phase 3 heel removal 16.3 m^3 (4,300 gal) was underestimated due to the difficulty in determining the depth of solids beneath a 3.8 cm (1.5 in) level from the video inspection because it did not provide clear views below the surface liquid. The heel removal in Phase 4 later indicated that 16.3 m^3 (4,300 gal) was an underestimation (SRR-CWDA-2011-00091).

DOE determined that it was no longer practical to continue operations with the Mantis based on the following factors:

- Visual Observation: Visual observation indicated that the mounds of material which existed prior to heel removal with the Mantis had been removed. DOE provided photos of the interior of both tanks after cleaning with the Mantis.
- Technology Limitations: Due to the physical features of the suction head and blades that corralled waste up into the suction head, the Mantis was more effective in areas with waste in mounds and lost effectiveness when only a thin layer of waste was present. Specifically, the screen over the Mantis suction inlet resulted in a 1.27 cm (0.5 in) air gap between the bottom of the waste tank and the Mantis suction inlet. This air gap limited the Mantis's ability to remove waste below the 1.27 cm (0.5 in) level in the tank. The blades also experienced interference with the lifting plates on the floor of each tank, which lead to their deterioration. As stated earlier, the lifting plates are artifacts of the construction of the tank; the 0.30m (1 ft) x 0.30 m (1 ft) x 1.27 cm (0.5 in) tall plates and are arranged in a grid pattern across the bottom of the tank.

- Significant Equipment Degradation: Several repairs were made to the Mantis which effectively doubled its operating life. However, eventually the forward blades were no longer effective and it also became apparent that repairs were no longer effective.
- Impacts on Tank Space and the Liquid Waste System: As the waste was leveled out into thin layers, the Mantis required increasing amounts of water to remove waste. When operations ceased, 950 m^3 (250,000 gal) of waste water had been generated in comparison to the 568 m^3 (150,000 gal) that was expected. The waste water was being stored in Tank 7, which had only 95 m^3 (25,000 gal) excess capacity at the end of operations. It was expected that continued operation would produce 30 m^3 (8,000 gal) per day, so Tank 7 would have soon run out of capacity if operations continued. In addition, equipment from the wastewater system was needed for processing a separate sludge batch operation. Sludge Batch 6 was being prepared for feed to the DWPF at this time, which required use of the 2S and 3H evaporators. If these evaporators were to be used for Tanks 18 and 19, as opposed to the sludge batch operations, it could have led to a shut-down of the DWPF.
- Transfer Line Radiation Readings: At the conclusion of cleaning, DOE took radiation measurements of 10 areas and compared them to measurements taken during cleaning operations. The radiation monitoring data provided by DOE for Tanks 18 and 19 are reproduced in Figure 3-8 and Figure 3-9 below.

Replacement of the Mantis and alternative technologies were also considered, along with costs, including worker dose, and other potential impacts (SRR-CWDA-2009-00030). DOE estimated that it would cost $2.8 M to replace the Mantis with similar equipment, and that worker doses were estimated to be from 2.5-12 person-mSv (0.25-1.2 person-rem) per tank. The lower estimate is for tasks to remove the existing Mantis and equipment and install a new Mantis in the existing riser, while the higher dose is to drill a new riser and install a new Mantis and equipment in the new riser (SRR-CWDA-2011-00091). Given the high costs, DOE determined that deployment of additional cleaning technologies was impractical.

Per the requirements of the FFA, DOE briefed the EPA on September 24, 2009, and SCDHEC on October 1, 2009, on the results of the Mantis cleaning of Tanks 18 and 19 (SRR-CWDA-2009-00030). Upon agreement by all three agencies, DOE proceeded with the final characterization of Tanks 18 and 19, which is described in Section 3.2 of this TER. DOE estimated that 99 percent of the original activity of the HRRs was removed. DOE stated that the lower removal rate of Pu-238 (17 percent) was due to the Pu-238 that remains in the scale and corrosion film on the tank walls. DOE reported that the samples from both the lower wall and the upper wall sample analyses returned unusually high concentration results for Pu-238. The concentrations for all other radionuclides (including other plutonium isotopes) decreased relative to the floor samples. Since, chemically all of the Pu isotopes behave the same, DOE expected that all Pu isotopes, including Pu-238, would have similar relative concentrations (floor versus wall concentrations).

Table 3-5 Tank 18 and 19 Volume Reduction Estimates (SRR-CWDA-2011-00091)

Inventory	Tank 18 Waste		Waste Tank 19 Waste	
	Approximate Gallons Remaining	Cumulative % Removed	Approximate Gallons Remaining[b]	Cumulative % Removed
Inventory Prior to Waste Removal	1,300,000	0	1,300,000	0
Inventory at Completion of Phase 1 Campaign (Supernatant Removal)	550,000	57.7	1,100,000	15.4
Inventory at Completion of Phase 2 Campaign (Bulk Removal - Solids for Tank 18 and Salt for Tank 19)	37,000	97.2	33,000	97.5
Inventory at Completion of Phase 3 Campaign (Heel Removal with ADMP for Tank 18 and Mixer Pumps for Tank 19)	4,300[a]	99.7	15,000	98.8
Inventory at Completion of Phase 4 Campaign (Heel Removal with Mantis)	4,000	99.7	2,000	99.8

[a] Inventory following Phase 3 is believed to have been underestimated. See Section 5.2.4.3 of the basis document DOE/SRS-WD-2010-001.
[b] Volumes for Phases 1,2 and 3 represent approximate volume of wet solids. Other volumes represent total waste volume. (DPSP-84-17-7, WSRC-TR-2002-00052, U-ESR-F-00042)
Multiply by 0.0038 to convert gallons to m^3

Figure 3-6 Tank 18 Volume Reduction (SRR-CWDA-2011-00091)

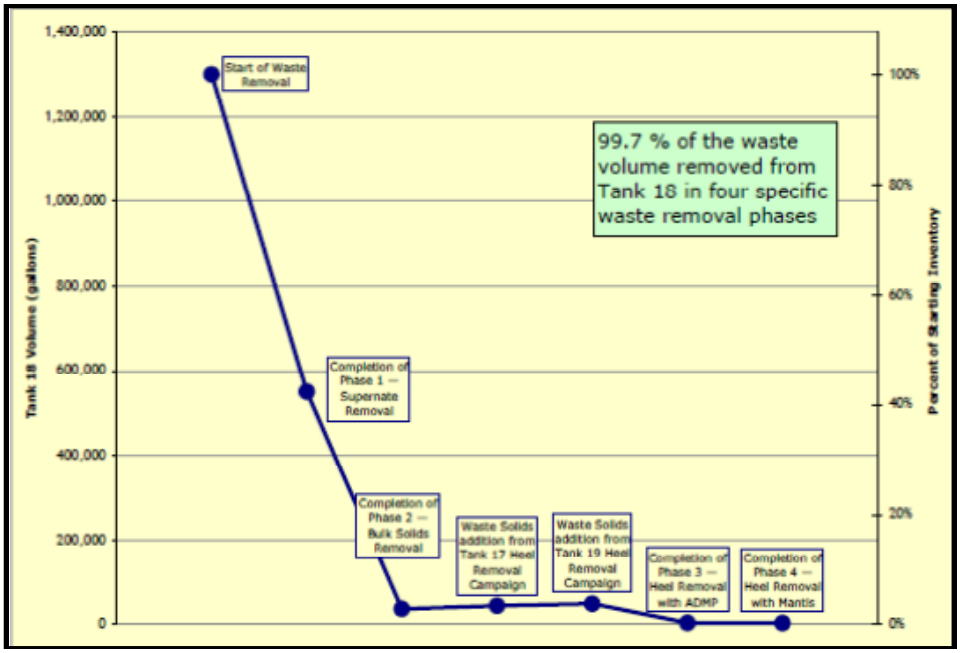


Figure 3-7 Tank 19 Volume Reduction (SRR-CWDA-2011-00091)

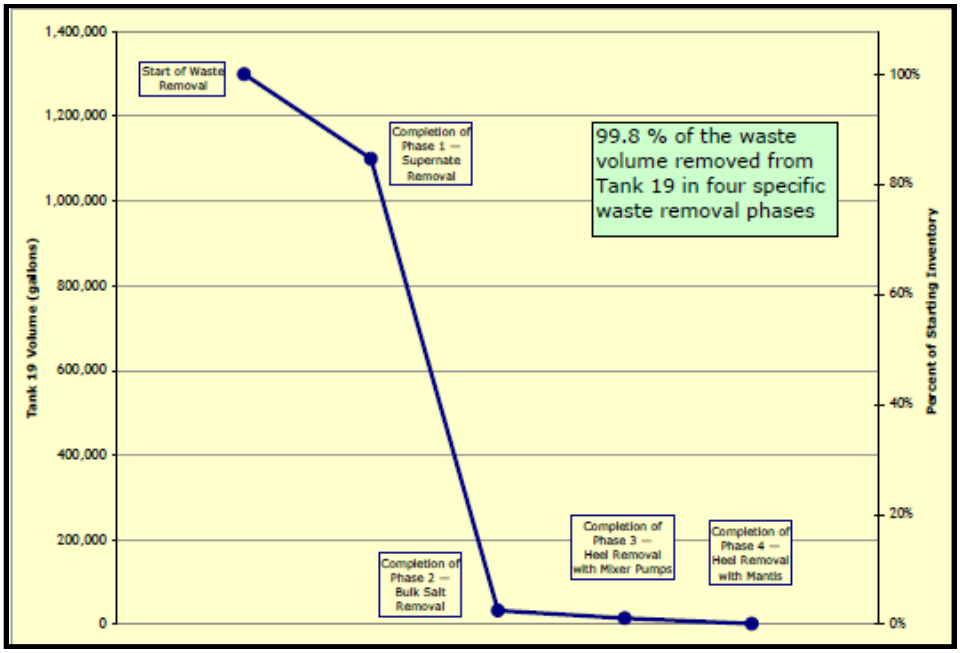


Figure 3-8 Tank 18 Transfer Line Radiation Reading

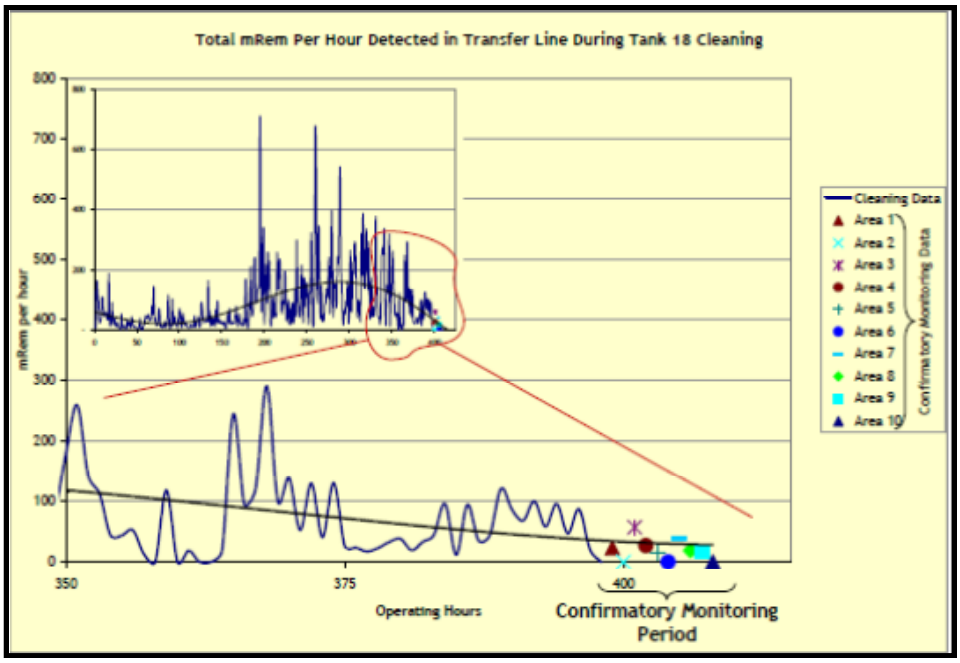


Figure 3-9 Tank 19 Transfer Line Radiation Reading

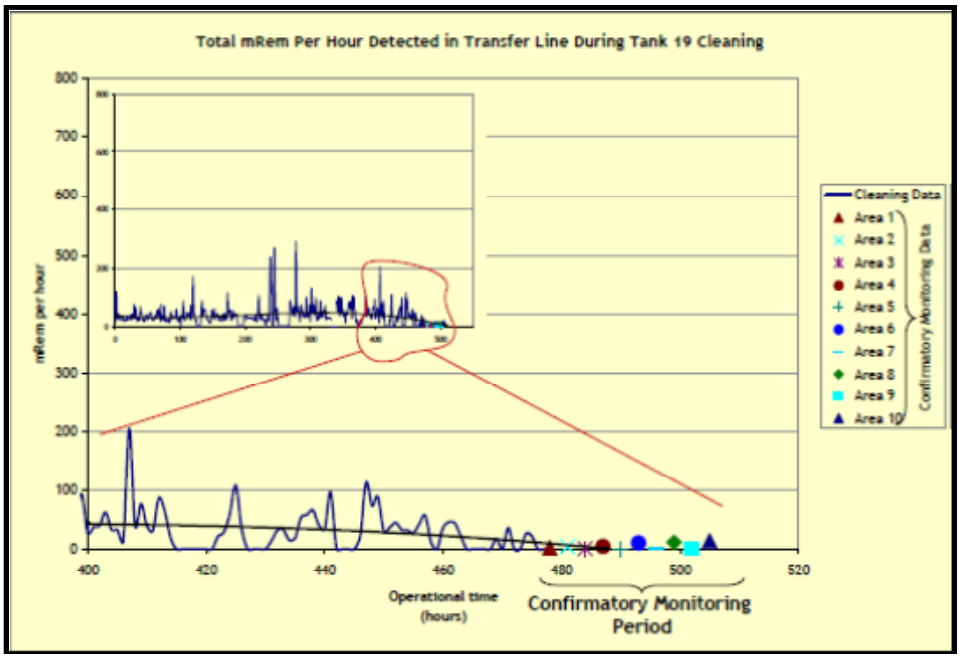


Table 3-6 Tank 18 Percentage of HRRs Removed

Radionuclides	Tank 18 Original Inventory as of 1986 (Ci)	Tank 18 Residual Inventory as of 2010 (Ci)	% Removed
Sr-90	$2.2x10^{5}$	$3.2x10^{3}$	98.5
Tc-99	$2.4x10^{2}$	$9.0x10^{-1}$	99.6
I-129	$5.7x10^{-2}$	$1.7x10^{-4}$	99.7
Cs-137	$9.5x10^{5}$	$1.2x10^{4}$	98.7
U-234	$6.2x10^{0}$	$3.1x10^{-1}$	95.0
Np-237	$2.4x10^{0}$	$1.9x10^{-1}$	92.1
Pu-238	$1.7x10^{3}$	$1.4x10^{3}$	17.6
Pu-239	$3.2x10^{3}$	$2.8x10^{2}$	91.3
Pu-240	$8.3x10^{2}$	$6.5x10^{1}$	92.2
Am-241	$8.3x10^{2}$	$1.6x10^{2}$	80.7
Total	**$1.2x10^{6}$**	**$1.7x10^{4}$**	**98.6**

Table 3-7 Tank 19 Percentage of HRRs Removed

Radionuclides	Tank 19 Original Inventory as of 1980 (Ci)	Tank 19 Residual Inventory as of 2010 (Ci)	% Removed
Sr-90	$4.4x10^{5}$	$8.9x10^{0}$	100.0
Tc-99	$4.5x10^{2}$	$3.8x10^{-1}$	99.9
I-129	$5.7x10^{-2}$	$1.7x10^{-4}$	99.7
Cs-137	$6.5x10^{5}$	$5.4x10^{3}$	99.2
U-234	$1.4x10^{0}$	$4.8x10^{-3}$	99.7
Np-237	$3.0x10^{-1}$	$1.5x10^{-3}$	99.5
Pu-238	$5.3x10^{3}$	$3.6x10^{0}$	99.9
Pu-239	$7.4x10^{2}$	$4.0x10^{0}$	99.5
Pu-240	$1.6x10^{2}$	$9.8x10^{-1}$	99.4
Am-241	$1.4x10^{2}$	$2.6x10^{0}$	98.1
Total	**$1.1x10^{6}$**	**$5.4x10^{3}$**	**99.5**

Figure 3-10 shows the current status of the volume distribution based on material mapping. As is apparent, the distribution of volume and the amount of waste remaining in the two tanks differ. The inventory volume remaining in Tank 18 is estimated to be twice that of Tank 19. DOE stated that the heterogeneity of volume both in each tank and between tanks is a cumulative result of the differences in technology effectiveness, largely influenced by access limitations, and placement of the SMPs.

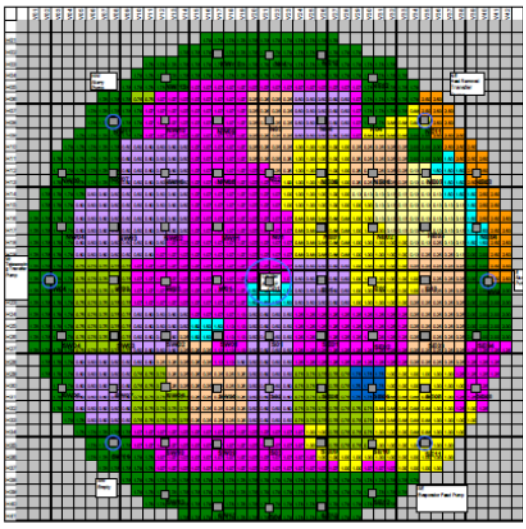
As described in Section 3.2, the Mantis, in addition to serving as a cleaning tool, was also employed to help survey the volume in the tanks: its wheels served as relative objects in photographs where the lifting plates were not visible. DOE used the Mantis to successfully

survey the central, northeastern, eastern and southeastern regions of Tank 18 (see Figure 3-10). In these regions, DOE confirmed that the Mantis's wheels were in contact with the bare waste tank floor, which enabled accurate measurements. However, the Mantis was unable to reach the northwestern, western, and southwestern regions due to a loss in traction caused by material building up inside the Mantis's wheels or drag weight of hose and tether system. Therefore, DOE used a weighted mean approach for assigning depths to these regions.

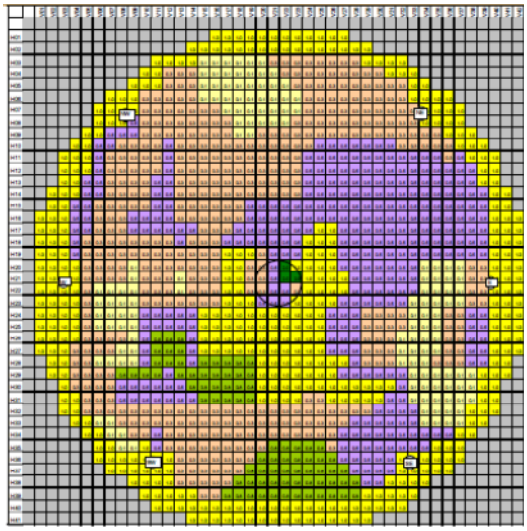
DOE analyzed the impacts of the final inventory on the PA results in a SA (see Section 3.2); the results of the SA are summarized below (SRR-CWDA-2010-00124). The baseline peak dose presented was determined with the Best Estimate Tank 18 and Tank 19 inventories using the GoldSim FTF model in deterministic mode utilizing the base case input parameters. In addition to the results shown in the table, the peak groundwater pathway dose in 20,000 years was presented as decreasing slightly (from 0.18 to 0.17 mSv/yr (18 to 17 mrem/yr)).

Figure 3-10 Tank 18 and 19 Volume Distribution Following Mantis Cleaning (SRR-CWDA-2009-00030)

Tank 18

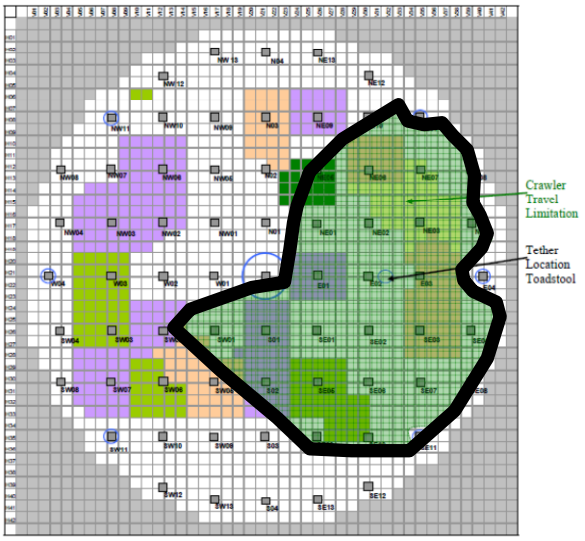


Tank 19



LEGEND
inches
2.50
2.25
2.00
1.75
1.50
1.25
1.00
0.75
0.50
0.25
0.125

Tank 18 Mantis Crawler Travel Limitation (shown in green overlay and outlined in bold)
U-ESR-F-00041



Tank 19 Projected Mantis Crawler Limitation (shown in green overlay and outlined in bold)
U-ESR-F-00042

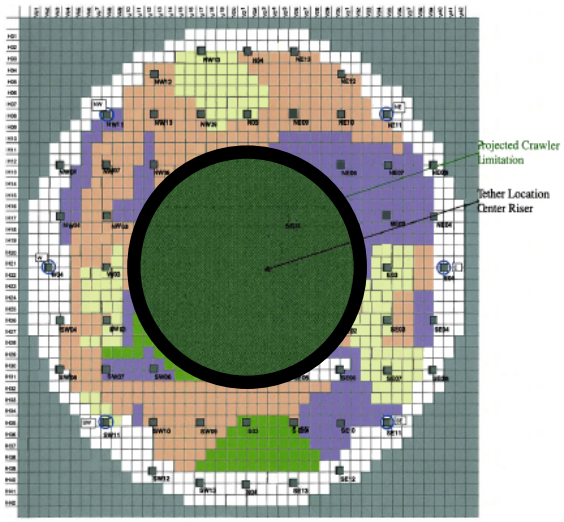


Table 3-8 Summary of Special Analysis (SA) Results for Tanks 18 and 19

	PA Rev. 1 Peak Within 10,000 Years (projected Tank 18/19 inventories)			Special Analysis Peak Within 10,000 Years (actual Tank 18/19 inventories)		
	All-Pathways Dose (mrem/yr)	Ground-water Pathway Dose (mrem/yr)	Air Pathway Dose (mrem/yr)	All-Pathways Dose (mrem/yr)	Ground-water Pathway Dose (mrem/yr)	Air Pathway Dose (mrem/yr)
100 meters from FTF	2.5 at ~ year 10,000	2.3 at ~ year 10,000	0.2	3.4 at ~ year 10,000	3.2 at ~ year 10,000	0.2
At Seepline	0.09 at ~ year 10,000	0.04 at ~ year 10,000	0.05	0.12 at ~ year 5,600	0.07 at ~ year 5,600	0.05

Note: Multiply by 0.01 to convert mrem/yr to mSv/yr

In addition to analyzing dose impact with the Best Estimate of the final inventories, DOE also performed: (i) deterministic sensitivity analysis; and (ii) a probabilistic analysis to assess uncertainty in the inventory. For the deterministic assessment, DOE calculated the peak dose using a higher and a lower inventory projection. The lower inventory was set at 50 percent of the Tank 18 and Tank 19 inventories that were assumed in the deterministic base case. The higher inventory was set by using the Reasonably Conservative Estimates (SRR-CWDA-2010-00117, SRR-CWDA-2010-00118). The lower inventory peak dose is 55 percent of the baseline peak dose in 10,000 years and is 92 percent of the baseline peak dose in 20,000 years. The higher (e.g., reasonably conservative) inventory peak dose is 107 percent of the baseline peak dose in both 10,000 and 20,000 years. The probabilistic model used inventory multipliers discussed in Section 3.2. The peak dose results for 1,000 realizations showed the 95th percentile of the peak dose in 10,000 years to be 0.35 mSv/yr (35 mrem/yr) versus 0.77 mSv/yr (77 mrem/yr) in the FTF PA Rev. 1 (SRS-REG-2007-00002, Rev. 1). Note that the 95th percentile value is lower than that calculated in the PA because many of the inventory distributions from the FTF PA Rev. 1 were narrowed when DOE used the 2010 residual characterization data.

3.7.2.2 Tanks 5 and 6

Tanks 5 and 6 have also undergone heel removal, but have not yet been finally characterized. Type I tanks, such as Tank 5 and 6, are the most challenging to clean due to internal cooling coils, support columns, and limited access ports. During mechanical bulk removal mounds formed in the northeast and southwest regions of Tank 6 due to cooling coil obstructions and limited mixing zones in the tank. DOE used hydrolancing to break up the southwest mound in Tank 6, but it could not reach the northeast mound. Also, the southwest mound reformed after another mechanical sludge removal campaign. In Tank 5, a mound formed in the southeast portion of the tank (SRR-CWDA-2010-00157). In 2005, after mechanical removal campaigns using submersible mixer pumps removed the bulk of the sludge, a heel of about 12.9 m^3 (3,400 gal) remained in Tank 5 and 22.8 m^3 (6,000 gal) in Tank 6.

Chemical Sludge Removal (CSR), using Oxalic Acid carried out in 2008, reduced the residual sludge heel to approximately 12.5 m^3 (3,300 gal) in Tank 5, and 13.3 m^3 (3,500 gal) in Tank 6. The chemical removal was carried out in three campaigns. Table 3-9 outlines the anticipated and measured volume removal effectiveness of the campaigns. DOE anticipated that, under ideal mixing conditions, a total of 90 percent of the remaining sludge would be dissolved by the oxalic acid. In reality, the chemical removal only decreased the volume of the sludge heel by 4 percent in Tank 5 and 42 percent in Tank 6. DOE states that the true effectiveness is difficult to measure due to the formation of metal oxalates. Oxalic Acid is a very strong organic acid that readily combines with metal ions such as iron, calcium, and magnesium to form metal oxalates. A layer of oxalates formed as a result of limited mixing in the first two campaigns. Therefore, Campaign 3 was modified to combine the third bulk OA addition with the OA spray wash to minimize the formation of oxalates and increase effectiveness. The difference in effectiveness between the two tanks has to do with the pH of the first acid campaign, the mixing conditions in the respective tanks, the number of pumps, and the length of time the pumps were operated. Most of the sludge mass remaining in the tanks was reported to be iron and nickel (SRNL-STI-2009-00492 and SRNL-STI-2009-00493).

Table 3-9 Chemical Removal Anticipated and Actual Volume Removal (SRR-CWDA-2011-00033 and SRR-CWDA-2011-00005)

Removal Stage	Anticipated Change in Volume	Tank 5 Measured Change in Volume	Tank 6 Measured Change in Volume
CSR Campaign 1	-70%	-20%	-60%
CSR Campaign 2	-50%	+31%	+36%
CSR Campaign 3	-30%	-8%	+8%
Total Change	**-90%**	**-4%**	**-42%**

*positive values indicate an increase in total solids due to metal oxalates formation

Because the acid removed some radionuclides more effectively than others and the formulation of metal oxalates led to volume increases, volume reduction is not a direct indication of HRR removal.

Table 3-10 lists the effectiveness of the CSRs in total in terms of removal of selected radionuclides. DOE attributes the differences in removal effectiveness (e.g., Tc-99, iron, plutonium, americium, and curium) to the different chemical compounds that existed in each tank, the sludge heel not being homogeneous in one or both of the tanks, or the lower pH used in Tank 6 (SRNL-STI-2009-00492 and SRNL-STI-2009-00493).

Table 3-10 Fraction of HRRs Removed with Chemical Sludge Removal (SRNL-STI-2009-00492 and SRNL-STI-2009-00493)

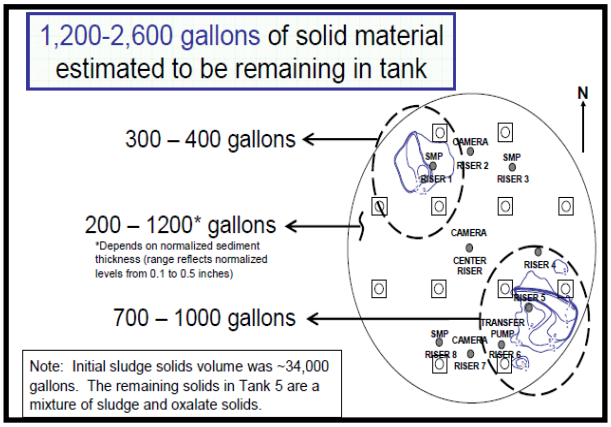
HRRs*	% Removal from Tank 5	% Removal Tank 6
Sr-90	83%	87%
Co-60*	20%	87%
Tc-99	41%	2%^
Cs-137	>90%	87%

Uranium Isotopes	>90%	>95%
Np-237	70%	73%
Pu-238	5%	17%
Pu-239	17%	19%
Pu-239/240	6%	15%
Am-241	1%	7%

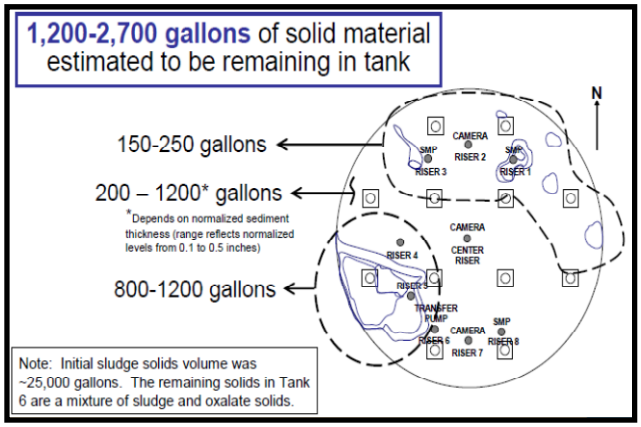
*Not all HRRs are listed: fraction of I-129 removed was not reported and Co-60 is not an HRR but is shown due to the difference in effectiveness of removal.
^Although information in this table suggests that Tc removal following oxalic acid washing was low, DOE provided additional information regarding its expectation that final Tc inventories for Tanks 5 and 6 will be less than 3.7 x 10^{10} Bq (1 Ci) (see July 20, 2011 meeting summary at ML112070026).

DOE took in-process liquid samples after each campaign, which showed that additional chemical removal would be ineffective, so an alternative technology was pursued. After evaluating 14 mechanical and 4 chemical technologies, DOE chose a method called Mechanical Feed and Bleed (MFB) that involved the use of three mixing pumps with liquid additives during transfer; DOE first used this method in 2010. DOE estimates that the final volume after MFB to be approximately 7.6 m^3 (2,000 gal) in each tank. Final mapping and characterization has not yet been completed. Figure 3-11 illustrates the estimated volume based on preliminary mapping. In Tank 5, DOE determined that additional MFB would not be able to reduce: (i) the mounds under Riser 5 because it remained after repeated runs; or (ii) the mound under Riser 1 because the SMP in Riser 1 was resting 16 inches off the tank floor because of cooling coil interference (SRR-CWDA-2011-00033). In Tank 6, the mound under Riser 5 in the southwest corner remained due to limited mixing zones (SRR-CWDA-2011-00005).

Figure 3-11 Tanks 5 and 6 Residual Volume



Tank 5 Residual Volume



Tank 6 Residual Volume

The Tank 5 and 6 annuli were inspected and cleaned using a wall crawler in 2009. A total of 44 leak sites for Tank 5 and a total of 11 leak sites for Tank 6 were discovered. Approximately 0.38 m^3 (100 gal) of dried salt solids were present in the Tank 6 annulus due to the leak sites, but less than 0.04 m^3 (10 gal) were present in the Tank 5 annulus. Water was not injected into the Tank 5 annulus due to the small amount of residual material after cleaning with the crawler; but water was injected into the Tank 6 annulus and allowed to soak before being removed to

clean the Tank 6 annulus further. A final determination of the volume of residual solids in the Tank 6 annulus is ongoing.

The end of DOE's Tank 5 and 6 removal activities is based on visual inspection, technology limitations, and impact to the liquid waste system. Visual inspections using cameras indicated significant reduction in the residual material. DOE indicated that the chemical technology isn't practical to use anymore because after a layer of oxalate coated the remaining sludge during the first two campaigns, additional chemical campaigns were expected to only increase the solids further because the layer of oxalates would impede contact between the sludge and acid. The MFB has ceased because of (i) limited capacity of the waste water system; (ii) the limit on the amount of well water available; and (iii) limited availability of Tank 7 (which was a receipt tank for the MFB for Tank 6) due to other sludge batch processing activities.

DOE evaluated alternative cleaning technologies, such as enhanced chemical cleaning or vacuum cleaning, and decided that they were inappropriate because of insufficient technology maturity, high costs, or incompatibility with in-tank obstructions (e.g., cooling coils). Several recommendations were made for the future chemical cleaning of Type I tanks. These recommendations include: washing the sludge heel and collecting a sample prior to chemical cleaning, using a lower pH, using X-ray diffraction to identify specific compounds, and ensuring sufficient mixing of the tank throughout cleaning (SRNL-STI-2009-00492 and SRNL-STI-2009-00493).

3.7.3 Removal to the MEP for Tanks That Have Not Undergone Heel Removal and Ancillary Structures

The interdependency of safety and process requirements of the various waste pretreatment facilities, and other constraints will influence the sequence of cleaning the remainder of the tanks. The ancillary equipment may be closed in conjunction with an individual tank, group of tanks, or independent of waste tank farm closures. DOE's anticipated schedule, outlined in the SRS FFA, provides dates for bulk waste removal efforts and removal from service of Type I and Type IV waste tanks. Type III and Type IIIA waste tanks will remain in service until there is no longer a need for them to support waste treatment.

DOE will follow the approach described in Section 3.7.1 of this TER for each tank or group of tanks and ancillary structures in the FTF. A final report documenting removal of the HRRs to the MEP may be written to include summaries of more than one tank or ancillary structure if several are removed from service at the same time.

DOE also provided information on the development of future technologies (V-ESR-G-00003). The Enhanced Chemical Cleaning (ECC) is an OA process that destroys oxalates before introduction to the waste receipt tank. After several days of continuous agitation, the spent solution is pumped to a waste receipt tank. The receipt waste tank holds excess sodium hydroxide to neutralize the low-pH spent solution, thus maintaining corrosion control. The prototype for ECC will be Tank 8. DOE is also testing supplemental acids mixed with OA, but further testing and evaluations are required before ECC is considered a baseline technology. DOE anticipates that the wall crawler will be able to sufficiently clean the annulus of most tanks, but a robotic manipulator arm will be needed for large deposits, like those in Tank 16.

As far as future cleaning of Type I tanks, DOE anticipates that waste removal in Tanks 5 and 6 will be representative of expected success for other Type I tanks (Tanks 1, 2, 3, 4, 7 and 8). Tanks 5 and 6 were sludge tanks. Tanks 4, 7, and 8 contain similar sludge and the designs are similar. For Tanks 1, 2, and 3, the majority of the waste remaining in these tanks is saltcake, which is easier to remove than sludge; removal of waste from these tanks is expected to be similar to or better than Tanks 5 and 6. Tank 7 is unique among the FTF Type I tanks because it has been used to consolidate sludge heels (e.g., zeolite-laden solids that originated in Tank 19 and passed through Tank 18). However, this material has been passed through a Waste Mixing Chamber (WMC), which reduced the particle size to less than 38 microns. Therefore, DOE does not anticipate future transfer of zeolite solids from Tank 7 to pose unique challenges. Still, it should be noted that because final inventories have not been provided, the true effectiveness of waste removal in Tanks 5 and 6 has not yet been demonstrated.

Considering the final radionuclide inventory, DOE will conduct a SA to ensure that the PA conclusions are still valid. The same approach used to perform the SA for Tanks 18 and 19 will be used for other cleaned tanks in the future. However, some of the hard-to-detect analytes might not be included in future sampling if they are of low risk-significance (e.g., Cl-36). The cost-benefit analysis that will serve as the rationale for evaluating removal to the MEP will be informed by the SA as well as the costs and risks associated with additional cleaning. The cost-benefit analysis for Tanks 18 and 19 is described in the following section.

3.7.4 Costs and Benefits of Additional Waste Retrieval for Tanks 18 and 19

DOE qualitatively analyzed the following technologies, and determined their use to be impractical:

Mechanical Cleaning:

- Ineffectiveness: With the large particle-sizes of the residuals remaining in the tanks (i.e., mostly heavier, rapidly settling spent zeolite), it is unlikely that significant additional waste removal would occur using SMPs due to the need to shutdown the mixing in the tank at a liquid level of 30 inches and the fast-settling nature of the remaining residuals.

- Schedule Impacts: New SMPs would have to be procured, which would take 12-18 months. A total of four SMPs would be required to fully eliminate the limited mixing zones, which would divert SMPs needed for other tanks, and would result in delays in waste removal activities in other tanks.

Chemical Cleaning:

- Ineffectiveness and Downstream Waste Impacts: The OA cleaning is relatively ineffective at removing spent zeolite, and would produce potentially detrimental downstream impacts to the Liquid Waste System.

- Schedule Impacts: ECC would not be a viable candidate for removing additional waste inventories from Tanks 18 and 19, and prioritizing the deployment of ECC in Tanks 18

and 19 would delay the waste removal and tank closure activities for other old-style tanks.

Vacuum Cleaning:

- Worker Risk: Repair of the Mantis (e.g., replacement of the blades) is not an option due to the associated high occupational exposure.

- Financial Cost: Building a new upgraded Mantis (e.g., redesigned rubber squeegee device, suction head, drive mechanism, strainer, etc.) would be too costly, and the effectiveness of an upgraded Mantis is unpredictable.

DOE performed a quantitative cost-benefit analysis assuming an upgraded Mantis could be developed and deployed in Tanks 18 and Tank 19 that could remove 50 percent of the existing residual waste. A technical basis was not provided for the assumption of 50 percent removal. DOE expected that an upgraded Mantis would be the least costly of all the potential technologies so it was chosen for the quantitative analysis.

The analysis concluded the following:

- DOE estimated that the development, design, testing and deployment would take at least two years.

- The cost of development and manufacture of an upgraded Mantis would be $8.0M (see Table 3-11).

- The cost of development and manufacture would have an adverse impact on the Liquid Waste System and delay cleaning activities in the remaining Type I tanks.

- Additional worker exposure is estimated to range from 2.5 to 12 person-mSv (0.25 to 1.2 person-rem).

- Assuming 50 percent removal, the expected reduction in dose over a period of the next 10,000 years would only be approximately 0.01mSv/yr (1.3 mrem/yr) to a member of the public living 100 m (330 ft) from the FTF (SRR-CWDA-2010-00124).

Table 3-11 Financial Cost of Newly Developed Upgraded Mantis

Description	Estimated Cost $k
Engineered Equipment	$4,050
Water Wash Tool	183
Transfer Line	523
Mantis Installation	119
Construction Equipment & Material	261
Demobilization	198
Execution	178

Sampler Crawler	42
Total Support Cost[a]	1246
Sampling Support	1267
Total Estimated Project Cost	**$8,067**

[a]The Total Support Cost includes costs for support provided by Design, Design Authority, Operations, Engineering, Maintenance, Craft, Training, Procedures, Environmental Compliance Authority, Radcon, Camera Crew and Generator Certification Official personnel (Source: SRR-CWDA-2011-00091 in Table 7.1-1)

3.8 NRC Evaluation of Removal to Maximum Extent Practical (RMEP)

3.8.1 NRC Evaluation of MEP Demonstration for Tanks 18 and 19

While DOE provided a number of compelling reasons why HRRs have been removed from Tanks 18 and 19 to the MEP, NRC staff has unresolved issues associated with DOE's assessment. One of NRC staff's primary concerns rests with the lack of consideration of the potential benefits of additional removal for HRRs such as Pu-239, that are expected to lead to the greatest risks associated with the FTF over longer periods of performance. DOE discusses how removal technologies were tailored to specific tank characteristics for Tanks 18 and 19 (e.g., Mantis was effective due to the absence of cooling coils, OA was ineffective due to the presence of zeolite) (SRR-CWDA-2011-00091), and DOE asserts that technology selection emphasizes HRR removal, but emphasis on HRR removal is not apparent. For example, Tank 18 is arguably the single-most risk-significant tank in FTF, contributing to a more certain[(1)] overall peak dose of around 3 mSv/yr (300 mrem/yr) at times beyond the period of compliance in DOE's base-case analysis from Pu-239 (SRR-REG-2007-00002, Rev. 1). Yet, it is not apparent that the long-term risks associated with Pu-239 in Tank 18 were considered in the cost-benefit analysis or in the selection of alternative technologies.

It is significant to note that estimates of Pu-239 activity remaining in Tank 18 of 140 Ci prior to Mantis operation estimated in DOE-WD-2005-002, Rev. 0 is lower than the final best estimate of 280 Ci (and average sample value of 200 Ci) following Mantis operation (and the volume reduction was estimated at 7 percent). While it is obvious that at least some small fraction of Pu-239 was removed from Tank 18 during Mantis operation and a portion of the final inventory was associated with the wall, due to the uncertainty in the inventory and volume estimates prior to and following Mantis operation, the effectiveness of the Mantis technology in removing HRRs from Tank 18 is difficult to discern. Additionally, the lack of adequate baseline information on Tank 18 residual waste may have hampered efforts for more efficient removal of HRRs from the tank. For example, the initial DOE estimate of 36 hours of Mantis operation needed to clean Tank 18 reflects an incomplete understanding of the amount of effort needed to remove residual waste from Tank 18. The significant under-estimate in operational hours is potentially due to (i) poor estimates of the amount and distribution of residual material remaining in Tank 18, and (ii) the ability of the Mantis to corral and suction residual waste in the tank. As Figure 3-10 suggests, Mantis travel limitations may also have hampered efforts to remove large quantities of material remaining in the western portion of the tank, although the NRC staff was unable to confirm this with DOE prior to finalization of this TER. DOE may have made different decisions regarding Mantis placement or operations if better estimates of residual waste distributions were available and/or information regarding the residual risks associated with Pu-239 in Tank 18 were considered.

DOE does not fully evaluate the basis for differences in removal effectiveness both within each tank and between Tanks 18 and 19, and how these differences are accounted for in the determination to cease cleaning operations. As shown in Figure 3-10, there is variability in depth across the floors of each tank, and also a considerable difference in how effective the technologies were for Tank 18 in comparison to Tank 19. For example, there is a greater amount of residual material in the center of Tank 18 as well as to the northwest and south. The mound in the north region of Tank 18 that formed after the use of the ADMP may have contributed to variability prior to using the Mantis, but DOE does not describe why the Mantis may have been less effective in this area or in the southern most part of Tank 18. As discussed above, the Mantis could have been less effective in these areas due to the travel limitations that are presented in a separate reference (U-ESR-F-00041) describing the use of the Mantis in the volume mapping (see Figure 3-10), but DOE does not draw a link between Mantis travel limitations and cleaning effectiveness. Also, the difference in effectiveness of similar cleaning technologies is surprising, especially given the similarities in the tank designs and because Tank 18 was predicted to have a much lower volume of waste than Tank 19 at the end of Phase 3. Differences between the waste in Tanks 18 and 19 that may have affected waste retrieval are not clear, and in fact some descriptions indicate that waste is identical (DOE-WD-2005-002, Rev. 0) while other descriptions indicate that the waste in Tank 19 was more amenable to retrieval (e.g., V-ESR-G-0003 describes Tank 19 waste as more sand-like compared to Tank 18). The Mantis technology was first operated in Tank 19, so it would seem that lessons learned from that operation could potentially increase performance in Tank 18. Equipment malfunctions occurred during retrieval efforts in both Tank 18 and Tank 19, but the Tank 18 Mantis was operated without the forward/downward spray or the ability to tilt the crawler for some period of operation. Furthermore, the Tank 18 travel limitations may have put additional constraints on waste retrieval. DOE does not fully describe if this variability within and between tanks was anticipated. DOE does not explain why methods to improve the effectiveness of the selected technology was apparently less successful in Tank 18.

DOE could have provided additional discussion regarding modifications to improve effectiveness or reuse of technologies in different stages of cleaning. Much of the residual material in Tank 18 is near the southwest riser because an SMP was not installed in the southwest riser since the evaporator feed jet and gravity drain line were fed through these risers. Although it may have been entirely impractical to adjust the location of these lines to make cleaning more effective, DOE does not discuss the feasibility or costs associated with this type of modification earlier in the cleaning process. Also, the Tank 18 Mantis was installed in a new riser in the east portion of the tank because removal of the ADMP from the center riser was deemed too costly; this decision may have limited access to the western parts of the tank and the impracticality of ADMP removal in light of the potential benefits of long-term risk reduction is not clearly demonstrated. Furthermore, DOE could have more fully discussed whether the residual material in the tanks would be more amenable to cleaning with another technology now that Mantis has broken up the zeolite and southwest mound. Another difference lies in the residual material on the walls of the tanks. In Tank 19, a water lance was used to wash the walls, but a similar technology was not used on Tank 18. During the washing of the walls in Tank 19, the water lance was also directed at the dispersal of mounds of solid waste on the waste tank floor in areas beneath installed waste tank equipment that the Mantis could not reach. DOE could have evaluated the potential to using the water lance for the cleaning of the walls and mound areas in Tank 18.

DOE did not fully discuss the ability of enhancements or alternative reagents to leach key radionuclides out of the heels. The NRC staff recognizes that according to DOE, for residual sludge inventories containing significant quantities of zeolite, the use of OA has adverse downstream impacts and is relatively ineffective. However, DOE is currently developing ECC, which will minimize the downstream impacts of OA and allow for additional cleaning opportunities. DOE's rationale as to why ECC would not be a viable candidate for Tanks 18 and 19 is: (i) because it is not yet ready for deployment as a baseline methodology; and (ii) doing so would delay the waste removal and closure for old-style tanks (e.g.,Type I) (SRR-CWDA-2011-00091). However, DOE did not evaluate the option of cleaning the Type I tanks prior to additional removal attempts with Tanks 18 and 19. Also, given that ECC is going to be deployed for Tank 8, it does not seem appropriate to rule it out the use of ECC based on technology maturity. In addition to ECC, DOE could mix alternative reagents with OA, which might be more effective at removal of Pu in Tank 18. After the chemical cleaning of Tanks 5 and 6, suggestions were made in SRNL-STI-2009-00492 to explore the use of alternative chemical treatments to selectively remove certain radionuclides, such as nitric acid or sodium carbonate for the removal of plutonium or americium.

DOE did not provide sufficient information regarding the relative costs and benefits associated with completed waste retrieval activities, or potential alternative waste retrieval activities. In terms of benefits, DOE assumed that an upgraded Mantis would remove 50 percent of the residual material, but a basis for this assumption is not fully discussed. DOE discusses the potential reduction to long-term dose under the base case configuration, but an evaluation of the potential benefits for alternative configurations are not presented in the cost-benefit analysis, nor are doses over longer periods of assessment (e.g., Pu-239 peak doses of 3 mSv/yr (300 mrem/yr) that are predicted in DOE's PA to occur beyond the period of performance). Finally, the only technology evaluated in the cost-benefit analysis is an upgraded Mantis. If technologies were explored that targeted key radionuclides that drive the uncertainty in dose contribution from these tanks (e.g., Pu-239 and Np-237), the potential benefit from additional cleaning could be greater. In terms of costs, the cost of the new upgraded Mantis is assumed to be $8.0 M. A prior cost estimate of repairing or replacing the Mantis cited in SRR-CWDA-2011-00091 was $2.8 M. The $2.8 M assumed that the existing above-grade Hose in Hose Transfer Lines (HIHTL) could still be used. Yet another presentation predicts the cost to be greater than $4.0 M (SRR-CWDA-2009-00030). The reasons for the cost differences could have been discussed.

It is not clear how the costs of alternative equipment configurations, deployment of a similar Mantis, upgraded Mantis, or alternative cleaning technology compare to costs of other similar DOE activities as recommended in NRC guidance (NRC, 2007). DOE stated that the Mantis had become ineffective, primarily because the blades had worn out, but that replacing components like the blades or otherwise repairing it was too risky in terms of worker dose. DOE also states that lessons learned from the Mantis operations identified potential design improvements and upgrades (e.g., redesigned rubber squeegee device, suction head, drive mechanism, strainer, etc.), which could potentially result in removing some additional fraction of waste inventories from the tanks. However, DOE does not discuss the option of building a new Mantis of the same design. A similar Mantis would seem to be less costly, more predictable and also effective, given that 2.5 inches remain in parts of Tank 18. Also, DOE plans to use a new Mantis for the HTF, but does not present the cost implications of sharing the Mantis that will be built for HTF.

3.8.2 NRC Evaluation of Approach for Demonstrating MEP for Tanks That Have Not Undergone Removal of Heel and Ancillary Structures

In response to NRC RAI (DOE, 2011), DOE outlines a general approach to demonstrate that the HRRs will be removed to the MEP. Although NRC staff thinks that DOE has the program in place to successfully proceed with cleaning of the tanks to the MEP, the overall description of the approach lacks details in the specific implementation of the process. For example, in Step 2, DOE provides examples of specific metrics for evaluating technology implementation, but does not lay out how these metrics should be interpreted to come to a determination that the anticipated end states have been reached. DOE states that it is not establishing specific removal goals, but that the distinction between anticipated end states and removal goals is not clear in this stage of the process. In Step 4 of DOE's approach (Technology Evaluation), DOE will either rely on a previously performed evaluation where conditions are similar or initiate a new technology evaluation if conditions are different. DOE does not provide detail on how it will define similarities or what differences in conditions will trigger a new evaluation. DOE's description implies that this step may be optional or that its execution is arbitrary in nature. It is the view of NRC staff that the practicality of additional removal would depend on a continued evaluation of the availability and maturity of alternative cleaning technologies, as opposed to a previously performed evaluation.

In reviewing DOE's approach to demonstrate compliance with Criterion 2 for future cleaned tanks, NRC staff had discussions with DOE to clarify the use of qualitative vs. quantitative evaluations. DOE clarified that it plans to evaluate the final quantitative inventories in an SA, similar to the one that was done for Tanks 18 and 19. DOE thinks that it may be appropriate to reduce the number of analytes in the final characterization of future tanks (isotopes and metals) if the prior analyses show that the hard-to-detect radionuclides that are not risk-significant show that they are, in fact, not present in the waste stream. DOE would need to provide evidence for why the previous waste streams represent the waste for the tanks under evaluation. The NRC staff finds this process acceptable as long as DOE follows its commitment to analyze all HRRs in the future, and provides justification if the number of analytes is reduced. Also, each final characterization should be accompanied by a Technical Task Request and Quality Assurance and Quality Control Plan that follows the same general guidelines as those for Tanks 18 and 19. During the staff's review, DOE noted that when conducting the SA, where the final inventory as opposed to the projected inventory is run in the PA, there may be no need to rerun models for every case if the final inventories are not significantly greater than the predicted inventories applied in the PA. If the final inventory is less than that which was projected, DOE will rely on the PA results to determine the risk. Staff finds this approach reasonable as long as DOE follows its commitment to complete a quantitative SA for all inventories that exceed the projected inventories.

While DOE provided some information on the effectiveness of chemical sludge removal (with oxalic acid) for Tanks 5 and 6, the technology's effectiveness could have been better assessed with better characterization of waste residuals. To this end, NRC staff recommends that, to the extent practical, DOE consider obtaining data on HRR inventories prior to and following major cleaning campaigns (e.g., before and after treatment of Type I tanks with OA to provide effectiveness measurements for chemical cleaning and MFB). It is important to note that volume estimates alone are not always the best measure of effectiveness and lack of consideration of uncertainty can lead to erroneous conclusions regarding a technology's

effectiveness. The benefits of waste characterization and technology-effectiveness evaluations are greater if they occur earlier in the tank closure process.

The description in DOE/SRS-WD-2011-001 of how the cost-benefit analysis is to be carried out does not provide sufficient detail for these analyses to be carried out consistently for each tank or group of tanks. Examples of the types of costs to be included are listed, but relatively little guidance is given on the assumptions regarding which technologies merit quantitative evaluation and how to quantify the decrease in potential future doses. It is not clear if a consistent format with the appropriate content will be followed for the future cost-benefit analyses. For example, a basis was not provided for the 50 percent additional bulk volume removal assumption in the cost-benefit analysis for Tanks 18 and 19.

In many cases, the cleaning strategies were ceased, disrupted, or negatively impacted by the unavailability of waste receipt tanks and avoidance of disruption of other ongoing demands for the liquid waste treatment system. DOE noted that a systems engineering analysis precedes all tank cleaning to try to avoid over demand on the liquid waste processing system, but unanticipated circumstances sometimes affect results. NRC encourages DOE to continue to analyze the lessons learned from these prior cleaning campaigns to prevent limitations of the liquid waste system from unexpectedly influencing the effectiveness of future cleaning campaigns. (e.g., future systems engineering plans should account for reasonable contingencies should an evaporator fail.)

NRC staff recommends that DOE continue its efforts to participate in technology exchanges so that it can stay informed of potential new cleaning technologies. New technologies or improvements to current technologies should be fully considered in the selection process for future tank cleaning. For example, a number of presentations made at the 2010 Waste Processing Technology Exchange in Atlanta, GA, indicate that research activities have recently been conducted at the SRS and the larger DOE complex to study the use of other mineral or organic acids for chemical cleaning and the effectiveness of these acids to remove select mineral phases and HRRs from tank waste streams. DOE should try to optimize operational parameters for existing technologies and technologies to be developed in the future to ensure that removal of HRRs is not hampered or made more difficult because of poor planning or lack of investment in waste characterization. DOE should be disciplined and methodical in its approach to the identification of HRRs, selection of cleaning technologies with emphasis on HRR removal, and execution and evaluation of waste retrieval technologies to ensure that it proceeds with removal of HRRs to the MEP. NRC notes that DOE has a process in place to meet Criterion 2 of the NDAA as it proceeds with closure of tanks in the FTF. NRC recommends that DOE continue to consult with NRC as tank farm closure progresses. NRC plans to monitor DOE's efforts in this area under the ALARA provisions of 10 CFR Part 61, Subpart C.

3.9 NRC Evaluation (Criterion 2)

NRC staff evaluated DOE's demonstration of compliance with NDAA Criterion 2 including DOE's approach to: (i) developing inventories for FTF tanks and auxiliary components; (ii) identification of HRRs; (iii) selection of treatment technologies; (iv) demonstration of removal to the MEP, including consideration of the costs and benefits of additional radionuclide removal. NRC staff had a number of comments and RAIs with respect to the projected and final tank

inventories, consideration of inventory uncertainty, selection of removal technologies, and criteria to demonstrate removal to the MEP (Camper, 2010) most of which were addressed in DOE's RAI responses (DOE, 2011).

NRC staff's primary review results related to Criterion 2 are as follows:

- DOE's approach to developing inventories for tanks that have not been cleaned is reasonable and appears to be generally conservative (tends to over- rather than under-predict inventory). NRC staff will continue to monitor DOE's efforts towards reducing the FTF source term as it pertains to the Criterion 3, including ALARA optimization.

- In general, DOE's approach to developing inventories for Tanks 18 and 19 is reasonable, although the approach to quantifying uncertainty could be improved.

- DOE's process for identification of HRRs is reasonable.

- DOE has a program in place to identify, evaluate, and implement cleaning technologies to remove HRRs to the MEP.

- With regard to Tank 18, insufficient information is provided regarding the relative costs and benefits of additional radionuclide removal.

- With regard to tanks that have not yet been cleaned, DOE has a general process in place to demonstrate removal of HRRs to the MEP, but this process could benefit from additional detail.

NRC staff's primary recommendations related to Criterion 2 are as follows:

- NRC staff recommends that DOE explore methods to improve the process by which residual waste volumes and associated uncertainty are estimated.

- NRC staff recommends that DOE continue to evaluate its HRR list as additional information becomes available, to the extent that the list of HRRs is used to inform decisions, such as the selection of radionuclides characterized in residual waste, selection of treatment technologies, and screening of radionuclides for the purpose of detailed PA calculations.

- As practical, NRC staff recommends that DOE continue to participate in technology exchanges and evaluate new cleaning technologies as they become available, rather than defaulting to previously selected technologies or relying on previous evaluations for technology selection.

- NRC staff recommends that DOE consider how it might better assess and optimize the effectiveness of selected technologies (e.g., obtain better baseline information).

- NRC staff recommends that DOE provide more emphasis on removal of HRRs in its technology selection process and provide a clear linkage between the Criterion 2

evaluation and the PA results, including consideration of long-term risks associated with the FTF facility.

- NRC staff recommends that DOE perform a more rigorous cost-benefit analysis related to additional HRR removal for Tank 18.

- NRC staff recommends that DOE provide additional detail on the methods to be used to demonstrate removal to the MEP to ensure consistent (non-arbitrary) application of the criterion.

A 24-month closure process is planned for Tanks 18 and 19 to account for sampling, characterization, initial drafting of closure documents, first-time review process, annulus and cooling coil closure, and a 4-month grout period (SRR-LWP-2009-00001). The next stage of the process for Tanks 18 and 19 is to fill the tanks with a cement-based grout. DOE states that this will have several advantages: (1) filling the void spaces; (2) reducing risk of subsidence; (3) providing a physical barrier from weather exposure; (4) providing a physical deterrent to intruders; and (5) creating a condition that discourages transport of remaining residue (V-ESR-G-00003). The staff thinks these advantages are realized in the longer time frame. For example, the tanks are not at significant risk of intrusion while the FTF is still operational and with institutional controls in place, or of subsidence in the near-term. Furthermore, for Type IV tanks, which have roof structures, the grout is not expected to add much of an additional barrier to weather exposure. A 2005 National Academy of Sciences (NAS) Report recommended that DOE postpone the grouting of tanks by several years, stating that this delay seemed to have essentially no effect on near or long-term risk but that focused research to enhance retrieval technologies such as robotic devices and chemical cleaning could increase DOE's confidence and reduce long-term risk (NAS, 2005).

One of the primary reasons cited by DOE for why additional cleaning of Tanks 18 and 19 is not practical is because it would result in demands on the liquid waste water system and would thereby delay the bulk waste removal of the remaining Type I tanks. However, it is not clear why DOE could not go forward with bulk waste removal and cleaning of the Type I tanks and allow additional time for upgrades to the Mantis, which are already planned for other Type IV tanks in HTF, to occur prior to making a decision on grouting Tank 18. If adhering to the schedule in the FFA is the primary concern, NRC staff has not been provided sufficient information to conclude that approval from SCDHEC and EPA for a delay in the FFA schedule for Tank 18 could not be obtained. The NAS came to a similar recommendation in their 2005 report which stated that DOE should decouple the tank cleaning from tank closing "…on a case-by-case basis where there are indications that near-term (5-10 year) techniques could become available to remove tank heels more effectively, safely, or at a lower cost," (NAS, 2005).

Furthermore, considering additional information provided by DOE in its RAI responses, Tank 18 now appears to be the single largest risk driver for the FTF facility with Tank 18 scheduled for closure per the FFA by December 2012. Given its risk-significance, the NRC staff thinks that DOE should more fully evaluate the practicality of additional radionuclide removal from Tank 18

and explore options for delaying final closure (i.e., grouting) of Tank 18[3] for the reasons listed below. It is important to note that the risk associated with a short delay in the grouting of Tank 18 on the order of a few years is not expected to be significant given ongoing operation and maintenance of the FTF and the fact that a large portion of the residual tank waste has been removed; however, a decision to delay of the grouting of Tank 18 should consider any associated short term risks.

- Insufficient information was provided to the NRC staff related to the costs and benefits of additional radionuclide removal and other factors influencing the decision regarding practicality of additional HRR removal from Tank 18. The NRC staff recommends DOE provide additional information or perform additional analysis to support the Criterion 2 demonstration for Tank 18.

- Significant technical uncertainties exist with respect to DOE's ability to meet the performance objectives in 10 CFR Part 61, Subpart C, that the NRC staff thinks can be addressed in the near-term (e.g., solubility studies--discussed further in section 4.2.9.5). Permanent closure activities such as grouting of the waste tank may make it more difficult for DOE to evaluate or reduce the risks associated with this waste tank in the future, if risk reduction is deemed necessary pending results of future research. Additionally, the results of the near term studies could reduce the extent to which other uncertainties will need to be addressed to support Criterion 3 of the NDAA for tank farm closure.

- A delay in Tank 18 grouting could provide additional time for alternative technologies to be developed (e.g., the improved Mantis design that is anticipated to be used on the H-Tank Farm, Type IV tanks), that could result in greater removal of HRRs from Tank 18, if additional HRR removal is deemed practical.

With respect to Tanks 5 and 6, absent a final inventory, NRC staff does not have sufficient information on the effectiveness of OA or MFB cleaning technologies to make a determination on Criterion 2. NRC staff recommends, to the extent practical, that DOE develop documentation of the following for Type I tanks: HRR inventory before OA, HRR inventory after OA, HRR after MFB. DOE could improve waste characterization, chemical cleaning technology development, and waste retrieval optimization prior to execution of cleaning. Finally, incorporation of lessons learned from those tanks that have undergone cleaning will be important. Considering uncertainty in the timing and magnitude of the peak dose, some radionuclides, tanks, components, and waste forms appear to pose a much greater risk than other radionuclides, tanks, components, and waste forms. Yet, it is not clear how the risk-significance of radionuclides was considered in developing a clean-up strategy to remove HRRs to the MEP. The Criterion 2 demonstration should have a clear linkage to the PA in that DOE should also indicate how it optimized risk reduction for the overall tank system. DOE could have more thoroughly considered the potential benefits associated with removal of additional waste

[3] Although the quality of information provided for Tank 19 is similar to that provided for Tank 18, given the lower risk-significance of Tank 19, NRC staff thinks that final closure of Tank 19 can proceed as planned.

from certain tanks and components or waste types that are risk-significant considering the most up-to-date PA results available.

4. CRITERION 3 (A) AND 3 (B)

Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA) Section 3116(a) states:

> 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 consultation, with the U.S. Nuclear Regulatory Commission (NRC), determines:
>
> (3) (A) does not exceed concentration limits for Class C low-level waste (LLW) 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 LLW 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.

Before the DOE can determine whether Section 3116 (a)(3)(A) or (a)(3)(B) above applies, they must first determine whether the waste exceeds concentration limits for Class C LLW provided in 10 CFR 61.55. After applying NRC's guidance on classification of waste incidental to reprocessing found in Section 3.5 of NUREG-1854 (NRC, 2007), DOE concluded that the stabilized FTF waste at closure would contain concentrations less than the limits for Class C waste. Therefore, NDAA Section 3116(a)(3)(A) applies (see Section 4.1 of this Technical Evaluation Report (TER) for details). However, if DOE or NRC concluded that the waste is Greater-Than-Class C, the additional requirement in NDAA Section 3116(3)(B)(iii) would apply. Namely, DOE has to consult with the NRC in developing its disposal plans. Although DOE indicated in its basis document that 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 elected to take full advantage of the consultation process established under NDAA Section 3116 (3)(B)(iii).

Whether the waste is greater than or less than Class C, DOE must also demonstrate that the waste will be disposed of in compliance with the performance objectives set out in Subpart C of 10 CFR Part 61 and pursuant to a state-approved closure plan or state-issued permit (see NDAA Section 3116 (3)(A)(i) and (ii) and (3)(B)(i) and (ii)). A South Carolina Department of Health and Environmental Control (SCDHEC) industrial waste water construction permit governs tank waste storage and removal operations for FTF closure. Stabilization of the FTF waste tanks and ancillary structures will be carried out pursuant to the FTF General Closure

Plan approved by the State of South Carolina for all FTF tanks and ancillary structures. DOE will also develop a specific Closure Module for each tank or ancillary structure or groupings of tanks and ancillary structures and submit it to the State of South Carolina for approval. Final tank stabilization activities will not proceed until the State approves the closure documentation.

The performance objectives of 10 CFR Part 61, Subpart C, require the assessment of protection of the general population from releases of radioactivity, protection of individuals from inadvertent intrusion into the waste, protection of individuals during operations, and evaluation of the stability of the disposal site after closure. Protection of the general population (including inadvertent intruders) is typically evaluated through a PA calculation that takes into account the relevant physical processes and the temporal evolution of the system. Section 4.2 of the TER presents NRC staff's assessment of DOE's PA (SRR-REG-2007-00002, Rev. 2) and compliance demonstration for all of the Subpart C performance objectives. Section 4.3 presents NRC staff's overall conclusion with respect to Criterion 3. The NDAA also requires NRC to assess compliance with the performance objectives in 10 CFR 61, Subpart C. Section 4.4 lists key monitoring areas that are important to the compliance demonstration (and NRC's ability to assess compliance).

4.1 Assessment of Waste Classification

An assessment of the classification of waste in accordance with 10 CFR 61.55 is required to determine whether the criterion in NDAA Section 3116 (a)(3)(A) or (a)(3)(B) applies.

4.1.1 Waste Classification

LLW intended for near surface disposal is normally classified as Class A, B, or C based on concentration limits for radionuclides listed in 10 CFR 61.55. Table 4-1 and Table 4-2 present DOE waste classification results for an example tank (Tank 18) and for FTF transfer lines, respectively DOE/SRS-WD-2010-001. The concentration limits in 10 CFR 61.55, Table 1 include a general class of radionuclides (alpha-emitting transuranic radionuclides with half-lives greater than 5 years) that DOE must identify. DOE listed several radionuclides that were members of this class (e.g., Np-237, Pu-239, Pu-240, Am-241, Am-243). Table 2 of 10 CFR 61.55 also lists another class of radionuclides, namely, short-lived radionuclides that have half-lives less than 5 years. This class of radionuclides from the HRR list has no material impact on waste classification, because the waste could not be classified as greater than Class C based on the concentrations of these radionuclides alone. A footnote to 10 CFR 61.55, Table 2, indicates that no limits have been established for nuclides with half-lives less than 5 years in Class B or C wastes and that the waste shall be Class B (or less) unless the concentrations of other nuclides dictate that the waste is Class C or greater, independent of these nuclides. The FTF waste addressed in this TER contains a mixture of long- and short-lived radionuclides; therefore, 10 CFR 61.55 (a)(5) was applied to determine waste classification. Because a mixture of radionuclides comprises the residual waste at the FTF, a sum of fractions approach is used to determine the waste classification as provided in 61.55(a)(7). It is significant to note that longer-lived transuranics (10 CFR 61.55, Table 1 radionuclides) dominate the intruder risk and waste classification calculations compared to shorter-lived radionuclides found in Table 2 of 10 CFR 61.55 for Tank 18, while short-lived radionuclides drive the risk for the transfer lines (see Tables 4-1 and 4-2).

NRC's guidance on waste classification found in Section 3.5 of NUREG-1854 (NRC, 2007) discusses three categories of averaging based on: (1) physical homogeneity; (2) stabilization; and (3) site-specific (intruder) analysis considerations. DOE elected to use the site-specific analysis category (Category 3) to determine waste classification. NRC developed waste class limits as part of the original 10 CFR Part 61 rulemaking (see Tables 1 and 2 in 10 CFR 61.55) using a generic analysis to establish concentration limits that would be protective of the inadvertent intruder at a variety of LLW disposal facilities that might be sited in the future. Because the waste classification system codified in 10 CFR 61.55 is based on a generic analyses for potential LLW disposal facilities, NRC developed Category 3 to provide some flexibility to applications for Waste Incidental to Reprocessing (WIR) facilities as staff recognized site- and problem-specific characteristics for tanks and auxiliary components at WIR facilities that may be significantly different than those assumed for LLW disposal facilities. These characteristics include the presence of robust engineered barriers (e.g., intruder barriers such as large rocks, steel tanks, and grouted waste forms), deeper depths of disposal for tank waste residuals, and potential differences in residual waste geometries and volumes.

NRC's guidance in NUREG-1854 recognizes that the risk from the near-surface disposal of radioactive waste is not just a function of concentration, but also of volume and accessibility. Consistent with this observation, the site-specific concentration averaging approach (Category 3) attempts to provide flexibility to allow consideration of volume and accessibility of the waste, as well as to consider changes in internal dose methods and uncertainty in the calculations. However, while risk information such as actual depth of disposal and volume of waste can be incorporated in the waste classification calculations under Category 3, the guidance also indicates that approaches used should err on the side of conservatism (i.e., calculations for waste classification should be at least as restrictive as the PA calculations). NRC staff guidance also indicates that application of a factor increase to the Class C limits based on considerations such as disposal depth, for example, might not be appropriate. Assumptions regarding mixing of higher-concentration waste with lower concentration waste may have already been made when developing the waste classification tables leading to a potential for "double counting" mixing of waste in cleaner, relatively uncontaminated materials. Although accessibility of the waste (e.g., depth and intruder barriers) is considered for concentration averaging applied to incidental waste, NRC staff guidance also indicates that managing of intruder risk with complex, engineered facilities to prohibit access to the waste for very long periods of time (more than 500 years) is not likely to be practical. Thus, there are practical limits as to the types of waste that might be deemed suitable for near-surface disposal. Waste classification is a process used to ensure that waste is properly managed (e.g., LLW disposal in the near surface, HLW disposal in a geologic repository) to protect public health and safety. NRC staff communicated many of these points to DOE in a scoping meeting held on July 13-14, 2010 (ADAMS Accession No. ML102000163) in advance of DOE's submittal of the draft Basis for Section 3116 Determination (DOE/SRS-WD-2010-001).

As stated above, DOE used an approach consistent with Category 3 of NRC's concentration averaging guidance described in Section 3.5 and Appendix B of NUREG-1854 (NRC, 2007). Due to the depth of residual waste assumed to be present in HLW tanks and auxiliary components (greater than 3 m or 10 ft below the closure cap), DOE determined that an excavation scenario was not plausible for the FTF. Typically, when waste is located at distances greater than 3 m or 10 ft, a well drilling scenario is evaluated as a more plausible intrusion event. In evaluating a well drilling scenario, DOE assumed that the robust intruder

barrier incorporated into the design of the closure cap and the tank system itself could be relied on to prevent intrusion into the tanks until beyond 500 years. Due to the greater vulnerability of the transfer lines compared to the tanks, DOE assumed that intrusion into the transfer lines could begin earlier, 100 years post closure, although DOE considered this a conservative assumption given the presence of the robust intruder barrier.

The inventory used for the purpose of waste classification is decayed to the year 2020, the date of anticipated closure of the FTF. This adjustment is important for short-lived radionuclides such as Cs-137 and Sr-90. The total inventory on all tank components is assumed to be present on the floor of the tanks or on auxiliary components with the exception of the transfer lines. The inventory is assumed to be homogeneously distributed on the inside surface of the transfer lines. The density of the waste is assumed to be similar to the density of the stabilizing (grout) materials in the concentration calculations.

NRC staff has evaluated DOE's methodology for classifying waste and finds the approach an acceptable application of Category 3 in NUREG-1854. Information provided for Tank 18 and transfer lines in the draft waste determination or basis for Section 3116 determination (DOE/SRS-WD-2010-001) indicates that the residual waste for these components does not exceed Class C concentration limits. NRC concurs with this assessment. Although DOE indicates its expectation that all tanks and auxiliary equipment will meet Class C limits, it did not evaluate all FTF components. During an RAI resolution meeting on January 20, 2011 (ADAMS Accession No. ML110250118), DOE indicated its intent to confirm waste classification at the time of individual component closure at which time the final inventory of the components will be confirmed. Section 3116(a)(3)(B)(iii) requires DOE to consult with NRC in its development of disposal plans if the waste exceeds Class C concentration limits. In the draft Basis for Section 3116 Determination (DOE/SRS-WD-2010-001), DOE stated its intent to "…take full advantage of the consultation process established by Section 3116," and asked NRC to "…identify what changes, if any, it would recommend to DOE's disposal plans," as described in the draft waste determination. NRC has reviewed DOE's disposal plans for the FTF waste as part of the extensive consultation process that is documented in this TER, thereby satisfying the requirements of Section 3116(a)(3)(B)(iii). Consequently, no additional DOE consultation with NRC is required for tanks containing residual waste that might exceed Class C concentrations following final sampling and inventory development.

Table 4-1 Sum of Fractions (SOF) for Tank 18

Radionuclide	Inventory (Ci)	Concentration Ci/m^3 or nCi/g^2	Site-Specific Factor	Adjusted Concentration Ci/m^3 or nCi/g	Class C Limit Ci/m^3 or nCi/g	Sum of Fractions (SOF)
C-14	$9.0x10^{-1}$	$6.0x10^{-2}$	$4.4x10^{-1}$	$2.6x10^{-2}$	$8.0x10^{0}$	$3.3x10^{-3}$
Ni-59	$3.3x10^{-1}$	$2.2x10^{-2}$	$3.0x10^{-1}$	$6.6x10^{-3}$	$2.2x10^{2}$	$3.0x10^{-5}$
Nb-94	$5.5x10^{-4}$	$3.6x10^{-5}$	$8.8x10^{-1}$	$3.2x10^{-5}$	$2.0x10^{-1}$	$1.6x10^{-4}$
Tc-99	$9.0x10^{-1}$	$6.0x10^{-2}$	$1.8x10^{-1}$	$1.1x10^{-2}$	$3.0x10^{0}$	$3.6x10^{-3}$
I-129	$2.7x10^{-4}$	$1.8x10^{-5}$	$1.9x10^{0}$	$3.4x10^{-5}$	$8.0x10^{-2}$	$4.2x10^{-4}$
Np-237	$1.9x10^{-1}$	$6.1x10^{0}$	$1.1x10^{0}$	$6.7x10^{0}$	$1.0x10^{2}$	$6.7x10^{-2}$

Pu-238	$1.3x10^{3}$	$4.2x10^{4}$	$1.8x10^{-5}$	$7.5x10^{-1}$	$1.0x10^{2}$	$7.5x10^{-3}$
Pu-239	$2.8x10^{2}$	$9.0x10^{3}$	$9.0x10^{-4}$	$8.1x10^{0}$	$1.0x10^{2}$	$8.1x10^{-2}$
Pu-240	$6.5x10^{1}$	$2.1x10^{3}$	$1.5x10^{-3}$	$3.1x10^{0}$	$1.0x10^{2}$	$3.1x10^{-2}$
Pu-241	$2.7x10^{2}$	$8.9x10^{3}$	$5.9x10^{-4}$	$5.3x10^{0}$	$3.5x10^{3}$	$1.5x10^{-3}$
Pu-242	$2.7x10^{-2}$	$8.6x10^{-1}$	$2.1x10^{-3}$	$1.8x10^{-3}$	$1.0x10^{2}$	$1.8x10^{-5}$
Pu-244	$6.2x10^{-6}$	$2.0x10^{-4}$	$4.5x10^{-3}$	$8.9x10^{-7}$	$1.0x10^{2}$	$8.9x10^{-9}$
Am-241	$1.6x10^{2}$	$5.2x10^{3}$	$8.9x10^{-4}$	$4.6x10^{0}$	$1.0x10^{2}$	$4.6x10^{-2}$
Am-242m	$3.8x10^{-2}$	$1.2x10^{0}$	$1.9x10^{-3}$	$2.3x10^{-3}$	$1.0x10^{2}$	$2.3x10^{-5}$
Am-243	$2.3x10^{0}$	$7.7x10^{1}$	$1.7x10^{-2}$	$1.3x10^{0}$	$1.0x10^{2}$	$1.3x10^{-2}$
Cm-243	$1.8x10^{-2}$	$5.7x10^{-1}$	$1.5x10^{-4}$	$8.6x10^{-5}$	$1.0x10^{2}$	$8.6x10^{-7}$
Cm-244	$9.8x10^{1}$	$3.2x10^{3}$	$2.7x10^{-5}$	$8.5x10^{-2}$	$1.0x10^{2}$	$8.5x10^{-4}$
Cm-245	$1.2x10^{-2}$	$3.8x10^{-1}$	$2.3x10^{-2}$	$8.8x10^{-3}$	$1.0x10^{2}$	$8.8x10^{-5}$
Cm-247	$2.1x10^{-6}$	$6.7x10^{-5}$	$1.0x10^{-1}$	$6.7x10^{-6}$	$1.0x10^{2}$	$6.7x10^{-8}$
Cm-248	$9.5x10^{-5}$	$3.1x10^{-3}$	$3.9x10^{-1}$	$1.2x10^{-3}$	$1.0x10^{2}$	$1.2x10^{-5}$
Cf-249	$2.3x10^{-3}$	$7.3x10^{-2}$	$5.2x10^{-4}$	$3.8x10^{-5}$	$1.0x10^{2}$	$3.8x10^{-7}$
					SOF	$2.5x10^{-1}$
H-3	$1.4x10^{-2}$	NA	NA	NA	NA	NA
Co-60	$1.2x10^{0}$	NA	NA	NA	NA	NA
Ni-63	$1.7x10^{1}$	$1.1x10^{0}$	$7.1x10^{-4}$	$7.7x10^{-4}$	$7.0x10^{2}$	$1.1x10^{-6}$
Sr-90[1]	$3.2x10^{3}$	$1.7x10^{2}$	$3.3x10^{-3}$	$5.5x10^{-1}$	$7.0x10^{3}$	$7.8x10^{-5}$
Cs-137[1]	$1.2x10^{4}$	$6.0x10^{2}$	$1.3x10^{-2}$	$7.8x10^{0}$	$4.6x10^{3}$	$1.7x10^{-3}$
					SOF	$1.8x10^{-3}$

[1] Sr and Cs concentrations decay corrected to 2020.
[2] The volume of waste assumed is assumed to be 15 m^3 and the weight of the waste is assumed to be $3.1x10^{+04}$ kg.

Table 4-2 Sum of Fractions (SOF) Calculation for 3-Inch Transfer Line

Radionuclide	Inventory (Ci/ft^2)	Concentration Ci/m^3 or nCi/g^2	Site-Specific Factor	Adjusted Concentration Ci/m^3 or nCi/g	Class C Limit Ci/m^3 or nCi/g	Sum of Fractions (SOF)
C-14	$4.2x10^{-8}$	$7.7x10^{-5}$	$3.0x10^{2}$	$2.3x10^{-2}$	$8.0x10^{0}$	$2.9x10^{-3}$
Ni-59	$1.7x10^{-6}$	$3.1x10^{-3}$	$5.2x10^{0}$	$1.6x10^{-2}$	$2.2x10^{2}$	$7.2x10^{-5}$
Nb-94	$3.7x10^{-8}$	$7.0x10^{-5}$	$6.9x10^{-1}$	$4.8x10^{-5}$	$2.0x10^{-1}$	$2.4x10^{-4}$
Tc-99	$1.5x10^{-5}$	$2.8x10^{-2}$	$3.6x10^{-1}$	$9.9x10^{-3}$	$3.0x10^{0}$	$3.3x10^{-3}$
I-129	$7.0x10^{-11}$	$1.3x10^{-7}$	$7.9x10^{2}$	$1.0x10^{-4}$	$8.0x10^{-2}$	$1.3x10^{-3}$
Np-237	$7.8x10^{-8}$	$1.8x10^{-2}$	$3.6x10^{2}$	$6.6x10^{0}$	$1.0x10^{2}$	$6.6x10^{-2}$
Pu-238	$1.8x10^{-4}$	$4.3x10^{1}$	$5.1x10^{-4}$	$2.2x10^{-2}$	$1.0x10^{2}$	$2.2x10^{-4}$

Pu-239	$7.1x10^{-5}$	$1.6x10^{1}$	$8.6x10^{-2}$	$1.4x10^{0}$	$1.0x10^{2}$	$1.4x10^{-2}$
Pu-240	$2.6x10^{-5}$	$6.2x10^{0}$	$2.6x10^{-1}$	$1.6x10^{0}$	$1.0x10^{2}$	$1.6x10^{-2}$
Pu-241	$1.2x10^{-4}$	$2.9x10^{1}$	$6.7x10^{-2}$	$1.9x10^{0}$	$3.5x10^{3}$	$5.5x10^{-4}$
Pu-242	$2.2x10^{-7}$	$5.1x10^{-2}$	$6.5x10^{-1}$	$3.3x10^{-2}$	$1.0x10^{2}$	$3.3x10^{-4}$
Pu-244	$1.0x10^{-10}$	$2.4x10^{-5}$	$3.5x10^{0}$	$8.3x10^{-5}$	$1.0x10^{2}$	$8.3x10^{-7}$
Am-241	$5.8x10^{-4}$	$1.4x10^{2}$	$4.4x10^{-3}$	$6.0x10^{-1}$	$1.0x10^{2}$	$6.0x10^{-3}$
Am-242m	$8.3x10^{-7}$	$1.9x10^{-1}$	$1.6x10^{-2}$	$3.1x10^{-3}$	$1.0x10^{2}$	$3.1x10^{-5}$
Am-243	$9.5x10^{-8}$	$2.2x10^{-2}$	$2.0x10^{0}$	$4.4x10^{-2}$	$1.0x10^{2}$	$4.4x10^{-4}$
Cm-243	$1.4x10^{-8}$	$3.3x10^{-3}$	$7.0x10^{-5}$	$2.3x10^{-7}$	$1.0x10^{2}$	$2.3x10^{-9}$
Cm-244	$2.7x10^{-5}$	$6.3x10^{0}$	$1.2x10^{-5}$	$7.5x10^{-5}$	$1.0x10^{2}$	$7.5x10^{-7}$
Cm-245	$3.4x10^{-10}$	$8.0x10^{-5}$	$7.4x10^{3}$	$5.9x10^{-1}$	$1.0x10^{2}$	$5.9x10^{-3}$
Cm-247	$1.3x10^{-22}$	$3.0x10^{-17}$	$8.8x10^{13}$	$2.6x10^{-3}$	$1.0x10^{2}$	$2.6x10^{-5}$
Cm-248	$3.0x10^{-23}$	$6.7x10^{-18}$	$1.5x10^{15}$	$1.0x10^{-2}$	$1.0x10^{2}$	$1.0x10^{-4}$
Cf-249	$8.0x10^{-25}$	$1.9x10^{-19}$	$1.7x10^{9}$	$3.2x10^{-10}$	$1.0x10^{2}$	$3.2x10^{-12}$
					SOF	$1.2x10^{-1}$
H-3	$8.1x10^{-7}$	NA	NA	NA	NA	NA
Co-60	$2.3x10^{-5}$	NA	NA	NA	NA	NA
Ni-63	$1.4x10^{-4}$	$2.5x10^{-1}$	$4.5x10^{-3}$	$1.1x10^{-3}$	$7.0x10^{2}$	$1.6x10^{-6}$
Sr-90[1]	$3.1x10^{-2}$	$5.5x10^{1}$	$1.4x10^{1}$	$7.7x10^{2}$	$7.0x10^{3}$	$1.1x10^{-1}$
Cs-137[1]	$5.8x10^{-3}$	$1.1x10^{1}$	$3.9x10^{1}$	$4.1x10^{2}$	$4.6x10^{3}$	$9.0x10^{-2}$
					SOF	$2.0x10^{-1}$

[1] Sr and Cs concentrations decay corrected to 2020.
[2] The volume of waste assumed is assumed to be 5.5E-05 m^3 and the weight of the waste is assumed to be 4.3E+03 g.

4.2 Performance Assessment to Demonstrate Compliance with 10 CFR Part 61, Subpart C Performance Objectives

For non-HLW determinations, DOE normally develops a PA to demonstrate that dose-based regulatory criteria found in 10 CFR Part 61, Subpart C, can be met. PA components include: 1) the evaluation of potential initiating events (both natural and anthropogenic) that can cause releases of radioactive material; 2) estimates of the release rates of radionuclides into the accessible environment; 3) modeling the fate and transport of radionuclides; and 4) evaluation of the potential pathways of exposure and associated risks (doses) to human health. The PAs submitted by DOE to support non-HLW waste determinations have included a collection of loosely integrated process models to demonstrate compliance with performance objectives in 10 CFR Part 61, Subpart C.

Various approaches to PA calculations (e.g., deterministic, probabilistic) have their advantages and disadvantages. A deterministic approach can be very valuable when compliance can be demonstrated easily with parameters and models that clearly tend to over-predict the potential risk posed by the disposal facility. This type of analysis requires little support of models and model parameters and thus can save a lot of time and money. However, compliance

demonstrations with simple deterministic models can be difficult for evaluations that assess compliance over periods that span tens of thousands of years in unique and complex engineered or natural systems that are difficult to represent conceptually and carry little to no model support from analogous systems, as these systems are difficult to identify and evaluate or may not exist. These more complex systems are typically characterized by many inter-related processes with large uncertainty. Models that attempt to simulate these systems typically respond in a nonlinear fashion to model inputs within a reasonable range of parameter space. Thus, system response is difficult to study using deterministic methods and deterministic approaches may not easily be shown to be demonstrably bounding. A probabilistic approach can have distinct advantages when there are a number of uncertainties that may significantly influence the results of a PA.

It is important to note that model support (i.e., data or information that supports the model or parameters used in the model) is necessary to provide confidence in the predictive capability of the PA model being evaluated. Because of the long time periods involved with most PA analyses, PA models cannot be validated in a traditional sense. However, the results of laboratory and field experiments, monitoring data, natural analogs, peer reviews, expert elicitation, and supporting submodels can provide multiple lines of evidence and collectively provide increased confidence in the predicative capability of the PA model. The amount of model support provided should be commensurate with the importance of the barrier or process being simulated to the compliance demonstration. Thus, a combination of approaches is often necessary to ensure the identification of and support for key processes and parameters.

4.2.1 Summary of Performance Objectives

The NDAA establishes the applicable criteria for determining that waste is not high level waste (HLW), and includes a requirement that the waste will be disposed of in compliance with the performance objectives set out in 10 CFR Part 61, Subpart C. The performance objectives provide criteria to ensure that the public, workers, and environment will be protected from releases of radioactivity.

The regulation at 10 CFR 61.41, states the following:

> 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." (NRC, 2001a)

The 0.25 mSv/yr (25 mrem/yr) limit applies for the postclosure period of a disposal facility. The other radiological control limits of 10 CFR Part 20, apply during facility operation (NRC, 2001b).

The regulation at 10 CFR 61.42," states the following:

> 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.

Although this performance objective does not specify a particular dose limit, compliance with the technical requirements of 10 CFR Part 61 and, in particular, with the classification system of 10 CFR 61.55, is considered to provide adequate protection to intruders at a near-surface land disposal facility. In the Draft Environmental Impact Statement (EIS) for Part 61 (NRC, 1981), NRC used a 5-mSv (500-mrem) dose limit to an acute inadvertent intruder to establish the concentration limits and other aspects of the waste classification system. In addition, 10 CFR Part 61 does not specify a time for institutional controls in the performance objectives, but 10 CFR 61.59(b) does require that "...controls may not be relied upon for more than 100 years."

The regulation at 10 CFR 61.43, "Protection of individuals during operations," states the following:

> 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 performance objective applies to both the public and to disposal facility workers.

The regulation at 10 CFR 61.44, states the following:

> 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.

The stability performance objective is consistent with a premise of 10 CFR Part 61 that the facility must be sited, designed, used, operated, and closed with the intention of providing permanent disposal. A disposal facility should not require long-term maintenance and care. Stability is particularly important considering the requirements in 10 CFR 61.59(b) that "...institutional controls must not be relied upon for more than 100 years following transfer of control of the disposal site to the owner."

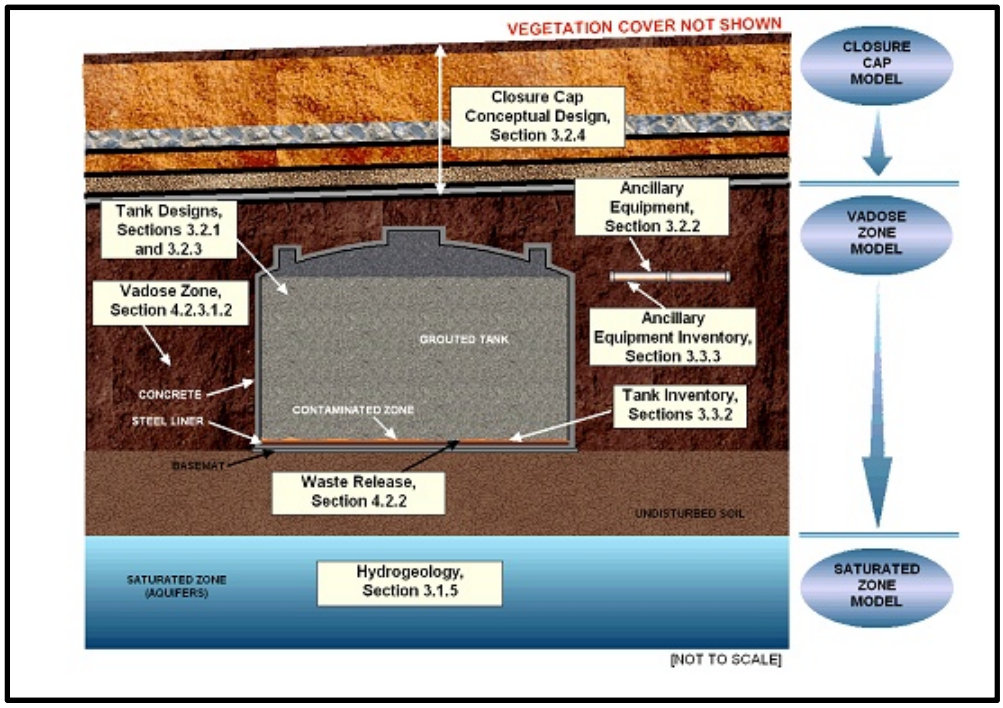
4.2.2 Summary of Performance Assessment Approach and Results

DOE developed a PA for the FTF (SRS-REG-2007-00002) to estimate the release of radiological constituents from 22 waste tanks and associated ancillary equipment. The releases and resulting radiological exposures to members of the public are estimated for comparison to

the Part 61, Subpart C performance objectives. To evaluate the long-term performance of the FTF waste tanks and ancillary equipment to limit radiological exposures to members of the public, DOE constructed an Integrated Site Conceptual Model (ISCM) with three main components: an engineered closure cap conceptual model, a vadose zone conceptual model, and a saturated zone conceptual model (see Figure 4-1). DOE states in response to RAI-PA-2 (SRR-CWDA-2011-00054, Rev. 1) that a comprehensive list of features, events, and processes (FEPs) were considered in developing the ISCM. For conceptual model purposes, the closure cap is considered a surface feature, and the region beneath the base of the closure cap is considered the vadose zone. The waste sources (tanks and ancillary equipment) are thus located within the vadose zone, along with undisturbed soil and engineered backfill. DOE considered six alternative scenarios, termed Configurations A through F, to evaluate potential system behavior for the waste tanks and a single configuration for the ancillary equipment in the PA. The configurations are reported to represent conditions that may be present, without regard to the mechanism that led to those conditions. The configurations abstract the performance of the closure cap, waste tanks or ancillary equipment, and the resulting release of radionuclides to the environment. DOE states that the alternate configurations should not be interpreted as representing a specific mechanism for liner/grout/concrete degradation or taken as belief that a given or set of conditions would actually be present in a closed waste tank system at some point in the future. Section 4.4.2 of (SRS-REG-2007-00002, Rev. 1) describes Configurations A through F. In response to RAI-PA-1 (DOE**,** 2011), DOE also developed Configuration G (DOE, 2011). DOE considers Configuration A to be the best estimate of the behavior of the closed FTF facility.

The primary output of the closure cap model is an estimate of the flow of water that enters the vadose zone. DOE estimates water infiltration through the closure cap using the Hydrologic Evaluation of Landfill Performance (HELP) model (EPA-600-R-94-168a and EPA-600-R-94-168b). The HELP model requires input of weather, soil, and closure cap design data to provide estimates of runoff, evapotranspiration, lateral drainage, infiltration, hydraulic head, and water storage. Infiltrating water is passed to the vadose zone model where DOE evaluated release scenarios of potential importance. Section 4.2.4.4 of this TER summarizes DOE modeling of the closure cap.

Figure 4-1 Integrated Site Conceptual Model (Image Credit: SRR-REG-2007-00002, Rev. 1) (Labels Correspond to Sections of DOE's PA)



In the vadose zone modeling, DOE estimates degradation rates and failure times for cementitious materials and steel liners for each of the four tank types (i.e., Types I, III, IIIA, and IV) that comprise the FTF. The estimated behavior of the cementitious materials and steel liners affects the chemical environment in the tanks and subsequent release of radionuclides from the contaminated zone (CZ). Radionuclides released from the CZ are then available for transport out of the waste tanks and into the vadose zone surrounding the tanks. DOE includes 54 radionuclides that were selected from a screening analysis. Section 3.1 summarizes the screening analysis and the resulting radionuclide inventory evaluated by DOE in the PA. DOE deterministically models the vadose zone domain for each FTF waste source individually within PORFLOW® (ACRi, 2000) to determine the tank-specific, time-dependent radionuclide flux entering the water table. PORFLOW® is also used to estimate average velocity profiles of water infiltrating through the FTF vadose zone for all configurations and tank types simulated in the probabilistic analysis using the GoldSim® modeling platform (see Section 4.2.18.2). The transfer line source term is modeled by distributing the assumed inventory equally over the entire FTF area. Pump tanks, a catch tank, and evaporator pots are modeled as point sources. Section 4.2.8 of this document summarizes DOE's approaches for modeling waste tank and auxiliary equipment degradation, radionuclide releases, and aqueous flow and transport in the vadose zone.

The releases from these scenarios are then input to environmental transport and exposure pathway models to evaluate radiological exposures to members of the public. Groundwater is the primary exposure pathway DOE considers. DOE used the PORFLOW® code (ACRi, 2000) to model flow and transport of contaminants through the saturated zone. The primary focus of

the local PORFLOW® saturated zone flow and transport model is concentrations of radionuclides in wells at 1 m (3 ft) (intruder) and 100 m (330 ft) (member of the public) compliance points for the dose analyses. Section 4.2.10 summarizes DOE's approach for modeling contaminant flow and transport in the saturated zone. DOE also used GoldSim® to evaluate radionuclide transport through the saturated zone in the probabilistic analysis.

Concentrations of radionuclides in groundwater at the compliance points are used to calculate doses to receptors through a number of residential and agricultural pathways. DOE used GoldSim® to conduct all-pathways and intruder analyses by using the contaminant transport results from PORFLOW® and dose conversion factors to calculate groundwater pathways and inadvertent intruder doses. Section 4.2.12 summarizes DOE's all-pathways analysis methodology for members of the public. Section 4.2.16 summarizes DOE's intruder pathways analysis methodology.

Gaseous phase transport to the surface is modeled separately by DOE to provide estimates of potential dose from releases of radioactivity into the environment to demonstrate compliance with 10 CFR 61.41. PORFLOW® is used in the PA to calculate vapor phase radionuclide diffusion to the ground surface for use in air transport calculations. DOE includes 9 radionuclides that were selected from a screening analysis. The Clean Air Act Assessment Package – 1988 (CAP-88) computer model (EPA, 1992) is used to estimate dose from radionuclide emissions to air. CAP-88 was used in the FTF PA to estimate annual dose to Maximally Exposed Individuals (MEI) considering plume and ground shine (gamma radiation), inhalation, and foodstuff ingestion pathways using the results of the vapor phase radionuclide diffusion to the surface results from PORFLOW® discussed above.

DOE uses a hybrid deterministic-probabilistic approach to assess the sensitivity of model results to input parameters and investigate uncertainties in parameters and models. The hybrid approach uses deterministic modeling to evaluate water flow and radionuclide release and transport. Limited sensitivities are assessed in the deterministic modeling: inventory, sorption parameters, and barrier performance. DOE used the PORFLOW® code to perform the deterministic analyses. The hybrid approach also uses probabilistic modeling to evaluate uncertainties in models and parameters. DOE used the GoldSim® code with water flow input from PORFLOW® to perform the probabilistic analyses. Section 4.2.18 summarizes DOE's approach for uncertainty and sensitivity analyses.

4.2.3 NRC Evaluation of PA Approach

DOE appears to have used a reasonable methodology for identifying FEPs that could affect the ability of the disposal system to meet performance objectives. However, DOE reports in response to RAI-PA-2 (DOE, 2011) that the alternative configurations are not explicitly linked to specific FEPs, but that the collective effects of FEPs are assumed to be addressed in the range of the modeling configurations. NRC staff is concerned that the assumption may not be valid. This issue underlies many of the NRC staff's concerns regarding model support and uncertainty expressed in later sections regarding components of the PA. Further, NRC staff is concerned about the lack of traceability of FEPs to alternative configurations. The lack of traceability or direct linkages to the PA has the potential to lead to poor future decisions about whether new information regarding FEPs warrants further consideration or a revision to the PA. NRC recommends that DOE perform a systematic scenario analysis process in which FEPs are

identified, screened, and dispositioned using transparent and traceable documentation of the FEPs considered, the screening arguments, and how FEPs are implemented in the models to support future waste determination efforts.

The PA presents a reasonable development of the types of exposure pathways important for receptors located at SRS; the receptor characteristics and exposure scenarios are reasonable (with exceptions noted in Section 4.2.1); and the performance objectives, dose methodology, point of compliance, exposure period, and institutional control period are all acceptable.

DOE evaluated a large range of alternative conceptual models (see Table 4-11), performed uncertainty and sensitivity analyses (see Section 4.2.18), provided or analyzed additional data from a number of RAIs, or used other means to demonstrate compliance with the performance objectives of 10 CFR Part 61, Subpart C. Table 4-3 provides an overview of DOE's PA results for comparison with the performance objectives of 10 CFR 61, Subpart C. The results are explained in more detail in subsequent sections. Provided all of DOE's key modeling assumptions are met, results of DOE's PA indicate that the performance objectives will most likely be met considering a 10,000 year compliance period.

This conclusion seems to be especially true for the 10 CFR 61.42 performance objective related to protection of the inadvertent intruder evaluated by NRC at a dose limit of 5 mSv/yr (500 mrem/yr). However, DOE's PA results also indicate that the 10 CFR 61.41 performance objective of 0.25 mSv/yr (25 mrem/yr) will be significantly exceeded (order of magnitude or more) at some point in the future. Because DOE's intruder analysis also evaluates potential doses to an inadvertent intruder at a well located 1 m (3 ft) from the FTF, concerns related to the 10 CFR 61.41 compliance demonstration also hold true for the 10 CFR 61.42 analysis. Once issues are resolved related to groundwater dependent pathways under 10 CFR 61.41, NRC staff expects issues related to the inadvertent intruder analysis will also be resolved.

4.2.3.1 Barriers Important to the Compliance Demonstration

NRC staff focused its review on barriers that served to delay the timing and magnitude of peak dose. Considering the magnitude of peak dose over longer periods of performance (3 to 6 mSv (300 to 600 mrem/yr), NRC staff carefully reviewed DOE's analysis and support for performance of two key barriers, in particular, that result in the delay of key radionuclide releases and peak dose for thousands to tens of thousands of years in DOE's PA. These two barriers are the following:

- Carbon Steel Tank Liners
- Chemical Barriers (present in the tank grout or contaminated zone)

As can be seen in Figure 4-2, the steel liner and chemical barriers are responsible for a delay in the peak dose for Tc from Type I tanks of 6 mSv/yr (600 mrem/yr) in the base case or

deterministic model[4] for around 13,000 years each or a total of 26,000 years. The potential peak doses from Tc could be mitigated by additional cleaning of Type I tanks (with preliminary cleaning results from Tanks 5 and 6 providing encouraging results (see discussion in DOE's RAI-UA-3 response (DOE, 2011)), or through additional waste characterization that could indicate more favorable solubility control of Tc than assumed in the PA. For Pu peak doses resulting from already cleaned Tank 18, DOE predicts doses of around 3 mSv/yr (300 mrem/yr) in its base case or deterministic model[5] at around 40,000 years. The primary barriers delaying the timing of the peak doses from Pu are the steel liner (around 3000 years), chemical barrier (approaching 30,000 years), basemat (around 1500 years), and natural system (around 8000 years). The peak dose associated with Pu from Tank 18 could be mitigated through: (i) additional waste characterization that might suggest greater solubility control of Pu than assumed in the PA; (ii) additional experimentation or modeling that would support more favorable assumptions regarding natural attenuation of Pu; or (iii) through additional inventory reduction of Tank 18 (effectiveness dependent on degree of solubility control). However, considering the fact that unacceptably high peak doses could occur within the 10,000 year period of compliance with only a factor of 3 or 4 faster time to collective failure of a combination of barriers for Tc or Pu, respectively, and considering the large uncertainty associated with predictions of long-term performance of engineered barriers, NRC staff are not convinced that the high peak doses currently presented in DOE's PA (or lower peak doses of unknown magnitude that might be associated with a more realistic model[6]) could not be realized within a 10,000 year compliance period.

With respect to magnitude of peak dose, Table 4-4 indicates that several barriers are relied on in DOE's PA to mitigate or reduce potential releases from F-Area tanks, including: (i) chemical barriers in the contaminated zone and tank grout that mitigate the release of Pu-239 and Np-237 based on solubility control (also applies to Tc prior to transition to Oxidized, Region III); (ii) the concrete basemat underneath the tanks that mitigates the release of Tc-99, Pu-239 and Np-237; (iii) natural system dilution and dispersion for Tc-99, Pu-239 and Np-237; and (iv) sorption in the subsurface for Pu-239 each of which reduce the magnitude of releases or concentrations along flow paths away from FTF sources by orders of magnitude. These barriers are the focus of NRC staff's technical review and of the technical discussion presented in the Sections of this TER that follow. Given the limited support for key modeling assumptions, including barrier performance, in DOE's PA and supporting references, NRC staff makes several recommendations related to additional follow-up work DOE could conduct during the monitoring period to strengthen its compliance demonstration.

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[4] Peak doses approach an order of magnitude higher, or 60 mSv/yr (6000 mrem/yr), for realizations in the probabilistic analysis that have similar deterministic analysis assumptions regarding Tc solubility (See realizations symbolized with blue dots in Figure 2a in the July 19 meeting summary attachment found at ADAMS Acc. No. ML112200656).

[5] Peak dose approach a factor of 3 higher, or 10 mSv/yr (1000 mrem/yr), for comparable realizations in DOE's probabilistic analysis (See Figure in July 19 meeting summary at ADAMS Acc. No. ML112200656).

[6] See Section 4.2.19.5, "Evaluation of Model Abstraction Uncertainty and Other DOE PA Conservatisms" for additional discussion on model simplifications that tend to bias the PA results high.

4.2.3.2 PA Overview Review Results and Recommendations

With respect to DOE's PA approach NRC staff notes the following:

- Due to lack of transparency and traceability of DOE's PA documentation, it is difficult to determine whether FEPs are comprehensively evaluated in DOE's PA.

- Support for assumed long-term performance of key barriers at FTF is limited (see additional discussion in Sections of this TER that follow).

With respect to DOE's PA approach NRC staff makes the following recommendations, along with a rating of risk-significance and priority[7]:

- NRC recommends that DOE perform a systematic scenario analysis process in which FEPs are identified, screened, and dispositioned using transparent and traceable documentation of the FEPs considered, the screening arguments, and how FEPs are implemented in the models to support future waste determination efforts. (Medium Risk-Significance, Intermediate-Term)

- NRC staff recommends DOE initiate additional follow-up work during the monitoring period to provide support for key modeling assumptions and barriers relied on for long-term performance in DOE's PA as discussed in more detail in the Sections that follow (see individual recommendations for risk-significance and timing information).

[7] Items of low risk-significance may reduce safety margin but are not expected to be able to alter compliance conclusions alone, while items of high risk-significance are expected to impact the compliance demonstration. Short term recommendations are expected to occur in the next couple of years, intermediate recommendations are expected to occur prior to tank farm closure, and long-term/maintenance recommendations are expected to be either (i) optional or (ii) contingent on results of other analyses.

Table 4-3 Summary of DOE Performance Assessment Results (values in red text are over the compliance limit)*

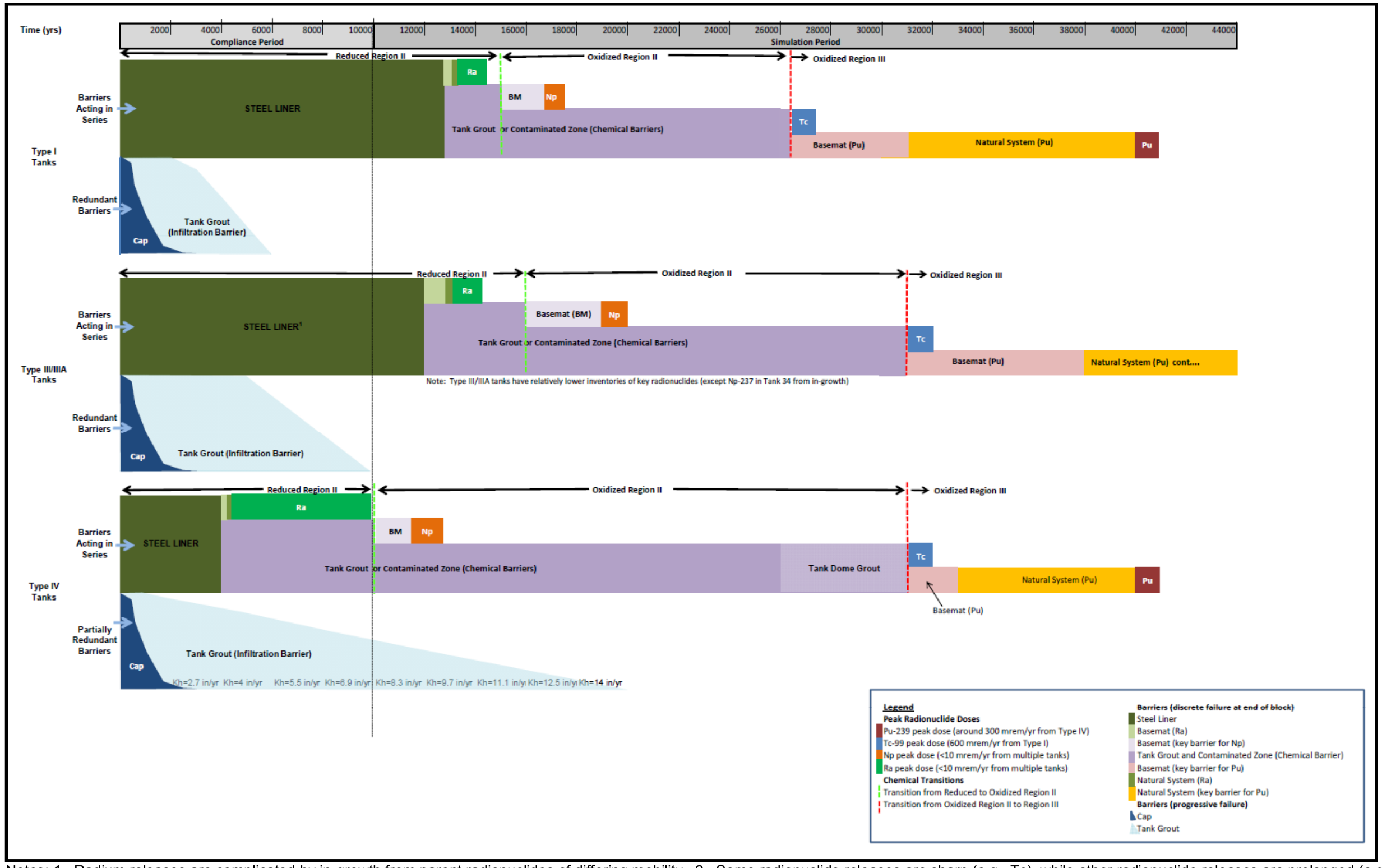
	Protection of the General Population 10 CFR 61.41	Notes	Protection of Individuals from Inadvertent Intrusion 10 CFR 61.42	Notes
Compliance Conclusion	Performance objective (PO) of 25 mrem/yr can be met within 10,000 year compliance period	PA results show doses are expected to be significantly above PO at some uncertain point in the future.	PO (evaluated at 500 mrem/yr) can be met	Robust compliance conclusion for drill cuttings. Uncertainty with dose from intruder well.
Deterministic (Case A)	2.5 mrem/yr	Within 10,000 years	73 mrem/yr (drill cuttings)	Within 10,000 years Chronic exposure
	18 mrem/yr	Within 20,000 years	1.6 mrem/yr (drill cuttings)	Within 10,000 years Acute exposure
	600 mrem/yr	Peak dose around 26,000 years	75 mrem/yr (groundwater)	Within 20,000 years Chronic exposure
Deterministic (Alternative or Sensitivity Cases)	15 mrem/yr	Case D (Within 10,000 years)	125 mrem/yr (drill cuttings)	4 inch transfer line Chronic exposure Sensitivity case
	10 mrem/yr	Case E (Within 10,000 years)	7 mrem/yr (drill cuttings)	Tank 18 Acute exposure Sensitivity case
	5 mrem/yr	Case F (Within 10,000 years)		
	125 mrem/yr	Case G (Within 10,000 years)		
	~550 mrem/yr	Case G (Within 20,000 years)		

*Multiply by 0.01 to change mrem to mSv.

Table 4-3 continued	**Protection of the General Population 10 CFR 61.41**	**Notes**	**Protection of Individuals from Inadvertent Intrusion 10 CFR 61.42**	**Notes**
Probabilistic (mean of peak) or Sensitivity*	12 or 21 mrem/yr	Within 10,000 years Case A		
	245 or 452 mrem/yr	Within 10,000 years Case D		
	28 or 85 mrem/yr	Within 10,000 years All Other Cases		
	2000 or 3400 mrem/yr	Within 100,000 years All Cases (average time of peak is 34,000 years)		
Probabilistic (peak of mean)	4.8 mrem/yr	Within 10,000 years Case A	639 mrem/yr	Within 10,000 years Chronic exposure Case A
	9.5 (14)# mrem/yr	Within 10,000 years All Cases		
	345 mrem/yr%	Peak Dose Around 38,000 years All Cases		
	110^ and 450 mrem/yr 49 and 220 mrem/yr 2.8 and 73 mrem/yr	10,000 and 20,000 years Case D Case E Case F		

*Two values are provided (i) considering well completion uncertainty in either the UTR or Gordon Aquifers (GA) or (ii) assuming wells are only completed at the point of maximum exposure in UTR (Gordon Aquifer concentrations and doses are by default 20 times lower than UTR).
#DOE provides the value of 9.5 mrem/yr in its PA (SRS-REG-2007-00002, Rev. 1) for well 6. Review of the Goldsim modeling files shows a value of 0.14 mSv/yr (14 mrem/yr).
%From DOE's probabilistic GoldSim model SRS FTF v2.4 100 ky All Cases r1000 s1.gsm
^RAI response RAI-UA-4 indicates the peak dose is 1.1 mSv/yr (110 mrem/yr), while PA Table 5.6-10 (SRS-REG-2007-00002, Rev. 1) indicates 1.01 mSv/yr (101 mrem/yr) for well 6.

Figure 4-2 Primary Barriers Delaying Timing of Peak Dose for Four Key Radionuclides



Notes: 1. Radium releases are complicated by in-growth from parent radionuclides of differing mobility. 2. Some radionuclide releases are sharp (e.g., Tc), while other radionuclide releases are prolonged (e.g., Pu)—no attempt was made to indicate the shape (or time interval) of the peak radionuclide doses at the 100 m (330 ft) well. 3. Only general information is provided; timings are approximate.

Table 4-4 Barriers Limiting Timing and Magnitude of Peak Dose

Barrier	Timing Tank Type			Magnitude Tank Type		
	I	III/IIIA	IV	I	III/IIIA	IV
Closure Cap%						
Steel Liner				*	*	
Tank Grout (Hydraulic)						*
	Radioelement			Radioelement		
	Tc	Pu	Np	Tc	Pu	Np
Tank Grout and Contaminated Zone (Chemical)				^		
Basemat						
Natural System Dilution and Dispersion						
Natural System Sorption						

 to  = 1000s to Tens of 1000s of year delay in timing of peak dose

 to  = Factor of 2 to 10 reduction in peak dose

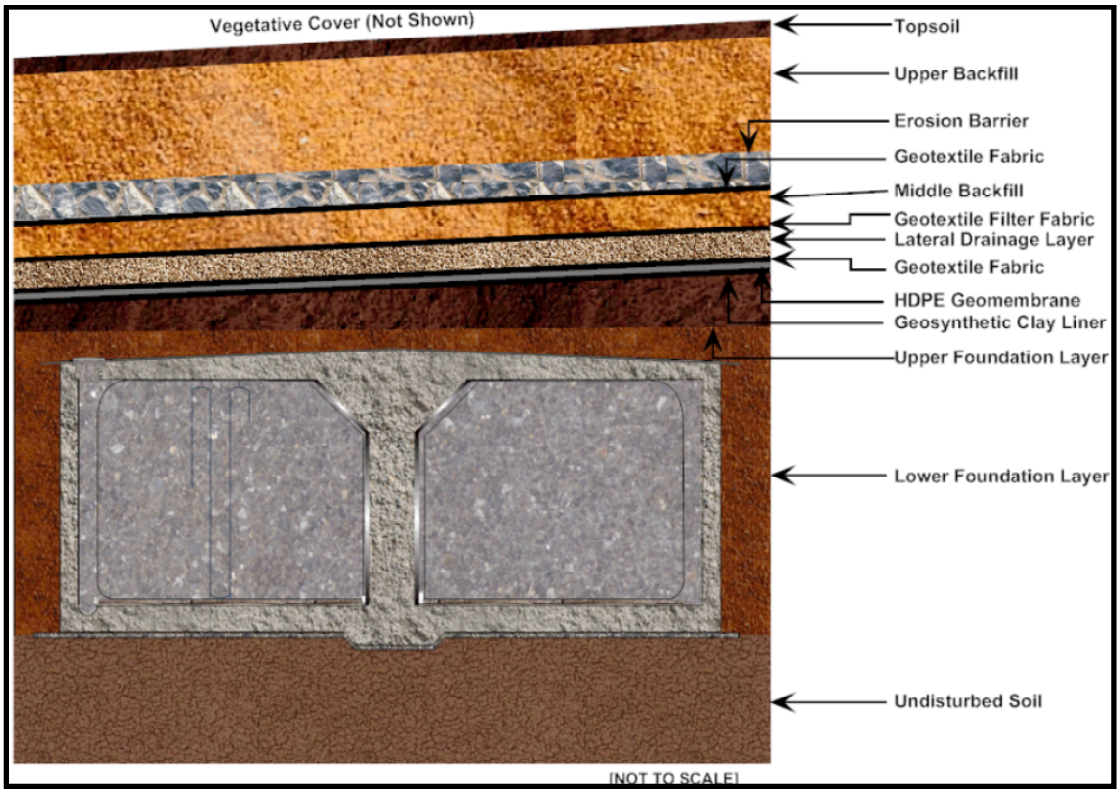
Note: Only general information is provided to provide an indication of relative importance of various barriers mitigating FTF releases.
%Generally, the cover is a redundant and less effective barrier compared to the steel liner.
* Longevity of barrier is significant relative to half-life of Pu and would therefore have noticeable effect on magnitude of Pu peak dose. Radium peak doses generally increase due to hold-up of parent radionuclides in the tanks.
^Reduces releases by orders of magnitude until transition to no solubility control at Oxidized, Region III (see Section 4.2.8 for more information about chemical transitions).

4.2.4 Infiltration and Erosion Control

Tank farm closure will involve pouring a tank-waste-stabilizing grout layer between the residual tank bottom waste and overlying tank roofs. Once grouting has been completed, a multilayer closure cap and drainage system will be installed. The closure cap is designed to: (1) limit infiltration by promoting runoff, evapotranspiration, and the shedding of water around the tanks and ancillary equipment; (2) provide physical stabilization of the site and (3) act as an intruder

deterrent. Performance is dependent on the various closure cap layers that are shown in Figure 4-3 and described in Section 4.3 of (WSRC-STI-2007-00184). After installation of the closure cap: (i) an initial 100-yr institutional control period will begin, during which active maintenance will be conducted to prevent pine forest succession and to repair any significant erosion; (ii) followed by a 10,000-yr post closure compliance period, with no active maintenance. The current closure cap design is preliminary and will be finalized closer to the time of FTF closure.

Figure 4-3 Conceptual Model of FTF Closure Cap



4.2.4.1 Infiltration

Limiting infiltration reduces the potential for advective transport of radionuclides. However, in the FTF PA barrier analysis, the closure cap was shown to have a minimal impact on peak radionuclide doses within 10,000 yrs. Infiltration rates reached steady-state conditions by 2,623 years after closure which is before the assumed timing of steel liner failure for all tank types in the Base case. Figure 4-2 illustrates the closure cap as a redundant barrier. In the absence of site-specific cap results in the PA, DOE assumed that, 2,623 years after closure, pine tree root penetrations will degrade the engineered closure cap. This will result in a steady-state infiltration rate after year 2,623 ranging from 27.05 to 29.6 cm/yr (10.65 to 11.67 in/yr).

DOE tested several closure cap configurations with the HELP model and selected a final conceptual design configuration that resulted in the least amount of net infiltration and provided for physical site-stability with an erosion barrier. This configuration consists of a low permeability composite hydraulic barrier layer with an overlying coarse sand lateral drainage layer, a riprap erosion barrier layer, and a vegetative cover and topsoil. The vegetative cover and topsoil will be selected to promote runoff and evaporation while minimizing erosion. The initial vegetative cover will be a persistent grass, such as Bahia (*Paspalum notatum*), that may

later be replaced with bamboo, if DOE determines that bamboo will slow invasion of loblolly pine trees. The proposed composite hydraulic barrier is a high-density polyethylene geomembrane underlain by a GCL. The GCL is assumed to plug all holes that may develop in the geomembrane until the holes have been penetrated by loblolly pine tap roots. DOE considered a range of potential closure cap degradation mechanisms which are discussed in Section 3.2.4.7 of the PA (SRS-REG-2007-00002, Rev. 1).

DOE ran one hundred HELP model simulations of the selected engineered closure cap design, with precipitation ranging from 75.7 to 174.2 cm/yr (29.8 to 68.6 in/yr) for the initial (i.e., Year 0) intact condition, with resulting water balance components as shown in Figure 4-4. The performance of the engineered closure cap was then simulated with the HELP model to estimate degradation-dependent infiltration rates below the cap throughout the compliance period with the results shown in Table 3.2-10 of the PA (SRS-REG-2007-00002, Rev. 1). Estimated average annual net infiltration through the closure cap was then used as an upper boundary condition to the FTF vadose zone model. DOE also simulated an intact soils-only closure cap for comparison to background water balance conditions. These HELP model water balance results, with a median infiltration rate of 40.4 cm/yr (15.9 in/yr), are comparable with the historical background infiltration rate of 37.7 cm/yr (14.85 in/yr).

4.2.4.2 Erosion Control

The ability of the closure cap to reduce infiltration and deter intrusion for long time periods is dependent on erosion controls. The closure cap design includes a minimum of 3 m (9.8 ft) of material above the waste tanks and significant ancillary equipment. Typically, agricultural and resident intruder scenarios consider a nominal excavation depth of 3 m (9.8 ft). Therefore, proper design, construction, and performance of the erosion barrier should limit surface water erosion and direct contact with the waste by intruders. The erosion barrier is designed to limit erosion to the underlying cap layers, however the vegetative cover, topsoil, and upper backfill layer, which provide water storage and promote evapotranspiration, are susceptible to erosion. One of the phenomena most likely to affect long-term stability is surface water erosion. DOE performed scoping-level calculations to design the cap layers to prevent gully formation and ensure that soil loss would not impact closure cap performance.

DOE evaluated the physical stability of the closure cap with respect to a probable maximum precipitation (PMP) event, consistent with NUREG-1623 (NRC, 2002). The PMP is defined as the theoretically greatest depth of precipitation that is physically possible during a given period of time over a given area at a particular geographic location. Based on the PMP, the design criteria for the vegetative cover, erosion barrier, side slopes, and toe of the side slopes was evaluated. Although the methodology presented in NUREG-1623 addresses a 1,000 year timeframe, DOE stated that the SRS-specific PMP event provides assurance of closure cap stability against gullying for the 10,000 year compliance period.

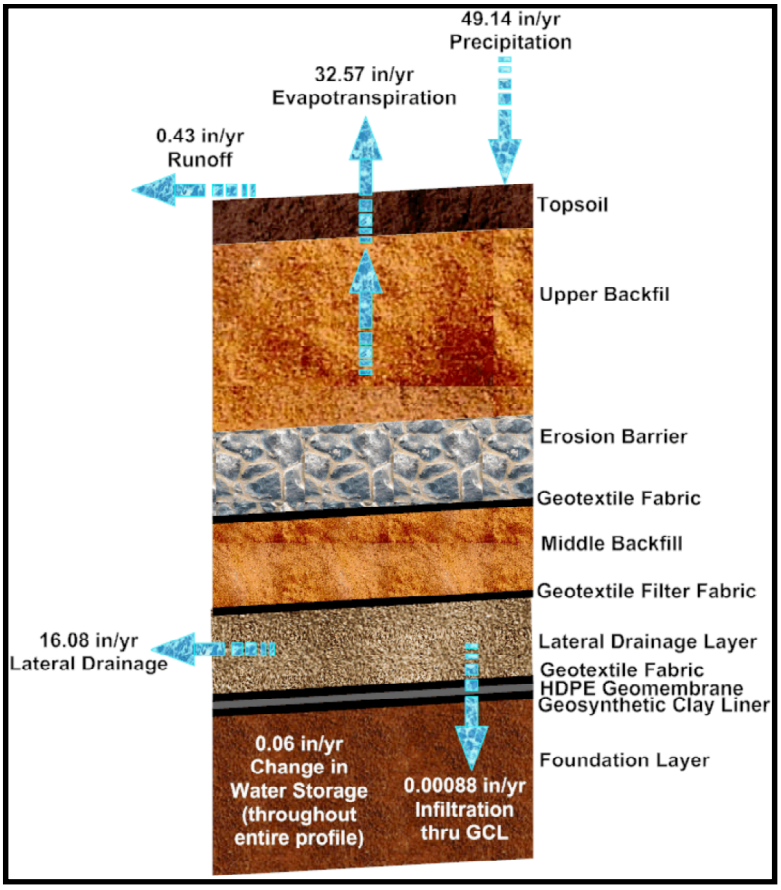
Closure cap modeling assumes that the erosion barrier will prevent animal intrusion into the lower layers, but will not prevent root penetration from pine trees. Root penetration and decay may create access paths for burrowing animals through the erosion barrier, however, it's assumed that by that time period the High Density Polyethylene (HDPE) and GCL layers are degraded by other mechanisms. Accordingly, animal burrowing was not considered a significant risk.

The projected long-term topsoil loss was determined according to the Universal Soil Loss Equation for both vegetative cover conditions (i.e., bahia grass and pine forest). DOE predicted

approximately 3.7 cm (1.45 in) of soil loss for the topsoil and no reduction in the upper backfill layer within 10,000 yrs, as discussed in Section 8.2 of WSRC-STI-2007-00184.

Riprap for the integrated drainage system ditches has not been sized because, given the early phase of the project, a detailed closure cap drainage system layout is not yet available. Riprap material for the erosion barrier, side slope, and toe of the side slope will be selected from local granite or mylonitic quartzite quarries.

Figure 4-4 Cap Water Balance



4.2.5 NRC Evaluation of Infiltration and Erosion Control

The closure cap design information provided by DOE is preliminary and will be finalized closer to the actual time of closure of FTF. NRC staff's review results are based largely on the conservatism associated with the proposed designs. DOE should ensure that appropriate conservatism is maintained with future design changes. It should be noted that the design for long-term infiltration control and long-term erosion control have different objectives and are subject to different degradation mechanisms. Acceptability of a design for one does not necessarily mean an acceptable design has been achieved for the other (e.g., designing the vegetative cover and topsoil to promote runoff may reduce infiltration in the near-term, but may increase long-term erosion). Sections 4.2, 4.3.1, 4.3.2, 4.4, and 4.6 of NUREG-1854, (NRC, 2007) contain the staff's review criteria pertaining to infiltration and erosion controls.

4.2.5.1 NRC Evaluation of Infiltration

Net infiltration is usually a sensitive parameter value in PAs because it is directly related to the flux of contaminants into the groundwater. However, DOE's sensitivity and uncertainty analyses did not identify any risk-significant closure cap parameters as a result of the modeled performance of the steel tank liners. Assumed degradation of the closure cap results in a steady-state infiltration by year 2,623 after closure, because of degradation processes. Accordingly, the cap would have a minimal impact on peak doses because base case steel tank liner failure dates occur after 2,500 yrs. If the assumptions about the performance of the vaults and steel liner are optimistic, the closure cap would become a more risk-significant barrier. This conclusion is supported by DOE's barrier analysis as presented in the Rev. 1 PA (SRS-REG-2007-00002, Rev. 1).

The assumed saturated hydraulic conductivity for the upper foundation layer of 1.0×10^{-6} cm/s (3.3×10^{-8} ft/s) constrains the HELP model to a maximum value of only 31.6 cm/yr (12.45 in/yr). Consequently, the long-term steady-state infiltration rate, which ranges from 27.05 to 29.6 cm/yr (10.65 to 11.67 in/yr), is less than the background value of 37.7 cm/yr (14.85 in/yr). DOE indicated that the assumed saturated hydraulic conductivity for the upper foundation layer is reasonable, as the value is typical for soil-bentonite blends. Although an increase from DOE's assumed long-term infiltration rate to the background value is not as risk-significant as other barriers, it would result in an increase in radionuclide release and a decrease in the timing of the chemical transitions for the contaminated zone. Accordingly, in the final cap design, DOE should further evaluate the long-term saturated hydraulic conductivity of this layer.

The lateral drainage layer is to consist of a 0.03 m (1 ft) thick layer of coarse sand with a high hydraulic conductivity that is designed to divert a significant portion of the infiltrating water away from the underlying tanks and ancillary equipment. A geotextile filter fabric will be placed on top of the lateral drainage layer to provide filtration between the underlying sand layer and the overlying middle backfill. DOE assumed that the degradation of the drainage layer (i.e., a reduction in hydraulic conductivity) is controlled by colloidal infilling of the pore space within the lateral drainage layer from the overlying backfill. DOE's approach did not take credit for the potential flushing of colloids out of the lateral drainage layer or any colloids that may be attenuated by the filter fabric. However, the predicted rate of infilling may not adequately account for the potential migration of larger particles into the lateral drainage layer. The process of infilling of the lateral drainage layer with larger particles would likely accelerate the rate of degradation of the drainage layer and infiltration through the closure cap could increase at earlier time periods.

DOE modeling indicated that saturated conditions will occur above the composite layer. An initial hydraulic head of 15.60 cm (6.14 in) is predicted on top of the HDPE geomembrane that increases until year 2,623 where it ranges from 72.44 to 77.09 cm (28.52 to 30.35 in) throughout the remainder of the 10,000 yr compliance period. DOE stated that conservative modeling assumptions (e.g., depth of evapotranspiration and degradation of the lateral drainage layer) led to an average annual head on the HDPE geomembrane that is greater than they anticipate. Should the buildup of hydraulic head occur, DOE does not believe it would adversely impact the physical stability of the closure cap, vegetation, erosion, or the performance of the composite layer. Based on limited model support, it is difficult to assess: (1) the likelihood hydraulic head buildup within the cover; or (2) its potential implications for closure cap performance.

Although DOE indicated the closure cap has a minimal impact on peak doses, it may be more risk-significant if the performance of additional barriers is determined to be optimistic. NRC staff

thinks that DOE's approach to assessing closure cap performance is reasonable for planning purposes. Due to the uncertainty in the modeled processes, additional model support is needed to ensure that the modeled system adequately represents the real-world system. In addition, a robust quality assurance/quality control program for the closure cap is important for cap performance. In particular, the ability of the composite hydraulic barrier to limit infiltration early in the compliance period is dependent on construction quality.

4.2.5.2 NRC Evaluation of Erosion

Erosion control is necessary to ensure that a thick cover of soil is maintained over the waste to eliminate exposure pathways to hypothetical inadvertent intruders and provide suitable conditions for the vegetative cover and topsoil. To mitigate the potential effects of surface water erosion, it is important that erosion protection designs be based on an appropriately conservative rainfall event. DOE's determination of the PMP event and the corresponding design criteria for the vegetative layer, erosion barrier, side slopes, and toe of the side slopes were consistent with NUREG-1623 (NRC, 2002). This guidance document specifically addresses a 1,000 year timeframe rather than the 10,000 year compliance period. DOE stated that the SRS-specific PMP has a low probability of occurrence and is a bounding event, thereby providing assurance of physical stability of the closure cap design for the 10,000-year compliance period. No recurrence interval is normally assigned to the PMP; however, NRC staff believes that the probability of such an event being equaled or exceeded is very low.

Long-term maintenance of the topsoil and vegetative cover is important to closure cap performance as evapotranspiration (82.73 to 84.22 cm/yr (32.57 to 33.16 in/yr)) dominates the modeled water balance distribution for SRS precipitation (124.82 cm (49.14 in)). The Universal Soil Loss Equation was used to predict topsoil losses throughout the compliance period. However, those predictions may not adequately account for complex and uncertain processes which may contribute to soil loss and initiation of gullying over long-time periods.

Although the closure cap design has not been finalized, a preliminary evaluation of erosion protection designs (e.g., evaluation of an acceptable rock source, the ability of an integrated drainage system to accommodate design features, etc.) may be important in the overall evaluation of closure cap performance.

DOE has provided sufficient information for NRC staff to conclude that the designed closure cap can provide adequate long-term erosion protection. The processes being modeled are highly uncertain, and adequate justification is needed to ensure that the modeling has appropriately accounted for these uncertainties. DOE should demonstrate that model predictions for the final cover design are sufficiently conservative, based on the amount of model support.

4.2.5.3 Infiltration and Erosion Review Results and Recommendations

With respect to infiltration and erosion, NRC staff notes the following:

1. DOE's approach to assessing closure cap performance is reasonable for planning purposes.

2. DOE has provided sufficient information regarding long-term erosion protection of the closure cap.

Notwithstanding 1 and 2 above, NRC staff would also note that the processes being modeled are highly uncertain. Therefore, adequate justification is needed to ensure that the modeling has appropriately accounted for these uncertainties. DOE will need to demonstrate that model predictions for the final cover design are sufficiently conservative, based on the amount of model support provided.

Recommendations for infiltration and erosion include the following, along with a rating of risk-significance and priority (see the footnote in Section 4.2.3.2 for a description of the risk-significance and priority of the recommendations):

1. Additional model support should be provided for: (i) the long-term hydraulic conductivity of the upper foundation layer and lateral drainage layer; and (ii) the long-term erosion of the topsoil layer. (Low Risk-Significance, Intermediate Term [Prior to Final Closure])

2. Prior to completing the final closure cap design, a preliminary evaluation of erosion protection designs (e.g., assessment of an acceptable rock source, the ability of an integrated drainage system to accommodate design features, etc.) should be conducted. (Low Risk-Significance, Intermediate Term [Prior to Final Closure])

4.2.6 Radionuclide Inventory

Chapter 3 (Section 3.1) first discusses radionuclide inventory because it is relevant to the Criterion 2 as well as the Criterion 3 evaluation. Section 3.1, provides information on (1) the screening process used to identify radionuclides for inclusion in the FTF inventory, (2) the development of the inventories for those radionuclides expected to be present in FTF tanks in risk-significant quantities and (3) the process for characterizing and evaluating the final inventories after the tanks are cleaned.

In summary, DOE took what it describes as a conservative approach to developing tank and auxiliary equipment inventories for the FTF facility consistent with previous NRC staff recommendations in scoping and PA review comments. For example, DOE adjusted its inventory estimates an order of magnitude higher in the Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) as compared to the initial inventory presented in Rev. 0 PA (SRS-REG-2007-00002, Rev. 0) for Type I and IIIA tank types.

Inventory can be more or less risk-significant considering the mix of radionuclides present in the waste and the expected chemical environment of the contaminated zone. For example, inventory is more risk-significant for those radionuclides that are not solubility limited in the expected iron rich, high pH, and low Eh environment initially present in the FTF facility tanks. For those radionuclides that are solubility limited, the potential dose contributions of those radionuclides are limited, although inventory may be important with respect to mass depletion (e.g., radionuclide mass may slowly leach out of the tanks at non-risk-significant levels, limiting or completely eliminating the impact of a later release at higher solubility limits). It is important to note that many key radionuclides at the FTF are, in fact, expected to be constrained to low aqueous phase concentrations or solubility limits for long periods of time well beyond the period of performance in some cases (e.g., Np, Tc, and Pu). It is also significant to note that the Np inventory is expected to be depleted prior to chemical transition to a higher, more risk-significant, solubility. Thus, the assumptions regarding the initial inventory for Np, as well as the solubility limits for Np and the time to chemical transition, all become important considerations in the compliance demonstration. Another key radionuclide, Ra-226, is not expected to be present initially in the tanks at risk-significant quantities but is expected to be produced over time by its

immediate parent Th-230, and from Th-230's predecessors U-234 and Pu-238, some of which are present in risk-significant quantities in the HLW tanks. Thus, little uncertainty may be associated with the Ra-226 inventory, although certainly the inventory of its parents may be an important consideration when assessing compliance. Of course, there is always the risk that the waste stream has a key radionuclide in FTF for which no inventory has been derived or has been grossly underestimated such that the true risk posed by the radionuclide may not be revealed in DOE's PA.

For all tanks that have not yet been cleaned or were in the process of being cleaned at the time of PA preparation (Type I Tanks 1-8; Type IIIA Tanks 25-28 and 44-47; and Type III Tanks 33-34), DOE developed a final inventory using the maximum concentration of any tank within a tank type grouping (e.g., Type I, III, IIIA). Concentrations of radionuclides in the tanks were determined from the WCS or via special calculation. With respect to the expected residual volume to remain following waste retrieval, DOE initially estimated that 1.6×10^{-3} m (0.06 in) would remain in the tanks. To account for uncertainty in the expected cleaning effectiveness, DOE also increased the estimated residual inventory for each tank by one order of magnitude. This increase in the inventory could be thought of as uncertainty in the concentration of residual waste in the tank or the amount (volume) of waste that can be removed from the tank based on uncertainties about cleaning effectiveness. The order of magnitude increase was not applied to Type III Tanks 33 and 34, as these tanks do not contain cooling coils; therefore, DOE expects that it will be able to effectively clean these tanks with a greater degree of certainty. Finally, the probabilistic analysis also considered uncertainty in the inventory estimates. The distribution is skewed low for tanks that have not yet been cleaned and for which the inventory estimates were revised upwards by an order of magnitude to account for additional conservatism in the inventory estimates.

For Type IV tanks that have been cleaned, Tanks 17 and 20, DOE developed an inventory based on post-waste retrieval data, although final characterization data were not available during Rev. 1 PA preparation for Tanks 18 and 19. DOE provided additional information in response to RAI-MEP-2 and CC-IN-1 (DOE, 2011) about the final inventory estimates and the breakdown of sampling and measurement uncertainty for Type IV Tanks 18 and 19, as additional information to finalize the inventory has become available since development of the Rev. 1 PA. In response to RAI-MEP-2, DOE also provided results of special calculations used to evaluate the impact of the final inventory estimates on the results in Rev. 1 PA. While DOE also considered final inventories and uncertainty in the inventory estimates for previously cleaned and grouted Type IV Tanks, 17 and 20, NRCs review does not focus on these tanks because the inventory of these tanks is relatively low compared to other tanks. Section 3.1 of this TER contains additional details regarding the development of inventories for cleaned tanks and the estimated inventories for tanks that have yet to be cleaned.

4.2.7 NRC Evaluation of Radionuclide Inventory

As stated above, in the absence of final estimates of residual radioactivity remaining in cleaned tanks, DOE attempted to develop conservative inventories for tanks that have yet to be cleaned. This approach is preferable because it is easier to defend and at the same time may not significantly affect the PA results, as many of the key radionuclides driving facility risk are solubility limited. DOE assumes some measure of risk in a Type 2 decision error (assuming that the performance objectives can be met when they cannot) if it (1) underestimates inventories for non-solubility limiting radionuclides, or if, in the end, (2) it cannot defend solubility limits (e.g., Tc-99) or, (3) significant mass depletion of HRRs occurs before transition to a higher solubility limiting phase and if the lengthy chemical transition times assumed in the PA modeling prove

overly optimistic. For example, mass depletion appears to affect releases associated with Np-237 (see sensitivity analysis results on page 694 of DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1). However, compliance risk is mitigated if DOE follows through with its commitment to develop post-retrieval inventory estimates based on sampling and evaluates the dose impacts associated with the final FTF inventory considering uncertainty in the final inventory and PA modeling assumptions (e.g., solubility assumptions). Therefore, NRC will follow DOE's commitment to characterize cleaned tanks following waste retrieval into the monitoring period.

In Section 3.2 of this TER, NRC staff reviews DOE's approach to developing the final inventory for Tanks 18 and 19. Section 3.2 also lists NRC staff's review results and recommendations with respect to DOE's plans to sample HRRs and to consider uncertainty in final inventory estimates (e.g., alternative methods to evaluate uncertainty in volumetric measurements) for tanks that have yet to be cleaned or for which final inventory estimates have not yet been developed (i.e., Tanks 5 and 6). Consult Section 3.2 for additional information.

NRC staff note the following with respect to inventory development:

- DOE's approach to developing inventories for FTF sources is reasonable and generally tends to err on the side of conservatism for tanks that have yet to be cleaned.

NRC staff makes the following recommendation with respect to inventory development:

- NRC staff supports DOE's commitment to sample each tank following waste retrieval activities and will follow-up with DOE on sampling and analysis of cleaned tanks during the monitoring period. (High to Moderate Risk-Significance, Short and Intermediate-Term)

4.2.8 Release and Near-Field Transport

DOE assesses compliance with the performance objectives in 10 CFR Part 61 Subpart C by considering two primary sources of contaminants: waste tanks and ancillary equipment. Waste tanks refer to the 22 subsurface tanks in the FTF that were used to store aqueous wastes, and ancillary equipment refers to equipment used in the FTF to transfer waste (e.g., transfer lines, pump tanks) and reduce waste volume (e.g., evaporator systems). Sections 1.1 and 3.1.3 provide descriptions of the tanks and ancillary equipment, respectively.

DOE states that multiple elements of the tank design will serve to minimize water infiltration. The waste tank concrete vaults and steel liners are expected to significantly retard water flow through the waste tanks. The cement-based material filling the emptied tanks and the FTF closure cap covering the tanks and ancillary equipment will provide additional barriers to water infiltration. DOE does not plan to grout the ancillary equipment, but expects the steel wall liners to significantly retard water flow into the ancillary equipment.

DOE uses the ISCM (See Section 4.2.2) to simulate the release of radiological and chemical constituents from the 22 underground waste tanks and associated ancillary equipment, and to simulate the migration of the contaminants through soil and groundwater (See Figure 4-1). The ISCM comprises simplified representations of the actual physical infrastructure of the waste tanks. These simplified representations include (1) depicting each discrete waste tank segment or area as homogeneous (i.e., ignoring interior elements, such as rebar and cooling coils and/or penetrations, such as tank risers and transfer lines), (2) using minimum segment thicknesses in the baseline analysis for areas with variable thicknesses (e.g., waste tank walls, tank tops); and

(3) assuming that the planned grouting of tank void areas (e.g., waste tank primary cavity, tank annulus, cooling coils) has occurred.

In the ISCM, water flow leaving the closure cap travels to the concrete waste tank top. Some flow is directed around the concrete top into the surrounding soil and some travels downward through the concrete. The ISCM assumes hydraulic conductivity of the concrete for both its initial and fully degraded states, with a linear increase in hydraulic conductivity with time between the initial and fully degraded states. DOE used information from a cementitious materials degradation analysis, summarized in Section 4.2.8.1., to derive the time for the concrete to reach a fully degraded state.

After passing through the concrete waste tank top, the water travels into the tank grout (for Type IV tanks and Type I, III, IIIA tanks after liner failure), or reaches the steel tank top liner (for Type I, III, IIIA tanks before liner failure). Before steel liner failure, the liner is modeled as impermeable to both advection and diffusion and water flow is diverted around the liner. After failure, the liner no longer serves as a barrier and has no further effect on the model results. DOE estimates the steel liner failure time for each tank type from a corrosion analysis, summarized in Section 4.2.8.2.

Water that enters the waste tank grout travels downward to the contaminated zone (CZ) at the bottom of the tank. The grout material properties are modeled as changing with time. Grout hydraulic conductivities are assumed for the grout's initial state and fully degraded states. The time to reach a fully degraded state is derived from a cementitious materials degradation analysis and the hydraulic conductivity is assumed to increase linearly with time between the initial and fully degraded states. Sorption parameters (i.e., K_ds) for the cementitious materials vary with the "age" of the materials. The "age" of the cementitious materials is dependent on the pH of the pore water, which in turn is dependent on the amount of water (number of pore volumes) that passes through the cementitious material over time. The grout hydraulic conductivity influences the water flow rate through the waste tank. Faster grout degradation causes the flow rate through the waste tank to reach steady state sooner. The grout K_ds affects the model results only when contaminants move upward from the CZ into the grout which can occur by diffusion in all tank types or by advection in Type IV tanks, which lack a steel liner at the top.

The ISCM flow and transport model assumes that the waste tank residual inventory is contained within a thin CZ layer at the grouted tank bottom. Water that passes through the tank grout leaches radionuclides in the CZ. The model accounts for the evolution with time of the physical and chemical conditions within the grouted waste tanks and its effect on radionuclide leaching from the CZ. For modeling purposes, the physical (e.g., porosity, hydraulic conductivity) and chemical (e.g., K_ds) properties of the CZ material used for the ISCM are assumed to be the same as those of the fill grout. Contaminant release from the CZ is assumed to be controlled by a fixed concentration limit (or solubility, in the broad sense of the term) and is tied to the chemical properties (e.g., pH, E_h) of the pore fluid passing through and interacting with the overlying tank grout. As more water passes through the grout, the pore fluid chemistry changes. Section 4.2.8.3 of this TER summarizes DOE's modeling of pore fluid chemistry and radionuclide release.

Transport of contaminants into the overlying tank grout through diffusion or advection is controlled by grout K_ds (see Section 4.2.8.4); no solubility control is assumed for cementitious materials. Contaminants released from the CZ do not leave the tank until the tank liner fails. DOE estimated the liner failure time for each tank type from a separate analysis (see Section

4.2.8.2) that is independent of the flow and transport model. The analysis assumes that the primary liner and secondary liner (if present) fail simultaneously. After contaminants exit the waste tank liner, they enter the concrete basemat below the tank liner. The basemat hydraulic conductivity and K_ds are modeled as changing with time. Hydraulic conductivities of the basemat are assumed for its initial and fully degraded states. The time to reach a fully degraded state is derived from a cementitious materials degradation analysis and a linear increase in hydraulic conductivity with time is assumed. In some configurations, fast flow paths through the basemat are modeled that result in a higher flow rate through the basemat. The model accounts for sorption onto basemat concrete, which retards contaminant transport. The cementitious materials K_ds vary with the concrete "age", vis-a-vis the concrete pore water pH, which in turn depends on the amount of water that passes through the concrete over time (see Section 4.2.8.3). After contaminants exit the basemat, they enter the vadose zone (e.g., soil) beneath the waste tank. DOE modeled the flow and transport of the near-field and vadose zone domain for each FTF waste source individually within PORFLOW to determine the source-specific, time-dependent radionuclide contaminant flux entering the water table. Section 4.2.8.4 summarizes DOE's modeling of the near-field and vadose zone.

For modeling purposes, DOE assumed the radionuclide inventory in ancillary equipment is within a secondary containment buried in soil. The pump tanks, catch tank, evaporator pots, and CTS pump tank are modeled as point sources located in the FTF at a central point of the individual components. The inventory associated with these waste sources is assumed to be surrounded by a stainless steel vessel and release is predicated upon steel failure. The transfer line inventory is modeled by distributing the assumed inventory uniformly throughout the FTF modeling cells. DOE did not explicitly model other FTF ancillary equipment, such as diversion boxes (FDB-1, FDB-2, FDB-3, FDB-4, FDB-5, FDB- 6), valve boxes (Valve Boxes 1-5, 28A and 28B, and LDB-17), pump pits, evaporator cells, and overheads tanks because these locations have not served as primary waste containment and, therefore, are not expected to contain significant radiological inventory at closure.

4.2.8.1 Cementitious Materials Degradation

In the ISCM, the rate and timing of degradation of the waste tank cementitious materials are based on analyses presented in WSRC-STI-2007-00607 and SRS-REG-2007-00027, and can vary with tank type. WSRC-STI-2007-00607 uses information from scientific literature to assess the potential for various degradation mechanisms to impact the FTF concrete and grout. The degradation mechanisms considered to be potentially important for the FTF cementitious materials are external sulfate attack, leaching, and carbonation. DOE did not consider internal sulfate attack in the analysis of grout and concrete vault degradation because literature data are reportedly insufficient to address the likelihood and consequences of this degradation mechanism. DOE also did not consider alkali-aggregate reaction because the aggregates used in SRS concrete and planned for use in tank grouts are not expected to be highly susceptible to this type of degradation. DOE also indicated that data from the literature is insufficient to address the likelihood and consequence of this mechanism on the degradation of the FTF cementitious materials. Finally, the analysis did not consider chloride-induced rebar corrosion because the concentration of chloride ions in SRS ground water and soil pore water is observed to be very low (approximately 2 parts per million (ppm)).

To model the degradation of FTF cementitious materials, DOE used simple empirical relationships and diffusion equations taken from published literature and reviewed in WSRC-STI-2007-00607. These relationships and equations allow the calculation of the depth of

penetration of potentially corrosive species into the cementitious material. The species involved in sulfate attack, leaching, and carbonation respectively are SO_4^{2-}, Ca^{2+}, and CO_2.

For Type I, III, and IIIA tanks, which contain cooling coils, carbonation is identified as the most aggressive degradation mechanism because it leads to enhanced corrosion of the cooling coil steel and forms expansive corrosion products that lead to cracking of the cementitious material. Concrete vault degradation is assumed to start once the carbonation effect has reached one-half the thickness of the concrete. Tank grout degradation is assumed to start once the carbonation effect has traversed the vault concrete and reached the grout. In addition, the tank annulus, primary liner and secondary liner are assumed to have a negligible impact on concrete/grout degradation, i.e., no credit is taken for the liners (see SRS-REG-2007-00027). The tank grout is assumed to be fully degraded once carbonation reaches half the grouted height. Tank grout degradation is calculated to begin as early as 2,600 years, 5,000 years, and 4,800 years for Type I, III, and IIIA tanks, respectively, with full degradation occurring after 13,000 years, 18,900 years, and 18,700 years, respectively.

For Type IV tanks, which have no cooling coils, leaching is considered as the major degradation mechanism for the tank grout, although carbonation is an important process for vault concrete degradation. Similar to Type I, III, and IIIA tanks, the tank grout degradation analysis is conducted independent of the tank liner performance, i.e., no credit is taken for the tank liner in calculating grout degradation. The tank grout is assumed to be fully degraded when the leaching reaches half the grouted height. Type IV tank grout degradation is calculated to begin as early as 800 years with full degradation occurring after 63,800 years.

Degradation of the concrete vault and tank grout results in increased hydraulic conductivities and effective diffusion coefficients of the cementitious materials. The PA assumes that the concrete and grout hydraulic conductivities increase by a factor of 100 as a result of degradation (see WSRC-STI-2007-00607). The effective diffusion coefficient for grout and concrete are assumed to increase by a factor of seven, which is approximately the same ratio as that between intact grout and concrete diffusion coefficient and backfill diffusion coefficient. Degraded grout and concrete is modeled as having the physical properties of the surrounding backfill.

Degradation of the grout and concrete vault also results in an evolution of the chemical environment in the cementitious material, which could affect radionuclide release from the CZ, discussed in Section 4.2.8.3.

The rate and timing of waste tank cementitious materials degradation described in the preceding paragraphs apply to DOE's deterministic assessment (Configuration A). In the probabilistic assessment, which also considers Configurations B, C, D, E, and F, DOE assumes that degradation of the cementitious materials begins at year 500, with full degradation occurring at year 501, and Configuration C, assumes that a fast flow path exists between the waste tank top and contaminated zone (e.g., from tank riser through cooling coil) and, in Configuration D, a fast flow path exists through the entire closed system (i.e., through a tank riser, through a cooling coil, through the tank fill grout, through pitting in the steel liner, and through the basemat).

4.2.8.2 Steel Degradation

The ISCM considers the integrity of the waste tank carbon steel liners and stainless steel transfer lines in modeling the release of contaminants. Before failure, the steel liners and

transfer lines are assumed to be impermeable with respect to both advection and diffusion. After failure, waste release occurs assuming the steel is absent or otherwise not a hindrance to advection and diffusion. With this approach the time of initial waste release is tied to the integrity of the waste tank primary liner and the transfer lines (For Type I, III and IIIA tanks with a secondary liner, the secondary liners were assumed to fail at the same time as the primary liner.) The rate and timing of failure of the tanks' steel liners and transfer lines due to corrosion are based on analyses presented in WSRC-STI-2007-00061 and WSRC-STI-2007-00460, respectively.

Carbon Steel Waste Tank Liners

DOE uses information from scientific literature to assess the potential effect of corrosion on the integrity of the carbon steel tank liners and to calculate the tank liner failure time (WSRC-STI-2007-00061). DOE also analyzed processes that affect the initial conditions of the liners at the time of closure including stress corrosion cracking and general corrosion. For example, three Type I tanks are known to have experienced stress corrosion cracking during their service lives. However, DOE analysis estimates that the effect on waste release is minimal because the crack area is small compared to the total surface area of the tank. DOE also indicated that none of the tanks have experienced general corrosion during service. Thus, DOE assumed that the tank initial conditions will not significantly affect the long-term performance of the tank under closure conditions.

In its base case scenario (Configuration A), DOE assumes a passive corrosion rate of 1 μm/yr (0.04 mil/yr) prior to carbonation- or chloride-induced steel depassivation that can lead to higher rates of corrosion. After steel depassivation, localized corrosion including pitting corrosion is excluded by DOE. In response to NRC's RAI-NF-2, DOE indicates that even if pitting corrosion occurs, the pit growth rate will decrease with time (DOE, 2011) and thinks that its probabilistic analysis encompasses other accelerated corrosion mechanisms such as pitting corrosion. In DOE's integrated PA, the entire liner is considered to have failed when the thinnest segment completely corrodes by general corrosion. DOE uses this liner failure model instead of an alternative conceptualization such as a patch model, in which the liner is discretized into patches that may fail individually over time, because discrete failure of the entire steel liner tends to maximize peak doses compared to a patch model that tends to result in temporal spreading of radionuclide releases (DOE, 2011).

Based on an empirical equation taken from the literature (Clear, 1976), the chloride-induced corrosion initiation time is modeled to be a function of the concrete thickness, water-to-cement ratio, and groundwater chloride concentration. After initiation, the corrosion propagation rate is controlled by oxygen diffusion through the concrete, which is a function of the oxygen diffusion coefficient in concrete, oxygen concentration in groundwater, and concrete thickness. Based on published carbonation analyses (Papadakis and Fardis, 1989; NUREG/CR-5542), the carbonation initiation time is modeled as a function of the carbon dioxide diffusion coefficient in concrete, total inorganic carbon in groundwater or soil moisture, calcium hydroxide bulk concentration in concrete solid, and the concrete thickness. After carbonation initiation, the carbon steel is assumed to be in an active corrosion propagation state with a corrosion rate of 254 μm/yr (10 mil/yr), which is much higher than the oxygen-controlled corrosion rate due to chloride depassivation assumed by DOE in its base case analysis. Thus, corrosion under grouted conditions was modeled as a competition between the initiation time induced by chloride depassivation and the initiation time induced by carbonation activation with post carbonation corrosion rates posing a much greater potential for consuming the thickness of the steel liner in a much shorter time. However, the initiation time for higher corrosion rates

associated with carbonation is generally longer compared to chloride-induced corrosion. (see discussion in Section 4.2.9.2).

In the deterministic approach, which is based on relevant information in the literature, the failure times are calculated by applying specific values to the aforementioned parameters to determine the initiation time and the oxygen-controlled corrosion propagation rate. In this approach, DOE assumed that corrosion propagates at an equivalent rate from both the interior and exterior surfaces of the tank wall. In response to RAI-NF-4, DOE thought that this approach partially accounts for uncertainties in input parameters and is more conservative (DOE, 2011). DOE also used stochastic approaches to estimate the liner failure time. In the stochastic approach, corrosion is assumed to proceed from only one side of the tank wall because DOE considered that the uncertainties in input parameters is accounted for in the distribution of the parameter values (DOE, 2011). In both approaches, it is assumed that the failure time is the sum of the initiation time (due to carbonation- or chloride-induced corrosion initiation) and the time for corrosion propagation through the wall. The propagation time is calculated from the remaining wall thickness at the initiation time divided by the propagation rate.

Several uncertain parameters are considered in the partial and comprehensive stochastic analysis including tank and concrete thicknesses, diffusion coefficients for O_2 and CO_2, and calcium hydroxide and chloride concentrations. The main differences between the partial and comprehensive analyses is that discrete distributions of carbon dioxide and oxygen diffusion coefficients are used in the partial analysis, whereas a continuous distribution of diffusion coefficients ranging from 1×10^{-10} to 1×10^{-1} cm^2/s is used in the comprehensive stochastic approach. Additionally, the passive corrosion rate is treated as an uncertain parameter (with a distribution ranging from 0.25 to 11 µm/yr (0.01 to 0.45 mil/yr) and with a median of 1 µm/yr (0.04 mil/yr) in the comprehensive stochastic analysis to account for other potential corrosion mechanisms (e.g., galvanic corrosion), faster transport pathways, faster corrosion at welds, stress corrosion cracking, and variability in the passive current density. The results of the analyses indicate that the oxygen and carbon dioxide diffusion coefficients are the critical parameters controlling the failure times. In the end, the partial stochastic analysis is used to define deterministic and stochastic steel liner failure times for use in the integrated PA using a diffusion coefficient of 1×10^{-6} cm^2/s (2 x 10^{-5} in^2/s) for Configurations A, B, and F or 1×10^{-4} cm^2/s (2 x 10^{-5} in^2/s) for Configurations C, D, and E with the median value of the steel liner failure time being used as the deterministic value. Table 4.2-35 in (SRS-REG-2007-00002, Rev. 1) indicates that the earliest time to failure using oxygen and carbon dioxide diffusion coefficients of 1×10^{-4} cm^2/s (2×10^{-5} in^2/s) is 75 years after FTF closure for Type IV tanks under Configurations C, D, and E. The latest liner failures are predicted to occur at 12,751 years after FTF closure for Type III/IIIA tanks under Configurations A, B, F assuming diffusion coefficients of 1×10^{-6} cm^2/s (2×10^{-7} in^2/s).

DOE performed the uncertainty and sensitivity analysis using a probabilistic model as discussed in Section 4.2.18. DOE assigns probabilities of 12.5 percent, 8.75 percent, and 18.75 percent for higher risk Configurations C, D, and E, for Type I, III, IIIA, and IV tanks, respectively (see Table 4-11). Because the probabilities of Configurations A, B, and F are much higher than those for Configurations C, D, and E, the steel liner failure time in the PA is dominated by the configurations with prolonged times to failure of the steel liners (Configurations A, B, and F).

Stainless Steel Transfer Lines

WSRC-STI-2007-00460 provides estimates for failure of the stainless steel transfer line core piping. Because less than 0.5 percent of the transfer lines are carbon steel, the PA model did not consider the earlier failure of the carbon steel transfer line.

The stainless steel transfer line failure time is estimated based on exposure to soil conditions. Both general corrosion and pitting corrosion are considered to be the degradation mechanisms. Data from the literature (Gerhold, 1981) under similar soil conditions show that both the general corrosion rate and pitting corrosion rate decrease with time. Based on published corrosion rate data, DOE used a general corrosion rate of 1 μm/yr (0.04 mil/yr) and a pitting corrosion rate of 25 μm/yr (1 mil/yr) as bounding cases to estimate failure times of transfer lines. Because the pitting corrosion rate is about 25 times higher than the general corrosion rate, pitting corrosion becomes the controlling degradation mechanism for the stainless steel transfer lines. Based on an empirical equation in the literature (Sullivan, 2003), the area breached by pitting corrosion is modeled to be a function of the number of pits per line, the maximum pit depth, and the transfer line wall thickness. The transfer line is considered to fail if pitting corrosion breaches 25 percent of the line wall thickness. As a result, the earliest transfer line failures are estimated to occur 510 years after FTF closure (for transfer lines with a minimum wall thickness of 0.295 cm (0.116in)). The latest transfer line failures are estimated to occur 532 years (for the transfer lines with the minimum wall thickness of 0.480 cm (0.189 in)). In the probabilistic PA analysis, the failure times are distributed between the time of the first pit penetration and the time of 100 percent pitting penetration. For example, the failure time of a transfer line with a minimum wall thickness of 0.295 cm (0.116in) is distributed between 116 and 2,900 years.

4.2.8.3 Source Term Release

As discussed in Section 4.2.8, the ISCM assumes that the residual waste occurs as a discrete layer, referred to as the CZ, at the bottom of the waste tanks after the tanks are filled with reducing grout. Infiltration from the surface that enters the waste tank provides the pore fluids that leach contaminants from the CZ. Radionuclide leaching from the CZ depends on the flow and chemical composition of pore fluid passing through the zone. As the grout-filled waste tanks “age”, the pore fluid chemistry also changes, which affects contaminant release from the CZ.

To account for the evolution of pore fluid chemistry and its effect on radionuclide release from the CZ, DOE used a conceptual model that considers four chemical states of the grout as it ages: (i) Reducing Region II, (ii) Reducing Region III, (iii) Oxidizing Region II, and (iv) Oxidizing Region III. The four chemical states are characterized by the pore water solution pH (i.e., Region II, Region III) and oxidation-reduction potential (E_h). DOE used Geochemist’s Workbench® to model the change in pore water composition (and associated radionuclide solubilities) over time. Initially, the pore water, buffered by the overlying grout, is expected to exhibit chemical conditions represented by Reducing Region II. As water passes through the waste tank, the pH and reducing capability of the grout is affected. The model assumes that the infiltrate has a chemical composition corresponding to that of average SRS area rainfall equilibrated with atmospheric oxygen. The geochemical model simulates the reaction of infiltrating water with grout components that include calcium-silicate-hydrate (C-S-H), hydrotalcite, gibbsite, pyrrhotite, hematite, and quartz. Geochemical modeling results indicate that the grout pore water E_h rises from a reducing −0.60 volts to −0.48 volts after 39 pore volumes, then maintains an E_h of −0.48 volts for 371 pore volumes, at which point the grout reducing capacity is exhausted and the E_h becomes oxidizing, rising to a value about +0.55

volts. Linear extrapolation of C-S-H dissolution, supported by separate geochemical modeling presented in WSRC-STI-2007-00544, Rev. 2, estimates that the grout pore water pH will remain near 11 for about 2,063 pore volumes, which is much longer than the period during which the pore water E_h will remain reducing. Thus, in the ISCM, the waste tank pore water chemistry changes from an initial Reducing Region II chemical state to Oxidizing Region II conditions after 371 pore volumes have passed through the tank grout and Oxidized Region III after 2,063 pore volumes have passed through the tank grout (i.e., Reduced Region III is by-passed).

In the ISCM, release of radionuclides from the CZ is either instantaneous or is controlled by imposing a concentration limit. As shown in Table 4.2-10 from DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1), instantaneous release is assumed for several elements: Bk, C (under most conditions), Cf, Cs, I, Nb, Pa, Rh, Se (under some conditions), Tc (under one set of conditions), and Te. For the other elements, the concentration of the radionuclide in water contacting the CZ is given a fixed value until the inventory is exhausted. This concentration limit is based either on (i) the solubility of a pure solid phase or (ii) on co-precipitation of the radionuclide as a trace element in an iron oxide solid (WSRC-STI-2007-00544, Rev. 2). When the first approach is taken (concentration based on the solubility of a pure solid phase), DOE states that the solubility controlling solid was chosen based on the likely presence of the phase under given conditions or, in the absence of firm data on the appropriate choice, based on a higher-solubility solid. For example, a hydroxide may be chosen over an oxide because oxides tend to have higher solubilities. The other method used for choosing a concentration limit is to model the concentration based on co-precipitation of the element with an Fe oxide (WSRC-STI-2007-00544, Rev. 2). With this approach, the concentration ratio between the radioelement and Fe in solution is assumed to be the same as in the solid, and the ratio in the solid phase is based on empirical data. DOE calculates the solubility of magnetite (reduced regions) and hematite (oxidized regions) in the particular chemical state and uses the resulting dissolved Fe concentration and the solid phase ratio to calculate the dissolved concentration of the radioelement.

4.2.8.4 Near-Field Flow and Transport Modeling

DOE modeled the flow and transport of the near-field and vadose zone domain for each FTF waste source individually within PORFLOW® to determine the source-specific, time-dependent radionuclide contaminant flux entering the water table. This section summarizes DOE's approaches for near-field flow and transport model construction, material properties, and model validation.

Near-Field and Vadose Zone Transport Model Construction and Boundary Conditions

DOE modeled the waste tanks and vadose zone transport in PORFLOW® within two-dimensional (*r*, *z*) axisymmetric cylindrical domains. Using symmetry, DOE places no-flow boundary conditions at the waste source centerlines and at the outer radius of the model domains, which extend 10 m (30 ft) beyond the perimeter of each waste tank. The vadose zone PORFLOW® model domains use approximately 5000 cells to represent the different waste sources. The vadose zone thickness between the basemats (Tank Type IV) or concrete slabs (Tank Types I, III/IIIA) underlying the tanks and the UTR-UZ water table ranges from 0.4 to 6.1 m (1.2 to 19.9 ft); however, this thickness is modeled as a constant for each tank type, based on the average lower vadose zone thickness for each group of tanks. Net infiltration rate and vadose zone thickness are naturally variable with time; however, short term variations in these parameters are not simulated. A time-dependent net infiltration flux that varies as a function of the degradation state of the engineered closure cap (See Section 4.2.4.2) is prescribed as the

upper boundary condition to the near-field and vadose zone PORFLOW® model. The lower boundary of the model coincides with the water table where pressure head is set to zero. DOE specifies an outflow boundary condition for radionuclide transport, whereby the flux leaving the vadose zone is by advection only.

Near-Field and Vadose Zone Transport Material Properties

Compacted excavated soil backfill was emplaced surrounding and 2.7 m (9 ft) above Type I tanks. Type IV tanks were immediately surrounded by vermiculite bags prior to backfilling with compacted excavated soil. DOE neglects to consider the material properties of vermiculite because the material volume is considered negligible (WSRC-STI-2006-00198). Compacted excavated soil backfill was also emplaced surrounding Type III and IIIA tanks, and above Type III and IV tanks. Type IIIA tanks were also backfilled at radii greater than 2.1 m (7 ft) with Controlled Low Strength Material (CLSM). DOE stated that use of this material as backfill "appears to be very limited" and can therefore be neglected in tank modeling (SRR-REG-2007-00002, Rev. 1, p. 316). DOE indicated that, from the outset, the hydraulic conductivities of the CLSM and the compacted excavated soil backfill are not so dissimilar as to focus infiltration through the compacted soil backfill zone adjacent the tanks. Time-dependent degradation of the CLSM should increase the saturated hydraulic conductivity of this medium and thereby further reduce the hydraulic property contrast between these two zones. The compacted, relatively fine-grained clayey or silty sand backfill used in the upper vadose zone is explicitly represented in the near-field PORFLOW model. DOE derived model parameters that represent compacted excavated soil backfill material properties based on laboratory measurements of controlled compacted backfill from other GSA sites (WSRC-STI-2006-00198). Backfill material properties are listed in Table 4-5. DOE assumed a saturated effective diffusion coefficient based upon literature values that are not specific to the GSA, but neglects hydrodynamic dispersion. DOE assumes vadose zone backfill material properties are constant, and do not change with time.

Table 4-5 Backfill Material Properties in PORFLOW® Vadose Zone Modeling

Material	Saturated Effective Diffusion Coefficient (cm^2/s)	Average Total Porosity (%)	Average Dry Bulk Density (g/cc)	Average Particle Density (g/cc)	Saturated Horizontal Hydraulic Conductivity (cm/s)	Saturated Vertical Hydraulic Conductivity (cm/s)
Backfill	5.3×10^{-6}	35	1.71	2.63	7.6×10^{-5}	4.1×10^{-5}

DOE derived model parameters that represent undisturbed vadose zone material properties below the tanks using data from the E-Area Low Level Waste Facility. E-Area subsurface materials are assumed to be similar to F-Area materials in geology and physiography (WSRC-STI-2006-00198). Undisturbed vadose zone material beneath the waste tanks is relatively coarse-grained sandy material of the lower vadose zone. Material properties assumed for the undisturbed vadose zone are listed in Table 4-6. DOE assumes a saturated effective diffusion coefficient based upon literature values that are not specific to the GSA, but neglects hydrodynamic dispersion. DOE assumed undisturbed vadose zone material properties are constant, and do not change with time.

Table 4-6 Undisturbed Material Properties in PORFLOW® Vadose Zone Modeling

Material	Saturated Effective Diffusion Coefficient (cm^2/s)	Average Total Porosity (%)	Average Dry Bulk Density (g/cc)	Average Particle Density (g/cc)	Saturated Horizontal Hydraulic Conductivity (cm/s)	Saturated Vertical Hydraulic Conductivity (cm/s)
Undisturbed Vadose Zone	5.3×10^{-6}	39	1.63	2.67	3.3×10^{-4}	9.1×10^{-5}

The bottom of the Type I tank excavations were filled with approximately 7.6 cm (3 in) of lean concrete to form a working surface for waste tank installation. Type IV tanks do not rest on a concrete working slab; however, a 5.1 cm (2 in) drainage and maintenance slab exists between the tanks. The bottom of the Tank Type III excavations were filled with a working slab that is a minimum of 10.2 cm (4 in) thick. The bottom of the Tank Type IIIA excavation was also filled with a working slab that is a minimum of 10.2 cm (4 in) thick; however, this slab was either broken up or punched through with 10.2 cm (4 in) diameter holes on 45.7 cm (18 in) centers prior to backfilling. Because the working slabs were not strengthened with rebar, DOE assumes heavy working equipment and tank construction and filling activities cracked the slabs such that they will not function to laterally divert water flow in the vadose zone. While estimates for the material properties of these low quality concrete working slabs exist, their presence is not explicitly modeled by DOE. Rather, the material in these locations is instead assumed to be undisturbed soil.

As radionuclides are released from the CZ, they migrate upward or downward by advection or diffusion. In either direction, the radionuclides must first migrate through a cementitious material—either tank grout or basemat concrete. Cementitious material K_ds are therefore used in the ISCM for radionuclide transport out of the tank. Table 4.2-33 in DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) shows the selected K_ds values for the various pore water chemical states reflecting the age of the cementitious materials. For purposes of assigning K_ds values, chemical conditions in the basemat concrete are assumed to be initially oxidized. Thus, only one transition from the initial Oxidized Region II to Oxidized Region III is simulated.

Once radionuclides are transported from the tanks or ancillary equipment, their vadose zone transport will be governed by the K_ds values shown in Table 4.2-29 of DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1). The values for undisturbed vadose zone soil underneath the tanks are the same as those selected for sandy sediment in the saturated zone, and backfill soil values surrounding the tanks in the upper vadose zone are the same as those for clayey sediments in the saturated zone. The vadose zone K_ds discuss ion, therefore, applies to the saturated zone K_ds discussion.

The sediment K_ds values were selected, as documented in several reports cited in Table 4.2.29 of (SRS-REG-2007-00002, Rev. 1), on the basis of site-specific data where possible, supplemented by analyses of literature data. For example, the Tc K_ds for sandy and clayey soils are based on the mean of 47 measurements of local samples (SRNL-TR-2009-00019, Table 1). Neptunium values are based on fairly limited data, whereas substantially more site-specific experiments on Pu have been conducted (WSRC-TR-2006-00004, Table 10). For Pu, the literature review produced two sets of values: one for Pu(III/IV), or reduced oxidation states, and one for Pu(V/VI), or oxidized oxidation states. For the PA, and as explained in WSRC-TR-2006-00004 Rev. 0, Table 10, a hybrid Pu K_d value is used, calculated with 10 percent of the

Pu(V/VI) value and 90 percent of the Pu(III/IV) value. For all elements in the deterministic analysis, DOE chose what were termed the "best" values, rather than conservative numbers.

DOE uses lognormal distributions with minimum and maximum values, and geometric standard deviations based on a factor of the geometric mean for (i) cementitious materials, (ii) vadose zone, and (iii) saturated zone sorption coefficients in its probabilistic assessment (Section 5.6.3.4, Table 5.6-4, SRS-REG-2007-00002, Rev. 1). The parameter distributions are calculated from a statistical analysis of samples from a single core collected in E-Area. In that analysis (WSRC-STI-2008-00285), K_ds were measured for several radioisotopes in 27 samples that represented the Upper Vadose, Lower Vadose, and Aquifer Zones. Neptunium was assigned specific minima and maxima, based on recent experimental results.

4.2.8.5 Near-Field and Vadose Zone Transport Model Validation

DOE used characterization and monitoring data from uncapped E-Area (adjacent to F-Area) to validate aspects of the PORFLOW® vadose zone model. These data included soil suction and water content, tracer test pore velocity, and tritium plume concentration. Soil suction data from E-Area indicate a range from 50 to 200 cm (20 to 79 in), and PORFLOW® vadose zone modeling produces upper vadose zone soil suction values of 83 cm (33 in) and lower vadose zone values of 170 cm (67 in) (SRS-REG-2007-00002, Rev. 1). Water content data suggest saturation ranges from 35 to 75 percent, and PORFLOW vadose zone modeling produces upper vadose zone saturation values of 91 percent and lower vadose zone values of 72 percent (SRS-REG-2007-00002, Rev. 1). Field and laboratory tracer test experiments indicate a pore velocity of 114 cm/yr (45 in/yr), and PORFLOW® vadose zone modeling produces upper vadose zone pore velocities of 86 cm/yr (34 in/yr) and lower vadose zone velocities of 109 cm/yr (43 in/yr) (SRS-REG-2007-00002, Rev. 1). Comparison between the measured tritium plume concentration from a disposal trench at E-Area and PORFLOW® vadose zone model results do not serve to validate the model, but PORFLOW® results are generally consistent with measured data and the vadose zone model is not invalidated by the data (SRS-REG-2007-00002, Rev. 1).

4.2.8.6 Near-Field and Vadose Zone Flow and Transport Data and Model Uncertainty

DOE considers uncertainty in the near-field flow field through the assignment of velocity profiles for six separate alternative configurations (see Section 4.2.18, Table 4-11 for a description of the alternative configurations) and three separate tank types (see Figures 5.6-31 through 5.6-36 in DOE's Rev. 1 PA, SRS-REG-2007-00002, Rev. 1). The eighteen velocity profiles are abstracted from the two-dimensional PORFLOW near-field model for use in the GoldSim probabilistic model for individual configurations and tank types. The same velocity profile is used for each configuration or tank type bin. Thus, uncertainty in flow through the system is evaluated in a limited sense. Nonetheless, the differences in velocity profiles between alternative configurations and tank types are significant (range from around 13 cm/yr for Configuration E [water table rise above the bottom of the tanks] and Type IV tanks to around 37 cm/yr (14.6 in/yr) for Configuration F [soil cover] and Type I tanks; or a range of about a factor of three), although maximum velocities for many configurations and tank types fall somewhere in the middle at around 25 cm/yr (9.84 in/yr).

Uncertainty in the nature and characteristics of flow through the grouted tanks (e.g., flow through fractures versus through the grout matrix) might be expected to have a more significant effect on waste release. DOE did attempt to consider fast flow or by-passing pathways through the system in Configurations C and D. However, in the end, flow through the system is

governed by the variation in hydraulic properties (or lack thereof) between the tank grout, which is assumed to be significantly degraded at the time of steel liner failure, and the properties of the fast flow pathway, which is modeled as a high conductivity column of cells in the PORFLOW model, with significant flows occurring through both. The implication of this result (i.e., significant flow through both the "fracture" and matrix) is the presumption on the part of DOE that the fast flow pathway, which inherently represents a by-passing pathway, does not, in fact, lead to accelerated releases of HRRs from the contaminated zone. Instead, the degraded tank grout is assumed to be in intimate contact with the contaminated zone and to provide significant Eh poising and buffering capacity to the contaminated zone, thereby mitigating the release of HRRs for long periods of time (tens of thousands of years) in DOE's PA.

In response to NRC's concerns expressed in RAI-PA-1 and CC-NF-10 (DOE, 2011), DOE performed additional analyses to study a potential conceptual model described as Condition 2 in DOE's PA (see page 264 of [SRS-REG-2007-00002, Rev. 1]). A Condition 2 scenario is one in which a fast flow path exists through the tank grout prior to significant (bulk) grout degradation. Therefore, flow through the system is dominated by fractures or cracks that might form in the system over time (e.g., shrinkage gaps, or cracks that might develop due to thermal or mechanical stresses imposed on the system during curing, or due to corrosion of steel components) and waste release is dominated by unconditioned flow through the system. As might be expected, Case G is rather risk-significant (see Table 4-3) and is, therefore, discussed in more detail in the evaluation Section 4.2.9 and uncertainty analysis Sections 4.2.18 and 4.2.19 below.

Another PORFLOW modeling uncertainty is related to the representation and simulation of fracture flow (e.g., flow along the gap between the steel liner and tank grout, along steel components such as cooling coils, or through cracks in the grout monolith). Cases C, D, E, and G include fast pathways that are modeled as discrete material layers in PORFLOW with the hydraulic properties (e.g., porosity, hydraulic conductivity, and moisture characteristic curves [MCCs] of sand). The adequacy of fracture modeling treatment is discussed in Section 4.2.9.

Finally, flow rates through the system as modeled in PORFLOW influence the rate at which chemical transitions to higher solubility limiting phases in the contaminated zone occur. The longevity of chemical-barrier performance within Type IV tanks is greater than Type I or III/IIIA tanks due to a combination of (1) shedding of water around the Type IV tanks due to a domed roof, (2) the additional volume of grout in the roofs of Type IV tanks, and (3) the lower hydraulic conductivity associated with the Type IV tank grout due to the absence of cooling coils for this tank type (see CC-NF-11 (Camper, 2010)). Given the rather significant differences in the times to transition to higher solubility limiting phases (see Figure 4—2) for risk-significant radionuclides such as Tc and Pu, the simulation of flow through (or around) the system including the factors listed above is important to the compliance demonstration.

In addition to uncertainty in the flow field, there is also uncertainty associated with flow and transport parameters such as K_ds (discussed above), diffusivities, hydraulic conductivities, and MCCs assigned to materials in DOE's near-field model that are a function of radionuclides (in the case of K_ds) and the nature and extent of grout degradation (in the case of K_ds, diffusivities and hydraulic conductivities). With respect to diffusion, NRC staff sought additional clarification regarding the significance of upwards diffusion in DOE's near-field PORFLOW® models in CC-NF-12 (DOE, 2011). DOE provided additional information regarding the impact of upward diffusion with respect to iodine--nearly 100 percent of the high solubility iodine is transported upward into the tank grout within 10 years in PORFLOW simulations (DOE, 2011).

4.2.9 NRC Evaluation of Release and Near-Field Transport

4.2.9.1 NRC Evaluation of Cementitious Materials Degradation

NRC staff reviewed DOE's assessment of cementitious material degradation as described in Section 4.2.3.2.3 of the Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) and in supporting analyses presented in WSRC-STI-2007-00607 and SRS-REG-2007-00027. Based on information in the literature on cement and concrete degradation, staff finds that DOE's assessment has appropriately considered the two important cementitious material degradation mechanisms. DOE identified carbonation as the dominant degradation mechanism for grout in Types I, III, and IIIA tanks because of enhanced corrosion of the steel cooling coils present in these tanks, whereas it identified leaching as the major degradation mechanism for grout in Tank IV tanks, which have no cooling coils. Carbonation-induced corrosion of reinforcement steel was identified as the dominant degradation mechanism for the concrete vaults in all tank types.

DOE degradation calculations for the base case (Configuration A) suggest relatively long lifetimes for the cementitious materials. For example, the concrete vaults in Type I, III/IIIA, and IV tanks do not start degrading until 1,300, 2,500, 2,400, and 400 years, respectively. These long lifetimes may be overly optimistic based on available information from existing concrete vaults at the SRS. Furthermore, DOE likely overestimated the initiation time of concrete vault degradation by carbonation and rebar corrosion. DOE degradation analysis assumes concrete degradation starts once the carbonation effect reached one-half the concrete thickness. Because steel reinforcements typically are placed within a few inches of the concrete surface, not at half the concrete thickness, rebar corrosion-induced concrete degradation likely would initiate sooner than assumed in DOE's degradation analysis. As stated in WSRC-STI-2007-00607, "…rebar corrosion results in degradation of the reinforced concrete structure (cracking and strength loss) when the depth of carbonation equals the thickness of the rebar cover."

The uncertainties in the initial condition and performance lifetime of the concrete vault contribute to uncertainties in the calculated steel liner failure times. Concrete vault degradation would allow the ingress of species that accelerate steel corrosion (e.g., water, oxygen, chloride, and carbon dioxide) and decrease the performance lifetime of the steel tank liners. Although the grouted annulus could be an additional barrier protecting the side walls of Type I, III and IIIA steel tanks, there is no annular grout that would protect the Type IV tank side walls nor the roofs of Type I, III, and IIIA tanks. Section 4.2.9.2 discusses NRC staff concerns about the effect of uncertainties in cementitious material degradation on steel liner performance.

4.2.9.2 NRC Evaluation of Steel Degradation

<u>NRC Evaluation of Carbon Steel Waste Tank Liner Failure</u>

NRC staff reviewed DOE's assessment of carbon steel tank liner and stainless steel transfer line failure, as described in Rev. 1 PA (SRS-REG-2007-00002, Rev. 1), Section 4.2.3.2.5, and in supporting analyses presented in WSRC-STI-2007-00061, Rev. 2, and WSRC-STI-2007-00460. Because chemical species that induce corrosion, such as water, chloride, oxygen, and carbon dioxide, need to be transported through the cementitious materials, staff finds that the uncertainty in carbon steel tank liner longevity is likely related primarily to the integrity and hydraulic properties of the cementitious materials in contact with the steel liner.

NRC staff notes that the cementitious material degradation results in WSRC-STI-2007-00607 are inconsistent with key assumptions in modeling steel liner failure times in WSRC-STI-2007-

00061, Rev. 2. Table 4-7 compares the concrete lifetime from Table 4.2-32 of (SRS-REG-2007-00002, Rev. 1) and the carbon steel tank liner failure time from Table 4.2-35 of (SRS-REG-2007-00002, Rev. 1) to the corrosion initiation times by carbonation and chloride diffusion that NRC staff calculated based on relevant equations in WSRC-STI-2007-00061, Rev. 2. The carbonation equation used in the steel liner corrosion analysis (WSRC-STI-2007-00061, Rev. 2) results in a much longer carbonation-induced initiation time compared to the equation used in the cementitious material degradation modeling (WSRC-STI-2007-00607); compare columns two and four in Table 4-7 below. Thus, the initiation time for carbonation induced corrosion is much longer than the chloride induced initiation time (compare columns four and five). As a result, the steel liner corrosion modeling (WSRC-STI-2007-00061, Rev. 2) indicates that except for Type IV tanks, steel liner corrosion is controlled by chloride induced corrosion, which results in much later times to steel liner failure given the much lower rates assumed for chloride-induced corrosion. DOE's response to RAI NF-3 (DOE, 2011) suggests that the base case cementitious material degradation modeling is overly conservative and that the approach used in the steel liner corrosion modeling is reasonable. The primary difference between the two modeling approaches is related to the assumed diffusion coefficient (1×10^{-4} cm^2/s in the cementitious material degradation modeling versus the 1×10^{-6} cm^2/s default value assumed in the steel liner corrosion modeling).

Table 4-7 Concrete Lifetime and Carbon Steel Tank Liner Failure Time and Corrosion Initiation Time by Carbonation and Chloride Diffusion

Tank Type and Configurations	Concrete full degradation in Table 4.2-32[1] (yr)	Carbon steel tank liner corrosion			
		Liner failure time from Table 4.2-35[1] (yr)	Corrosion initiation time assuming diffusion coefficient of 1×10^{-6} cm^2/s (yr)		Steel tank corrosion initiation mechanism[4]
			Carbonation[2]	Chloride diffusion[3]	
Type I A, B, F	2,600	12,747	98,000	4,156	Chloride diffusion
Type III/IIIA A, B, F	5,000/4,800	12,751	182,000	6,068	Chloride diffusion
Type IV A, B, F	800	3638	3,237	519	Chloride diffusion, then carbonation

[1] From SRS-REG-2007-00002

[2] Calculated based on Equation in Subramanian (2008, p. 23, corrosion modeling)

[3] Calculated based on Equation in Subramanian (2008, p. 25, corrosion modeling)

[4] Determined based on three cases of potential corrosion in Subramanian (2008, p. 50-58, corrosion modeling)

NRC staff is concerned that more pessimistic assumptions regarding the hydraulic performance of the concrete vaults over time would lead to faster failure times for the steel liners than assumed in DOE's PA. Visual assessment of the concrete vaults is limited by the design of the underground waste tanks; however, the limited information that is available on the condition of the concrete vaults may not be consistent with the base case assumptions in the steel liner

corrosion modeling or the assignment of probabilities to the various configurations within the PA with respect to steel liner failure. Observations of cracking and groundwater in-leakage (WSRC-TR-93-761, WSRC-STI-2009-00352, SRR-STI-2010-00283, DPSPU-82-11-10, DOE/SRS-WD-2010-001, SRNS-STI-2008-00096, and SRS-REG-2007-00002, Rev. 1 indicate a limited ability of the vault concrete to act as a fully effective and uniform barrier to the transport of steel corrosion accelerating species (e.g., water, oxygen, chloride, carbon dioxide) over the lifetime of the steel liners. Additionally, observation of trenching, scarifying**,** and patching of concrete vault tops might further expedite degradation. Corrosion of steel components contained within the concrete vaults including transfer lines, reinforcement, pre-stressing bands, etc. are also reasonably expected to lead to cracking and degradation of the concrete vaults over time. Given the uncertainty in the initial conditions and long-term performance of the cementitious materials, it is not clear that the assumptions regarding cementitous material performance in DOE's corrosion modeling are supported or appropriate.

Furthermore, although DOE considers faster failure times for the steel liner in Configurations C, D, and E, DOE characterizes these higher risk configurations as low probability or unlikely events. Thus, faster failure times for the steel liner are either diluted in the overall probabilistic assessment or are dismissed when representing results of alternative configurations individually by DOE. With respect to modeling steel liner failure as a discrete event, NRC staff agrees that radionuclide release through partial failure of the steel liner (or use of a patch model) would likely lead to lower peak doses. However, radionuclide releases would occur earlier in time, within the period of compliance, which is not currently possible for 18 of 22 tanks at FTF. Given the fact that the peak doses in DOE's PA are predicted to be an order of magnitude or higher than the performance objectives at later times, earlier or partial steel liner failure may present a scenario with greater compliance risk than the current basecase scenario where steel liner failure is expected to occur discretely later in time and beyond the period of compliance for most FTF tanks. Only a small fraction of the releases predicted to occur in the base case configuration beyond the period of performance need to occur in an alternative, more realistic scenario within the period of performance to exceed the performance objectives.

In summary, a diffusion coefficient of 1×10^{-6} cm^2/s is selected as a conservative, but realistic value for intact concrete. However, it is not clear to NRC staff that this value sufficiently and adequately accounts for the observed or predicted degradation of the concrete vaults or the presence of joints and gaps within the concrete vaults or between the steel liner and the concrete vaults that might lead to by-passing pathways through the grouted vault system and more aggressive service conditions than assumed in DOE's modeling. Based on site observations and uncertainty in vault performance over thousands to tens of thousands of years simulated in the steel liner corrosion modeling, it is not clear that the modeling treatment of the steel liner—failing discretely and later in time beyond the period of compliance for most FTF tanks—is adequately supported. Pending results of solubility or leaching experiments recommended in Section 4.2.9.3 NRC staff may also recommend additional experiments or alternative modeling treatment of the steel liner to strengthen the compliance demonstration.

NRC Evaluation of Stainless Steel Transfer Line Failure

Based on literature information on stainless steel degradation in underground environments, especially in contact with soil, NRC staff finds that the uncertainty in stainless steel transfer line longevity is related primarily to general corrosion and pitting corrosion induced by chloride in the presence of oxygen. Staff finds the stainless steel general corrosion rate of 0.04 mil/yr and the pitting corrosion rate of 1 mil/yr are reasonable as bounding values for estimating the failure

times. Moreover, staff finds appropriate DOE's assessment that pitting corrosion is the controlling mechanism for transfer line failure.

4.2.9.3 NRC Evaluation of Source Term Release

NRC staff reviewed DOE's analysis of water chemistry contacting and leaching contaminants from the CZ, as described in Section 4.2.2.1 of SRS-REG-2007-00002, Rev. 1 and in supporting analyses presented in WSRC-STI-2007-00544. DOE's analysis of the evolution of chemical conditions (i.e., pH) in the grout and the longevity of reducing conditions important to retention of redox-sensitive radionuclides in the waste tanks is based on modeling using the Geochemist's Workbench® geochemical code. NRC staff notes that there is little experimental support for the Geochemist's Workbench® modeling results. NRC staff's concerns extend, not only to the long-term evolution of chemical conditions (i.e., pH and E_h), but also to the solubilities of the various radionuclides in the CZ.

NRC Evaluation of pH

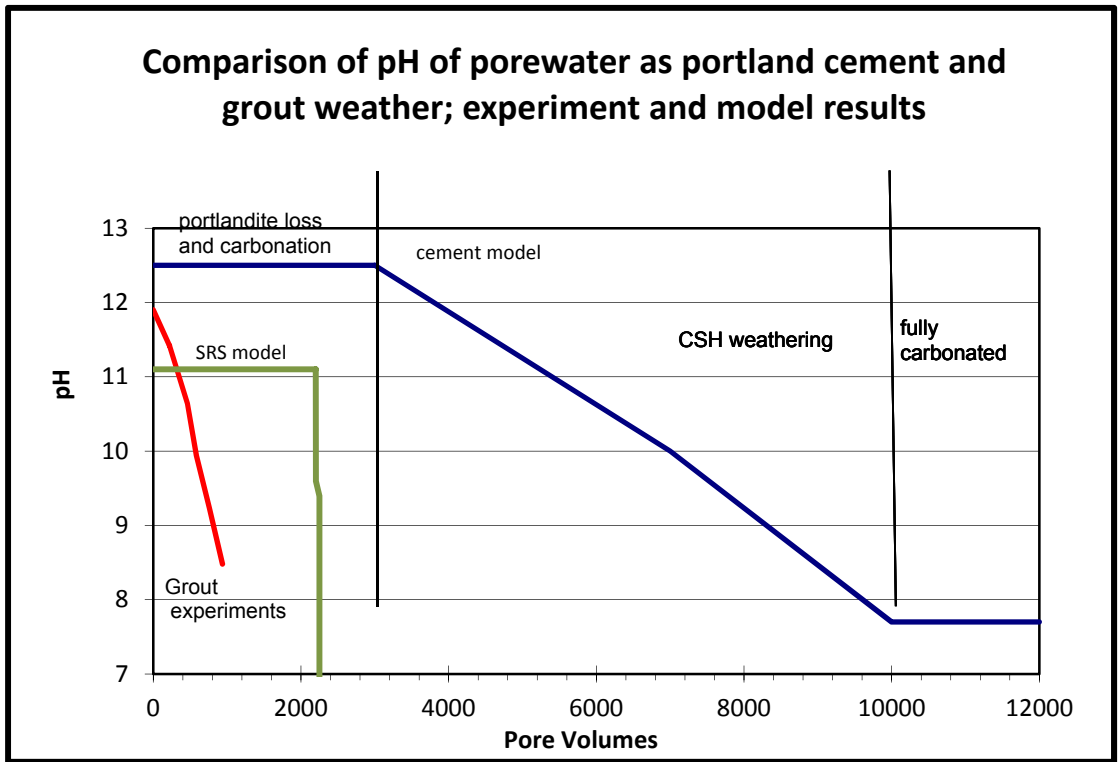
First, the normative mineralogy of the hydrated grout assumed for the calculation is different from that typically observed for hydrated portland cements. Because the mineralogy used in modeling grout degradation determines the pH evolution of grout pore water, which in turn affects the calculated radionuclide solubility, using an incorrect mineralogy in the model could lead to non-conservative solubilities and releases of radionuclides from the CZ. DOE conducted additional analyses in response to RAI-NF-7 (DOE, 2011). Geochemist's Workbench® calculations were performed using two alternate normative mineralogies to evaluate the effect on pH. The resulting pHs are slightly different for the different mineralogies, varying in the range 10.6 to 11.2 after 5,000 pore volumes of infiltrate. Additional DOE solubility calculations indicate that the solubility of most radionuclides do not change significantly in the pH range of 10.5 to 12.4. The calculated solubility of some radionuclides decreases at the lower end of the pH range, and a few radionuclides increase in solubility at the lower pH values, although by no more than 1.5 orders of magnitude. Based on the additional information, DOE thinks the grout mineralogy assumed in its original analysis is reasonable.

The pH of the grout system is a major controlling factor not only on solubility of radionuclides but on the formation of different dissolved species with potentially different mobilities in the environment. Therefore it is important to understand the long-term behavior of the tank grout with respect to evolution of pH. Figure 4-5 shows three graphs of pH versus pore volumes of water; two are modeling calculations and one curve is experimentally determined. The SRS model is shown as redrawn from Denham (2010). Also shown are data from column experiments from Fuhrmann and Gillow (2009) in which tank backfill grouts were analyzed. These experiments with tank backfill grouts, for compositions similar to those used by SRS, indicate that pH will drop almost linearly for grouts that contain low percentages of portland cement. The cement model is redrawn from Berner (1992) which is for portland cement itself. In grouts, the very small quantity of portland cement provides little portlandite to maintain the high pH. Moreover, it is suspected that the large percentage of fly ash in the formulation consumes substantial quantities of portlandite during hydration and early reactions. As a result the pH profile is less of a step function than a relatively steep, almost linear downward trend. A semi-quantitative assessment of the effect of fly ash on the pH buffering capacity of Portland cement is provided in Section 8.5.1 of Pabalan, et al. (2009).

NRC staff is not convinced that a step function as used in the PA is appropriate. The experimental column data in Figure 4-5 indicates that rubblized reducing grout attains a pH of

around 8.4 in just less than 1000 pore volumes. As a result, it is extensively carbonated. While the water used was a bicarbonate groundwater (synthesized), NRC staff think that the essential trend would be similar using SRS water. The difference in behavior suggests that the basic assumptions of solid phases within the grout are erroneous. It is unclear what the chemical effects will be on radionuclide dissolution and their subsequent retention/mobilization. However, the difference in observed versus modeled pH evolution suggests that modeled redox behavior may be different as well. It will be important to provide experimental support of modeling results for a parameter as important as pH.

Figure 4-5 Evolution of the pH of Tank Backfill Grout, Comparing Measured Column Data to Model Results



NRC Evaluation of Reduction-Oxidation Potential

Second, NRC staff recognizes that there are substantial questions associated with the measurement, modeling and long-term estimates of E_h and reducing capacity in cementitious systems. For example, the calculated pore water E_h in slag-bearing grout, derived from Geochemist's Workbench® analyses that assumes equilibrium with pyrrhotite or pyrite, is significantly lower (i.e., more reducing) than E_h values reported in the literature on fluids reacted with blast furnace slag and slag–cement mixtures. Higher E_h values than those assumed in the PA could result in higher solubilities, lower K_ds, and higher releases of redox sensitive radioelements such as technetium, plutonium, and neptunium. DOE provided additional information from the literature in response to RAI-NF-1 (DOE, 2011) that indicates E_h measured in systems involving sulfur redox couples almost always is more positive (more oxidizing) than E_h calculated based on analyzed concentrations of individual sulfur species. In fact, DOE calculated the E_h of pore fluid in contact with a slag–cement blend based on the sulfur species concentrations and pH that Angus and Glasser (1985) reported. The calculated E_h was –350

mV, which is more negative than the –230 mV Angus and Glasser (1985) measured. DOE also calculated the solubilities of technetium, plutonium, and neptunium over a range of E_h values. The calculated plutonium and neptunium solubilities (1.7×10^{-9} mol/L and 1.6×10^{-9} mol/L, respectively) are invariant over the E_h range from the stability of water to –200 mV. The calculated technetium solubility (2.8×10^{-8} mol/L) is invariant in the E_h range of –388 to –298 mV, but increases sharply at higher E_h. NRC staff is concerned that relatively small shifts in E_h, will result in remobilization of Tc-99.

Third, equilibrium modeling assumes all the reducing components in the grout are available for reaction with infiltrating water and is very much affected by assumptions of fracturing, water ingress, its oxygen content, and knowledge of the solids controlling redox capacity. The reducing capacity of the grout used in the model is based on laboratory measurements using slag samples that were finely ground to increase the reactive surface area. Other measurements (Lee and Batchelor, 2003) suggest that the reduction capacity may be substantially less, although the methods are open to question. DOE's estimates of the durability of reducing conditions rely heavily on equilibrium calculations that ignore potential kinetic limitations of dissolution/precipitation and redox reactions. This problem could appear in two distinct forms. In the first, in fractured grout systems, the flow rate may be fast relative to chemical reactions. As a result interactions between infiltrating water and reductants in the grout matrix will be diffusion-limited (Painter and Pabalan, 2009) or limited by depletion of buffering phases along the fracture–matrix interface (Harris, et al., 1997). These limitations could be significant particularly for technetium, which is an important contributor to dose and, as DOE solubility calculations indicate, becomes very soluble at E_h greater than –298 mV. In the second, at high pH, oxidation is associated with rapid precipitation of Fe oxyhydroxides on the pyrite (or pyrrhotite) surface, armoring the surface and protecting the pyrite from further oxidation. As a result it may be that the availability of reducing minerals for complete reaction is quite limited. This is especially so in static conditions (Pierez-Lopez et al, 2005). In both cases, the use of an equilibrium based approach to E_h control is very likely not appropriate.

Finally, NRC staff is concerned that, in actual field conditions, only a fraction of the reducing component will be accessible for reaction with the infiltrate, particularly if flow occurs through the fractures, and the reactive surface area and reducing capacity of grout emplaced in the field are likely to be much smaller than that of finely ground laboratory samples. Thus, the longevity of reducing chemical conditions and the retention of redox-sensitive elements such as technetium, plutonium, and neptunium, possibly are overestimated in the PA. DOE's response to RAI-NF-1 (DOE, 2011) acknowledges that some grout components may remain unavailable for reaction. However, DOE considered that the uncertainty in grout reactive surface area and reducing capacity are adequately considered in the PA sensitivity cases through one of two methods: (i) the redox transition times were varied by ±30 percent; or (ii) in extreme cases, the pore water E_h is assumed to be unaffected by the slag and the solubility limits associated with oxidizing conditions are applied. DOE also states that iron, which could be present in the grout as magnetite or hematite, could contribute to the grout reducing capacity but was not included in the base case calculations. DOE presented additional model results that indicated the presence of hematite or magnetite in the grout degradation calculations would result in redox transition times later than assumed in the base case calculations. Nonetheless, considering all of the factors mentioned above, NRC staff is not convinced that the ±30 percent in the chemical transition times adequately bounds the uncertainty in the first chemical transition time. While DOE barrier analysis and one alternative Configuration (Case G) study the impact of faster chemical transition times, these analyses are not part of the base case analysis and the alternative Configuration that results in doses exceeding the performance objective is screened

as highly unlikely. Detailed experimental studies would serve to increase confidence in the modeling approach and results.

NRC Evaluation of Radionuclide Solubility

NRC staff reviewed DOE's ISCM abstraction for radionuclide release, as described in Section 4.2.2 of (SRS-REG-2007-00002, Rev. 1) and supporting documents, most notably WSRC-STI-2007-00544 Rev. 2, RAI responses in SRR-CWDA-2011-00054 Rev. 0, and RAI responses in (DOE, 2011). In the ISCM, DOE assumes that the CZ is located immediately above the tank bottom, that the chemistry of the tank grout and the CZ are intimately related, but that solubility controlled release from the tanks is not necessarily dependent on transport of radiological constituents through the tank grout overlying the CZ as the radionuclide inventory is not initially assumed to be bound in the tank grout. NRC staff agrees that the adoption of a concentration limit, such as a solubility limit, to define the aqueous concentration of a contaminant is a reasonable approach to release modeling.

For several key radionuclides, the pure phase solubility limit models seem reasonable although identification of some of the controlling compounds appears problematic. For U, the becquerelite solubility of 3.4×10^{-5} M for Oxidizing Region III is at the high end of the range identified in an independent evaluation of solubility limits in cementitious systems (Pabalan, et al., 2009). The Np solubilities ranging from 2×10^{-9} M under reducing conditions to 1.0×10^{-4} M under oxidizing conditions are consistent with Pabalan, et al., (2009). The Pu solubility limit of 6×10^{-5} M for Oxidizing Region III appears to be conservative (high) for cementitious systems, and the lack of solubility control for Tc for Oxidizing Region III is appropriate (Pabalan, et al., 2009). As noted in Section 4.2.8.3, the Fe coprecipitation model yields concentrations under certain conditions for Tc, U, and Pu that are very low—even under oxidizing conditions for Region II. These low values strongly inhibit release of these radioelements until Oxidizing Region III is reached, which is usually well beyond 20,000 years in the models. Because Tc and Pu are strong dose contributors in models after they are released from the tanks under Oxidizing Region III conditions, the technical basis for the very low concentrations for the Fe coprecipitation model should be well supported. Aside from the low concentration values themselves, the most unexpected outcomes of this model are (i) the decrease in concentrations of these redox-sensitive elements when transitioning from Reducing to Oxidizing Region II and (ii) the very low Tc concentration under oxidizing conditions (Oxidizing Region II).

The Fe coprecipitation model assumes (i) a ratio between the radioelement and Fe in the tank residue, (ii) that the ratio in solution is the same, and (iii) that the Fe solution concentration is controlled by magnetite (reducing) or hematite (oxidizing) solubility. In a series of RAIs—NF-22 and NF-23 in SRR-CWDA-2009-00054 Rev. 0, and RAI-NF-8 and RAI-NF-9 in (DOE, 2011)—NRC staff requested further information supporting the model and the concentration limits DOE developed from it. NRC staff focuses on Tc and Pu, because these are the radioelements for which Fe coprecipitation has the most important effects on delaying release and suppressing modeled doses. DOE's PA model assumes essentially all Tc and Pu in tank residuals will remain bound in residual Fe phases until conditions reach Oxidizing Region III. If DOE adopts the Fe coprecipitation model for Np in future assessments, similar uncertainties will need to be addressed.

Given the observed changes of pH over about 1,000 pore volumes (Figure 4—5) in grouts to be used for tank backfill systems, it becomes important to understand the solubility of key radionuclides as pH changes. In turn, if geochemical modeling is used to calculate solubilities, it is necessary to know the solid phases controlling their solubility because the solubilities of

different compounds can vary tremendously as pH changes. In Section 4.2.2.1.2 (Table 4.2-10) and Section 4.2.2.3.1 of SRS-REG-2007-00002, Tables 4.2-13 and 4.2-14 indicate the probability distributions and solution concentrations of various compounds assumed to control solubility of key radionuclides under reducing and oxidizing conditions, respectively. For Tc, U and Pu, 50 percent of the probabilities assume Fe co-precipitation. While this assignment is based on reasonable assumptions made from the literature, it is not based on site-specific observations. In fact, evidence suggests that other compounds may be important and could control solubilities. As shown in Poirier et al (2010), Tc in one tank seemed to follow Fe removal, suggesting at least some co-precipitation. In another tank, even though 73 percent of the Fe was removed, very little Tc was mobilized. The implication is that radionuclide behavior with respect to incorporation into various compounds and their solubilities may vary from one tank to another.

In another example, the 2010 paper by King *et al* gives concentrations of U and Pu in what appear to be solubility controlled solutions (both from under and above saturation). Five different actual supernate samples were analyzed and then spiked with U or Pu to drive precipitation. Observed ranges of concentrations that represent solubility limits are: 4×10^{-5} M to 2.5×10^{-4} M of U and around 8×10^{-6} M of Pu. The Pu values are higher than any of those given in Tables 4.2-13 and 4.2-14 in SRS-REG-2007-00002. For U, the observed concentrations are higher than any in the reduced condition given in Table 4.2-13 and higher than most in the oxidized condition (Table 4.2-14). The authors identify clarkeite as the likely U solubility controlling mineral for 4 out of 5 samples. The relatively high solubility of this mineral will control concentrations while it persists in the solid phase. DOE presents a number of images and spectra from SEM/EDS from before and after oxalic acid cleaning of tank waste in WSRC-STI-2007-00192. Uranium is observed in the unwashed samples, often with Fe and Al. Although a large fraction of U is dissolved by the acid, samples from filtrate after oxalic acid cleaning clearly show discrete U compounds present, apparently oxides or hydroxides. The fraction of these compounds to total U is unknown. It is unknown if Pu is co-precipitated with Fe since the method is not sensitive enough to allow this determination. It is apparent from this report that the waste, as one would expect, is very heterogeneous and radionuclides (based on U) are retained in a number of compounds.

The PA uses the assumed co-precipitation of uranium, plutonium, and technetium with low solubility iron-bearing minerals (e.g., hematite, goethite, and magnetite) which results in very low predicted release rates. To provide support for this risk-significant barrier, characterization and leaching tests on residual tank waste are critical. As discussed in DOE's response to RAI-NF-8, each tank is to be sampled for determination of its final inventory. It was noted that "analysis may be performed to further understanding on the waste release assumptions regarding iron co-precipitation and solubility controlling phases". NRC staff strongly concurs with this statement.

NRC staff thinks that, if the solubility modeling approach is used, solid phase analysis of the residual waste is needed to provide information on the mineralogical composition and the elemental associations of the radionuclides to the various minerals. Methods that may be of use include X-ray Diffraction Mineralogy, Raman Spectroscopy, and Scanning Electron Microscopy with elemental analysis capabilities. However, the possibility of adequately determining the compounds that contain the various key radionuclides is doubtful. This is especially so if co-precipitation with Fe compounds does in fact control solubilities because radionuclide concentrations in this material, based on the molar ratios given in Table 4.2-17 of SRS-REG-2007-00002, Rev. 1, would be below detection limits (with the possible exception of U) of most methods for solids analysis. Characterization studies have shown that on the micron

scale the tank waste solids are heterogeneous and that radionuclides are simultaneously retained in a variety of compounds. Perhaps more importantly, washing the tanks with oxalic acid, will remove some large percentages of U (Poirier et al, 2010) but other radionuclides will remain in the sludge. It is conceivable that some may be reacted with oxalic acid to produce new, possibly low solubility compounds, not in the original waste (e.g., Pu oxalate). The very low solubility radionuclide compounds or co-precipitated radionuclides will not control solubility until other higher solubility compounds have become depleted. NRC staff is concerned about the use of very low solubility compounds in models without adequate support that no other, higher solubility, compounds are present. Staff believes that if chemical agents are used to try to remove radionuclides from the tank residues, then a detailed understanding of the effects that these reagents have on the solids remaining and the speciation of leachate should be attained.

Given the difficulty in quantifying the solid phases that contain radionuclides, NRC staff has concluded that it will be important to define the solubility of samples of the waste experimentally. Leaching tests should be conducted on residual waste samples to verify the release of radionuclides under conditions relevant to the contaminated zone throughout the compliance period. The purpose of these tests is to assess solubilities of important constituents, so static tests of sufficient volume to allow for enough sample intervals to approach constant concentrations are appropriate. The tests may be based on Method B in ASTM C-1285 or a similar static method. Sampling intervals can be informed by results in Hobbs (1999) and King (2010), but the tests should continue until constant concentrations of radionuclides are attained. It will be important to conduct the solubility tests under two end-member conditions: site-specific groundwater and chemical conditions that are influenced by contact with grout (e.g., elevated pH, etc). The static tests will indicate maximum solution concentrations. Other leach tests as described in Krupka et al (2006) and Cantrell (2006) will be needed to determine concentrations of important elements in solution after depletion of higher solubility compounds has taken place. These tests may take the form of a series of sequential batch tests to assess releases of more and less soluble fractions. For some elements, these compounds will control long-term concentrations in solution. Leachate analysis should include all elements/species of interest to aid in understanding the behavior of the system, including radionuclides of interest, pH, major stable elements such as Ca, Fe, Al, Si and major anions. This information is needed as input to geochemical modeling of tank leachate chemistry. Test results should be compared to calculated releases determined by modeling (e.g. Tables 4, 5, and 11 in Denham, 2010).

The interaction of infiltrating water with the various engineered barriers and their respective environmental conditions may lead to the complexation of radionuclides as they dissolve and are released from the contaminated zone. Hobbs (1999) also recommends studying the effects of organic complexing agents that might be present in the waste. Because speciation of dissolved radionuclides controls their mobility in soils, evaluation of these complexes may be important if the potential exists for these species to migrate through the engineered and natural systems. Depending on the experimental setup, the leachates from the residual waste samples may be able to serve as the precursor solutions for K_ds studies with various SRS soils. Verification of partitioning of key radionuclides onto natural sediment at the SRS site under realistic conditions would further reduce uncertainty in the PA predictions.

Variability in residual waste composition from tank to tank as well as within each tank should be evaluated and lessons learned from these studies could inform future sampling efforts (e.g., number and location of samples). For probabilistic analyses, DOE developed estimates of the likelihood of different solubility-limiting solid phases (PA Section 4.2.2.3.1). In a response to an NRC request to better explain how the probability estimates were derived, DOE did not explicitly address how the professional judgments were developed (RAI-NF-11 in (DOE, 2011)). If

stochastic analyses using these probability estimates continue to be used to support the PA, DOE should better justify the values.

In summary, NRC staff recommends careful characterization of tank residues to assess the potential long-term leachability of the waste including analysis of key radionuclide association with solid phases comprising the waste residue and leach tests. Ideally, characterization of tank residues following waste retrieval operations will be performed to ensure representativeness of experimental results. NRC staff also recommends experiments to study the pH and eH evolution of grouts proposed for FTF facility closure.

4.2.9.4 NRC Evaluation of Near-Field Flow and Transport Modeling

NRC Evaluation of Near-Field and Vadose Zone Transport Model Construction and Boundary Conditions

NRC staff reviewed DOE's near-field model constructed with the PORFLOW modeling code and generally found the near-field model to be sufficient for its intended use. However, several issues were identified during the review related to modeling parameterization, integration, and potential excessive dispersion as discussed below.

NRC Evaluation of Near-Field and Vadose Zone Transport Material Properties

Despite the relatively short transport pathway in the concrete basemat, retardation of Np and Pu appears to have significant effect on dose. Basemat conditions are either Oxidized Region II or Oxidized Region III. Cementitious material K_ds values for these elements are, therefore, important. The adopted K_ds values for Np and Pu in the concrete basemat are subject to uncertainty related to the possibility of solubility control during sorption experiments (RAI-NF-7 in SRR-CWDA-2011-00054 Rev. 1; SRNL-STI-2009-00473). DOE RAI responses have not yet established that the high K_ds values they have adopted are reasonable for representing sorption rather than solubility; furthermore, literature syntheses used to justify selected values (e.g., Tables 17 and 18 in SRNL-STI-2009-00473) reveal a diversity of experimental results that include K_ds values significantly lower than the adopted values. Furthermore, as discussed in Section 4.2.8.4, some of the Pu and Np cementitious material K_ds values are likely to be increased significantly in future assessments. These factors combine to make it clear that cementitious material K_ds for Np and Pu deserve special attention, both in assuring the range of literature values is adequately considered and in designing and interpreting laboratory sorption experiments. In addition, DOE should continue to seek appropriate literature data or new experiments that reflect conditions in the aged, oxidized basemat (e.g., RAI-NF-12 in (DOE, 2011)).

As mentioned in Section 4.2.8.4, DOE applies uncertainty distributions developed for natural system K_ds to cementitious material K_ds. DOE explains in response to CC-NF-8 (SRR-CWDA-2011-00054 Rev. 1) that it is reasonable to assume that variability in sorption characteristics of a relatively homogenous material like concrete or grout would be lower than variability in natural sedimentary rocks. Therefore, in the view of DOE, adopting the sedimentary relative uncertainty terms for cementitious material K_ds probably overestimates uncertainty in the latter. NRC staff thinks DOE should further consider the appropriateness of using the sediment statistics to define cementitious material K_ds uncertainty (e.g., as more data become available). In addition, as discussed in Section 4.2.9.4, NRC staff thinks more analyses are needed to refine the statistical approach to constraining K_d uncertainty.

With respect to assignment of hydraulic properties to vadose zone soils, the natural and geoengineered material properties assumed for vadose zone hydrologic modeling are all based upon data obtained from E-Area and Z-Area (WSRC-STI-2006-00198). For this reason, it was the recommendation of (SRNL-ESB-2007-00008) that additional field data be acquired and analyzed from at least one location associated with Tanks 1–8, 17–20, and 33–34, and at least one location associated with Tanks 25–28 and 44–47 to support the material property estimates for the FTF vadose zone (see CC-FF-1 (Camper, 2010)). NRC staff concurs with the recommendation of (SRNL-ESB-2007-00008). The results of any additional confirmatory work would be evaluated during the monitoring period.

NRC staff evaluated the information on natural system K_ds values presented in DOE's Rev. 1 PA (Sections 4.2.3.2.2 and 4.2.3.2.6, Table 4.2-29, SRS-REG-2007-00002, Rev. 1), supporting documents, and DOE responses to NRC RAIs (SRR-CWDA-2009-00054 Rev. 0; (DOE, 2011)). This discussion applies to data for both the vadose and saturated zones, because the two models rely on the same two sets of values (sandy and clayey). The NRC evaluation relied, in part, on comparisons to an NRC-sponsored compilation of site-specific K_d measurements for SRS (Prikryl and Pickett, 2007). The overall DOE approach to establishing K_ds is appropriate—employing site-specific data when possible and relying on literature data when site-specific data are lacking or sparse. NRC staff finds, in general, that the values used for the deterministic analysis are reasonable, but notes in the following section where FTF- specific data collection would be helpful.

Sandy sediment K_d values for Pu(III/IV) and Pu(V/VI) from PA Table 4.2-29 of 300 and 16 mL/g, respectively-are consistent with the analysis of Prikryl and Pickett (2007). The same may be said of the Pu(III/IV) K_ds of 6,000 mL/g for clayey sediments. The Pu(V/VI) clayey sediment value of 5,000 mL/g, however, appears too high. Prikryl and Pickett (2007) referred to a number of studies that yielded significantly lower values. As WSRC-TR-2006-00004 Rev. 0, Table 10, points out, initial Pu(V) is typically observed to be reduced to Pu(IV) on the sediment surface, particularly in clay-bearing sediment experiments. This observation appears to underlie that report's decision to recommend such a high clayey value for Pu(V/VI). This line of reasoning, however, suggests that the value being recommended is actually a K_d for Pu(III/IV). NRC staff points out—while recognizing the experimental challenges—that an attempt should be made to estimate the K_d of Pu(V/VI) irrespective of the reduction effect. While DOE's response to CC-FF-9 is helpful (SRS-REG-2007-00002, Rev. 1), NRC staff still questions DOE's approach of combining the Pu(III/IV) and Pu(V/VI) K_ds values according to an assumption that the Pu is composed of 90 percent of the former and 10 percent of the latter (SRR-CWDA-2011-00054 Rev. 1, CC-FF-9). These "hybrid" values do not capture the relatively fast transport characteristics of that smaller Pu(V/VI) fraction; the slow character of Pu(III/IV) dominates the hybrid K_d. NRC staff thinks a more accurate representation of the transport of multivalent Pu would be to treat the two species separately. In the RAI response, DOE stated that they plan future work related to Pu sorption in the natural system. NRC staff will follow this technical issue and DOE's efforts to address this issue during the monitoring period.

The radionuclide transport effects of the potentially alkaline plume leaving the bottom of a breached tank was the subject of an RAI (SRR-CWDA-2009-00054 Rev 0, FF-2). DOE stated in the response, and reiterated in Section 8.2 of the Rev. 1 PA, that they will continue to try to better understand the "influence of cementitious material on vadose zone geochemistry" (DOE's PA Rev. 1 (SRS-REG-2007-00002, Rev. 1), page 814).

NRC Evaluation of Near-Field and Vadose Zone Transport Data and Model Uncertainty

With respect to transport properties, NRC staff is not confident that the assigned K_d values determined from a single borehole in E-Area are necessarily appropriate for use in F-Area models. The justification for the similarity is given in section 4.2.3.2.2 of the PA (SRS-REG-2007-00002, Rev. 1). Core logs from the two areas are compared in Figures 4.2-19 and 4.2-20 (SRS-REG-2007-00002, Rev. 1). While these show similarities, none of the data are explicitly related to chemical or mineralogical parameters that might infer similar K_d values. K_ds are sensitive to pH and to clay content as discussed in WSRC-TR-2006-00004 and subtle differences in the material that cannot be determined from core logs may be important. NRC recommends that a set of K_d values be determined for key radionuclides using site-specific materials from the downstream area of the FTF. These can then be used to make a correlation to those values obtained for materials from E-Area.

For the GoldSim stochastic analysis, lognormal uncertainty distributions for sorption coefficients (Section 5.6.3.4, SRS-REG-2007-00002, Rev. 1) are calculated on the basis of a statistical analysis of samples from a core collected in E-Area. In that analysis (WSRC-STI-2008-00285), K_ds was measured for radioisotopes of Cd, Ce, Co, Cs, Hg, Sr, and Y in 27 samples that represented the Upper Vadose, Lower Vadose, and Aquifer Zones. SRNL-STI-2009-00150 Rev. 1 recommended an Upper Vadose Zone range for clayey soils and a range for sandy soils that was an average of those for the Lower Vadose Zone and Aquifer. DOE's Rev. 1 PA (Section 5.6.3.4 of SRS-REG-2007-00002, Rev. 1) adopted minimum and maximum multipliers based on material type for the lognormal distributions and also calculated standard deviations based on multipliers of the geometric mean (PA Table 5.6-4). Neptunium was assigned specific minima and maxima, based on recent experimental results.

NRC staff is not convinced that DOE has adopted an appropriate approach to assigning uncertainty bounds to sandy and clayey sediment K_ds—and, by extension, cementitious materials—for the purposes of stochastic analyses (described in Section 4.2.8.4 of this TER). First of all, the multipliers to establish minima and maxima are not radionuclide-specific. In fact, the set of elements used in the statistical analysis— Cd, Ce, Co, Cs, Hg, Sr, and Y—does not include any of the risk-significant species in DOE's FTF PA results. Neptunium in an exception in that its bounds were assigned explicitly. As a result it is not clear if the assigned distributions (based on the elements listed above) in any way represent the radionuclide specific distributions. The complex and potentially variable behavior of the key elements such as Pu and Tc may not be captured by this analog approach, nor may variability related to groundwater chemistry. While the quality of the original analysis is not in question, the data was obtained from a single borehole in E-Area (WSRC-STI-2008-00285) and an argument was not made in later reports that the analysis was applicable to the transport pathways in the FTF PA (SRNL-STI-2009-00150, Rev. 1; PA Section 5.6.3.4 of SRS-REG-2007-00002, Rev. 1).

The modeled uncertainty ranges are relatively narrow, being no larger than ± 75 percent. The recommendation in SRNL-STI-2009-00150 Rev. 1 that the sandy sediment distribution range be narrower than the observed confidence level range for the Aquifer Zone, suggests that uncertainty in much of the saturated zone will be underestimated. Furthermore, it is not clear from the description of the algorithm (PA Section 5.6.3.4 and Table 5.6-4 of SRS-REG-2007-00002, Rev. 1) how the geometric standard deviations are calculated.

DOE noted (CC-NF-8 response in (DOE, 2011)) that Section 8.2 in DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) states that DOE will continue to refine their representations of uncertainty in future stochastic analyses. NRC staff concludes that the treatment of sediment K_d is in need of either stronger and clearer technical basis, or refinement to more accurately represent the potential variability in sorption behavior relevant to the FTF PA.

With respect to flow uncertainty, DOE modeled six separate configurations to evaluate the impact of flow on the modeling results. Flow uncertainty was not otherwise evaluated in DOE's PA as the flow velocities are hard-wired in the probabilistic model by tank type and configuration. While variability in flow rates represented in alternative configurations might be expected to have a significant effect on the compliance demonstration (peak doses within 10,000 years), other factors influencing the concentrations and dose at the point of compliance are more significant (e.g. having an order of magnitude or more impact on concentration and dose) such as steel liner failure times, chemical transition times, and basemat by-pass. Other parameters affecting concentrations and dose at the point of compliance, that are independent of configuration and tank type, include solubility limiting phases, and basemat and natural system K_ds that might also overshadow the impact of flow uncertainty on the modeling results over longer time periods. In fact, the differences in peak dose between alternative configurations seem to be less pronounced over longer time periods suggesting that steel liner failure and chemical transition times affecting timing of peak dose account for most of the variability in the results for alternative configurations over shorter periods of performance, 10,000 years, while other parameters (that are independent of configuration) impact the magnitude of peak dose over longer time periods. While certain anomalies with flow fields abstracted from PORFLOW exist that have not fully been explored by NRC staff (e.g., shedding of water from the roofs of Type IV tanks, potential flow impedance into the tanks associated with the selection of MCC data for use in the PA, and non-intuitive results associated with Configuration E [water table rise scenario] and Configuration F [soil cover scenario] doses), in general, NRC staff thinks that DOE has adequately evaluated flow model uncertainty in the Rev. 1 PA and has included reasonable flow velocities in its models that are near the maximum infiltration rates predicted to occur through the engineered cover of around 30 cm/yr (12 in/yr) over longer periods of time.

Perhaps, more importantly, NRC questioned DOE's conceptual model for the nature of flow through FTF tanks (fracture or by-pass versus bulk matrix flow). Case G is presented in RAI-PA-1 with results showing doses that could exceed the performance objective within the 10,000 year performance period in this alternative flow model case. The characteristics of Configuration G that lead to unacceptable results are the times to steel liner failure (500 years for Type I and III/IIIA tanks and 75 years for Type IV tanks) and chemical transition times (that result directly from the change in the conceptual model for flow through the tank system). While DOE also assumes that iron co-precipitation does not control radionuclide release, the primary affect of this assumption is that the releases of Pu-239 occurs earlier in the compliance period upon transition to oxidized Region II rather than following transition to oxidized Region III at a solubility limit approximately two orders of magnitude lower than the oxidized Region III solubility limit. Another difference between Configuration G and the base case Configuration A is the inclusion of a fast pathway through the basemat that allows partial by-pass of the attenuating properties of the basemat. Nevertheless, the change in conceptual model associated with flow through the tank system and the resultant faster times to chemical transition undoubtedly has the largest affect on the results. NRC staff thinks that DOE has adequately addressed flow model uncertainty through the consideration of the alternative Case G. However, based on the risk-significance of the conceptual model and waste release uncertainty in DOE's PA, NRC staff recommends that DOE provide additional support for the

conceptual model for fluid flow through the grouted tank and for solubility limiting phases and transition times. NRC will continue to evaluate these uncertainties during monitoring in its assessment of compliance of the FTF with the performance objectives in 10 CFR Part 61, Subpart C as tank farm closure progresses.

With respect to the use of MCCs for sand to model fast pathways, DOE's approach may not adequately reflect real-world fracture flow. Flow through intact matrices is fundamentally different than fracture flow and applying material properties of a continuous material (e.g., sand) to a discontinuous material (e.g., an annulus or a fracture) may under-represent flow. Representation of flow through fractures is a challenge due to the inherent variability and uncertainty in real-world flow behavior and the corresponding abstraction of that behavior into a model. If model results are sensitive to the range of variability and uncertainty in fracture flow, model support (e.g., fracture flow experiments coupled with field-scale observations) may be necessary.

Given the rather significant differences in the times to transition to higher solubility limiting phases (see Figure 4-2) for risk-significant radionuclides such as Tc and Pu, the simulation of flow through (or around) the system including the factors listed above may be important to the compliance demonstration. NRC staff will continue to track assumptions related to chemical transition times most notably, the conceptual model for fluid flow (matrix versus fracture), progress in grout degradation modeling, and modeling assumptions that might affect flow through the system (MCC data).

Finally, NRC notes a significant amount of dispersion particularly for highly mobile HRRs out of the contaminated zone in DOE's near-field model. In the case of Tc-99, transition to oxidized Region III and no solubility control leads to virtually instantaneous spreading of Tc-99 in the contaminated zone upwards into the tank grout against the direction of advection (note the contaminant mass is quickly transported back down again into the contaminated zone and basemat), as well as downwards into the basemat with the direction of advection. While flow out of the tank system is rapid upon chemical transition, the re-distribution of Tc-99 appears to be rather significant leading to significantly reduced concentrations along flow paths away from the contaminated zone.[8] NRC will continue to evaluate this phenomena during the monitoring period to assess the reasonableness of this modeling result.

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[8]NRC staff estimates that the concentration in the contaminated zone (CZ) (and similar concentration at the top of the basemat) is a factor of 66X times lower one year following chemical transition to Oxidized Region III than what it would be if all of the Tc-99 mass were in the CZ with most of the mass being transported back into (and out of) the CZ from advection in less than 2 years following chemical transition. The maximum concentration underneath the basemat is a factor of 8X less than the maximum concentration in the CZ following chemical transition and is a factor of 25X less upon entry into the saturated zone. Given the minimal amount of attenuation (low K_ds for concrete and soils), these results suggest a significant amount of dispersion in the near-field and far-field models. Differences in vertical thickness of grid elements between the CZ and adjacent cells; and between the near-field vadose zone and far-field saturated zone models may partially account for the rather significant decrease in Tc-99 concentrations along the flow path away from the CZ. Additionally, finer time discretization may be necessary for this very swift moving constituent upon chemical transition to no solubility control considering Courant number criteria.

4.2.9.5 Near-field Review Results and Recommendations

NRC staff performed a comprehensive review of DOE PA documentation and supporting references related to near-field modeling. Primary review results and recommendations are presented below.

NRC staff notes the following with respect to near-field modeling:

- Major degradation processes appear to be considered in DOE's cementitious material degradation modeling, although NRC notes large uncertainties associated with modeling degradation processes over the long time periods relied on for performance in DOE's PA.

- Major corrosion processes appear to be considered in the steel liner degradation modeling; however, DOE assumptions regarding the performance of cementitious materials that are relied on as a barrier to steel liner corrosion may be over-stated and the importance of aggressive corrosion processes under-stated in DOE's PA based on site-specific and other observations.

- Near-field modeling simplifications (e.g., assumed discrete failure of tank liners) may lead to overall higher peak dose but lower dose within the compliance period and, consequently, lower compliance risk than more realistic approaches to modeling liner failure.

- Basemat K_ds can be especially risk-significant for certain key radionuclides; additional experimental support could increase confidence in selected values given (1) the wide range of values found in the literature, (2) the potential mis-application of experimental data (sorption versus solubility control), and (3) to ensure K_ds representative of aged concrete are obtained.

- DOE's overall approach to establishing natural system K_ds is appropriate and the deterministic values are reasonable with a few exceptions noted in Section 4.2.9.4.

- Although DOE's overall approach to modeling waste release is reasonable, DOE assumptions regarding solubility limiting phases, solubility limits, and chemical transition times are risk-significant and have not been confirmed through waste characterization and experimentation.

- Additional characterization, experimentation, or modeling may reduce uncertainty in assignment of risk-significant parameters such as basemat and natural system K_ds.

- DOE's PA modeling results indicate potential excessive dispersion in the near-field and far-field models.

While the work performed by DOE in the area of near-field modeling is commendable given the limited amount of data available in the literature to support key modeling assumptions, characterization of DOE waste tank residuals and observation of short- and longer-term performance of key components critical to DOE's compliance demonstration (i.e., tank liners) is

lacking. NRC staff recognizes the difficulty in obtaining additional information to reduce technical uncertainties identified in NRC staff's review. However, NRC staff has identified one technical area where additional information can readily be obtained that appears to be especially risk-significant as described below. Other secondary recommendations discussed in Section 4.2.9 are of lesser risk-significance; recommendations for execution of secondary recommendations is contingent on the results of experiments recommended below.

NRC staff makes the following primary recommendation related to near-field modeling and support that it considers crucial to DOE's compliance demonstration (High Risk-Significance, Short Term (Tank 18) and Intermediate-Term (other tanks)):

NRC staff recommends DOE conduct waste release experiments:

1. To increase experimental support for key modeling assumptions about behavior of grout over time including evolution of pH and E_h (High Risk-Significance, Short to Intermediate Term);

2. To identify key radionuclide association with solid phases comprising the residue in representative tanks to support key modeling assumptions (Medium to High Risk-Significance, Intermediate Term); and

3. Leach tests on multiple samples from each tank. Tests should consist of:

    a) Static tests to determine constant concentrations of elements of concern under conditions of exposure to local ground water and to grout leachate (High Risk-Significance, Short and Intermediate Term); and

    b) Semi-dynamic leach tests to try to distinguish releases from high solubility compounds from those of low solubility compounds (Medium to High Risk-Significance, Intermediate Term).

These experiments would consider the effects of reagents (e.g., oxalic acid) used to remove radionuclides from the tank residue, including formation of new compounds that may alter leachability of radionuclides.

4.2.10 Hydrology and Far-Field Transport

4.2.10.1 Model Construction

DOE uses the GSA database of hydrostratigraphic and hydrogeologic data to define and parameterize three-dimensional groundwater flow and contaminant transport models for the FTF. This database includes picks for tops of hydrostratigraphic units, sediment core descriptions, water levels in the upper and lower zones of the UTR aquifer, water levels in the Gordon aquifer, and permeability data from laboratory tests, multiple and single well pump tests, and slug tests.

The UTR and Gordon aquifers will be impacted by radionuclides leaking from the FTF facility. Contamination is not expected to affect the deeper Crouch Branch aquifer, however, because of an upward flow gradient between the Crouch Branch and Gordon aquifers near UTR creek. Groundwater flow in the UTR aquifer is driven by recharge. The underlying Gordon aquifer is strongly influenced by its discharge to UTR creek. The Gordon aquifer is recharged by

downward leakage from the UTR aquifer above and upward leakage from the Crouch Branch aquifer below. Conceptually, recharge of the Gordon aquifer by the Crouch Branch aquifer is very small in comparison to recharge from the UTR aquifer and is thus neglected in modeling.

The primary focus of the regional GSA/PORFLOW saturated zone flow and transport model (see Figure 4-6) is seepline concentrations for the stream dose analysis. The interior areal resolution of this model is 61 × 61 m^2 (200 × 200 ft^2); peripheral grid cells at the margin of the model domain are larger. A maximum of 108 grid cells span the domain from east to west, and a maximum of 77 grid cells span the domain from north to south. Vertical resolution varies with hydrostratigraphic picks and topography. The UTR-UZ aquifer is represented by up to 10 cells in the vertical direction, and the vadose zone is considered to be part of this unit. The tan clay confining zone (TCCZ) is represented by 2 cells in the vertical direction, and is assumed to be laterally continuous with no pinchouts. The UTR-LZ aquifer is represented by 5 cells in the vertical direction. The underlying GCU and Gordon aquifer are each represented by 2 cells in the vertical direction. Thus, a maximum of 21 grid cells represent the GSA hydrogeology from ground surface to the base of the Gordon aquifer. The regional model domain comprises 102,294 cells. A fixed time step of one year is used for saturated zone flow and transport modeling (SRS-REG-2007-00002, Rev. 1).

The primary focus of the local FTF/PORFLOW saturated zone flow and transport model is well concentrations at 1 m (3 ft) and 100 m (330 ft) compliance points for the water supply well dose analyses (see Figure 4-7). The areal resolution of the local (FTF) PORFLOW model is 15 × 15 m^2 (50 × 50 ft^2). In the horizontal plane, each GSA/PORFLOW model grid cell is divided four ways in each coordinate direction into 16 FTF/PORFLOW transport model grid cells; vertical resolution, however, is preserved (DOE, 2011). This constitutes a 4×4×1 mesh refinement.

The velocity field for the local FTF/PORFLOW model is generated with a mass-conserving linear interpolation scheme directly from the regional GSA/PORFLOW velocity model; thus, the local FTF/PORFLOW model does not necessitate a separate flow model with its own boundary conditions and material properties. Within the lateral confines of the local FTF/PORFLOW model, the velocity field includes the complete vertical extent of the regional GSA/PORFLOW model.

Figure 4-6 Perspective View of Regional GSA/PORFLOW Flow and Transport Model Domain (Image Credit: WSRC-TR-2004-00106, Figure 2-3)

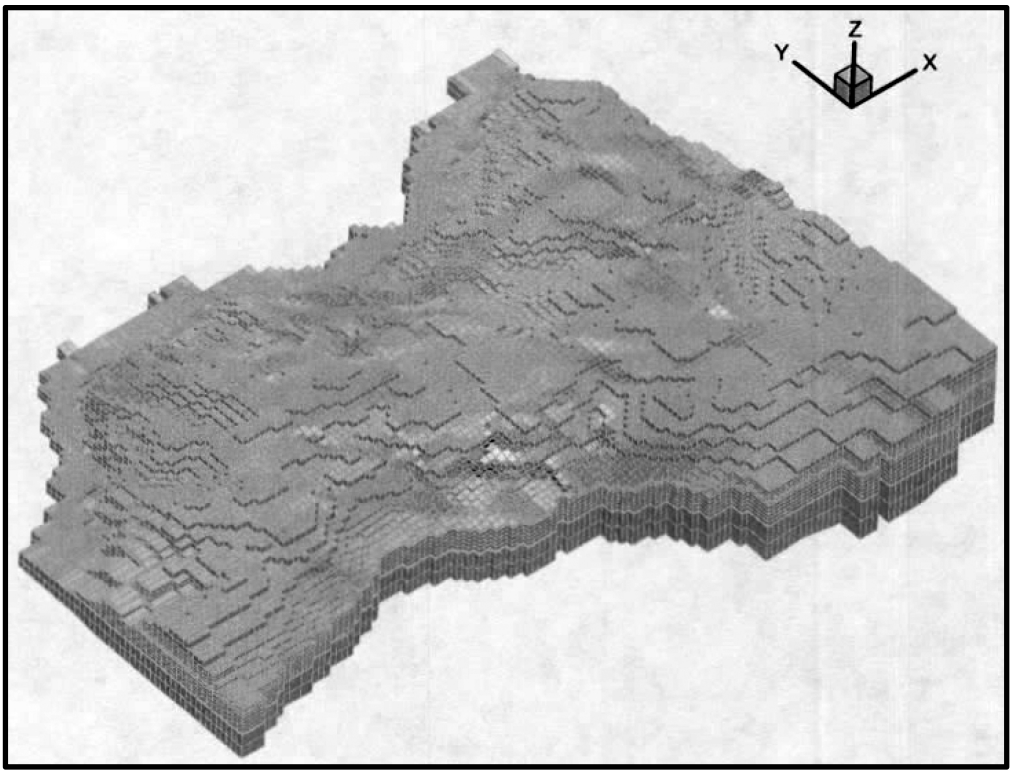
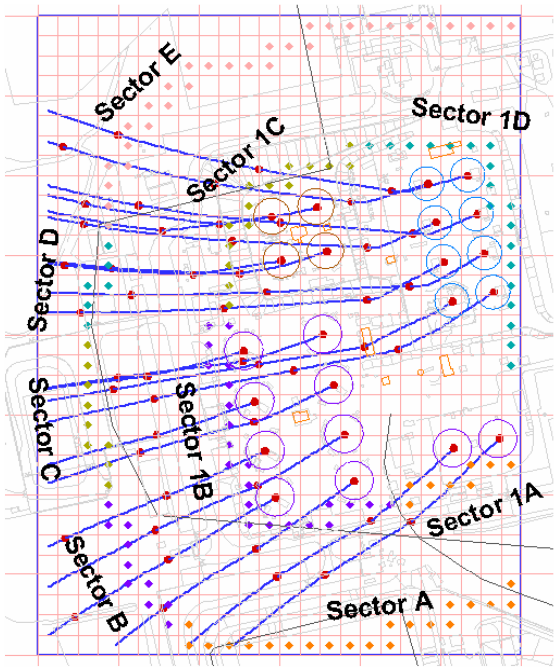


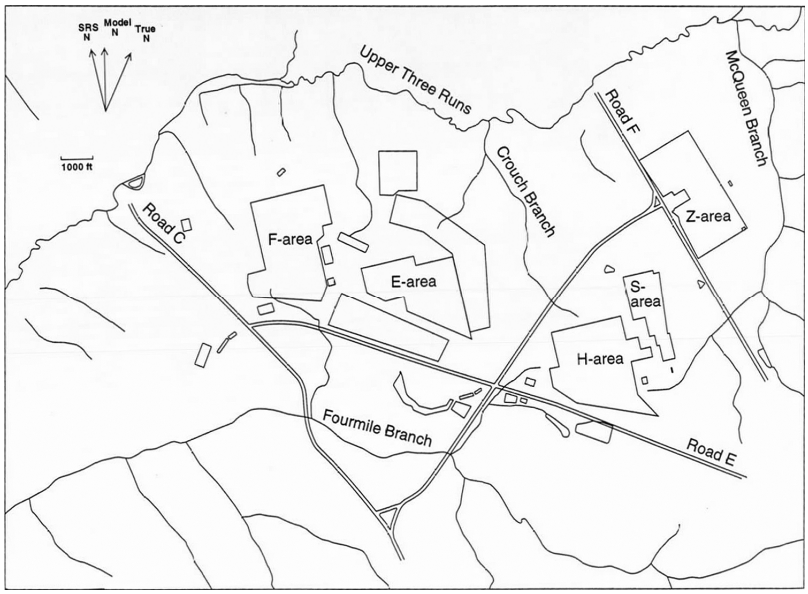
Figure 4-7 FTF Saturated Zone Model with 1 m (3 ft) and 100 m (330 ft) Compliance Evaluation Sectors. Blue Streamlines Represent Each Plume Centerline. Red Dots Represent Tank Centers and a Series of Ten-Year Time Markers. Colored Diamonds Mark Compliance Boundaries Grouped by Sector (Image Credit: SRR-REG-2007-00002, Rev. 1 Figure 5.2-5)



4.2.10.2 Boundary Conditions

The GSA/PORFLOW saturated zone model domain encompasses the GSA and its surface water discharge points (see Figure 4-8). Streams define three lateral domain boundaries: UTR forms the northern boundary; McQueen Branch forms the eastern boundary; and Four Mile Branch forms the southern boundary. Four Mile Branch and McQueen Branch provide natural, no-flow boundary conditions for the UTR aquifer. This aquifer unit is absent at the northern model boundary due to UTR stream incision. The western boundary is arbitrarily chosen where hydraulic head values from a contour map of measured water elevations are prescribed. The Gordon aquifer only discharges to UTR creek, so hydraulic head values from a contour map are prescribed over the west, south and east faces of the model, but UTR creek provides a natural no-flow boundary condition on the north face of the model. The grid cells that represent the base of the Gordon aquifer are prescribed a general head boundary condition; upwelling recharge from the Crouch Branch aquifer below is considered negligible. The upper surface of the PORFLOW saturated zone model is prescribed a spatially variable recharge boundary condition with an average recharge flux of 37.3 cm/yr (14.7 in/yr) (WSRC-TR-2004-00106, Figure 3-7) appropriate for non-capped GSA sediments); the lower magnitude water flux exiting the PORFLOW vadose zone model is not used.

Figure 4-8 Map View Context Image of the General Separations Area at the Savannah River Site (Image Credit: Figure 1 of WSRC-TR-96-0399, Vol. 2, Rev. 1)



Contaminant fluxes that exit the individual vadose zone models become an upper boundary condition for the local FTF/PORFLOW saturated zone transport model. The radionuclide contaminant fluxes are assigned to the FTF/PORFLOW model grid by uniform distribution to the water table grid cells with centroids lying within the footprint of the waste source. Fluxes from transfer lines are uniformly distributed over the FTF footprint.

4.2.10.3 Material Properties and Parameters

DOE indirectly used the GSA database of hydrologic data to parameterize the material properties for the regional GSA/PORFLOW model of the saturated zone. This database includes hydraulic conductivity data from laboratory tests, pump tests (multiple and single well tests), and slug tests. These data were used directly to parameterize a predecessor GSA/FACT model that was later replaced by the GSA/PORFLOW model (WSRC-TR-2004-00106).

The initial (uncalibrated) hydraulic conductivity assignments to GSA/FACT model grid cells were based upon a correlation between hydraulic conductivity and total mud fraction (both calcareous and siliciclastic), where mud is the summation of silt and clay fractions (see CC-FF-5 response figures in (DOE, 2011)). Apparently, known core lithologies (i.e., mud fraction) from discrete intervals were correlated with known field- or laboratory-measured hydraulic conductivities from the same intervals, although DOE does not clearly state this in its RAI response. Initial GSA/FACT hydraulic conductivity fields were subsequently modified during model calibration. Then, during the migration from GSA/FACT to GSA/PORFLOW, the initial FACT-calibrated horizontal hydraulic conductivities in the UTR-UZ and UTR-LZ were further increased by 25 and 35 percent (WSRC-TR-2004-00106).

Grid cells in the regional GSA/PORFLOW model having calibrated vertical hydraulic conductivity greater than 1.0×10^{-7} cm/s (3.3×10^{-9} ft/s) are defined as sandy, while those having calibrated hydraulic conductivity less than this are defined as clayey (WSRC-STI-2006-00198). The tan

clay is assigned a constant calibrated vertical hydraulic conductivity of 1.8 x 10^{-6} cm/s (5.9 x 10^{-8} ft/s) in F-Area, such that it functions as a relatively ineffective silty aquitard. In contrast, the GCU is assigned a uniform calibrated vertical conductivity of 3.5 x 10^{-8} cm/s (1.2 x10^{-9} ft/s), such that it functions as a significant clayey aquitard (WSRC-TR-96-0399). The calibrated Gordon aquifer unit, for which there is limited field data, was assigned a uniform horizontal hydraulic conductivity of 1.3 x 10^{-2} cm/s (4.4 x 10^{-4} ft/s) because the initial heterogeneous conductivity field was inconsistent with measured hydraulic heads. The selected value is said to be consistent with multiple well pumping test data and prior modeling (WSRC-TR-96-0399). The uniform calibrated hydraulic conductivity values for the GCU and aquifer were not modified during the migration from GSA/FACT to GSA/PORFLOW (WSRC-TR-2004-00106).

The geostatistical analyses of Evans (1995) revealed that the fluvial facies of the UTR-UZ are more heterogeneous than the shoreface marine facies of the UTR-LZ, consistent with the final calibrated hydraulic conductivities used by (SRR-REG-2007-00002, Rev. 1) (e.g., Figure 4-9 and Figure 4-10).

Figure 4-9 Calibrated Horizontal Hydraulic Conductivity Assignments to the Regional GSA/PORFLOW Model (Fourmile Branch is South and Upper Three Runs Creek is North of the F-Tank Farm)

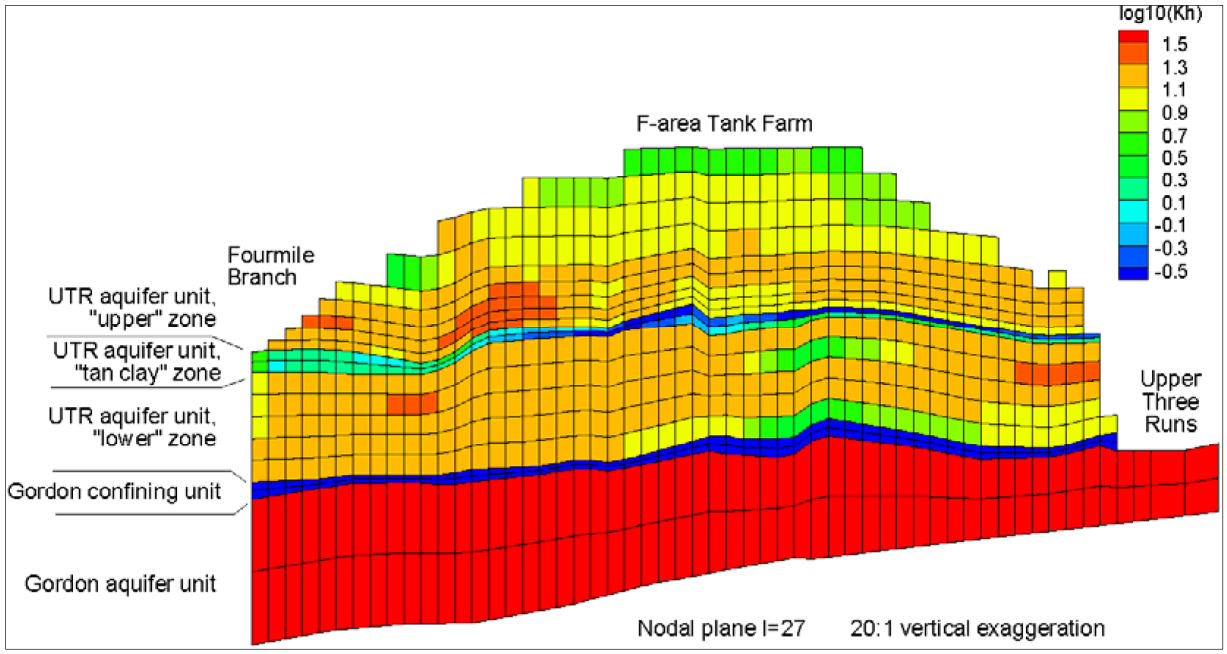
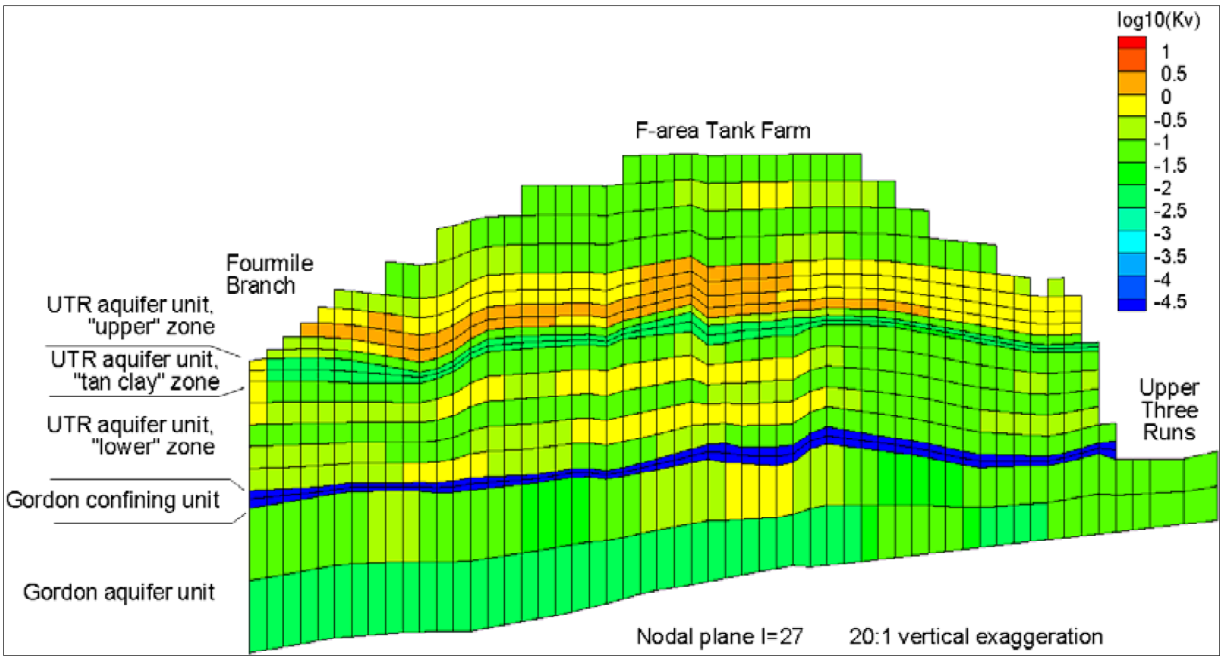


Figure 4-10 Calibrated Vertical Hydraulic Conductivity Assignments to the Regional GSA/PORFLOW Model(Fourmile Branch is South and Upper Three Runs Creek is North of the F-Tank Farm)



Effective diffusion coefficients (Table 4-8) are assigned to saturated zone sediments based upon whether they are defined as sandy or clayey, consistent with vadose zone modeling properties for E-Area (WSRC-STI-2006-00198). GSA/PORFLOW transport modeling assumes an effective porosity value (Table 4-8) for all aquifers and aquitards to account for dead-end pores that do not participate in radionuclide transport (WSRC-TR-96-0399; WSRC-STI-2006-00198). Because the UTR and Gordon aquifer material properties are similar to the lower vadose zone material properties, DOE converts average values for lower vadose zone bulk density (1.62 g/cc [0.94 oz/in^3]) and particle density (2.66 g/cc [1.54 oz/in^3]) to effective values for saturated zone bulk density (1.04 g/cc [0.6 oz/in^3]) and particle density (1.39 g/cc [0.8 oz/in^3]) using the average value (39 percent) and effective value (25 percent) of lower vadose zone porosity (WSRC-STI-2006-00198). DOE also assumes these same effective material property values for the TCCZ and GCU.

Given that the vadose zone is included within the regional GSA/PORFLOW model mesh, water retention characteristic curves are required. Pseudo-soil water retention characteristic curves, which exhibit less nonlinearity than real soil water curves, are used to transfer recharge from the land surface to the water table under steady conditions (WSRC-TR-2004-00106, Figure 2-7). Horizontal and vertical saturated hydraulic conductivities of cells with computed saturation less than 90 percent are set to 3.5 x 10^{-5} cm/s (1.2 x 10^{-6} ft/s) to minimize lateral flow and thereby ensure that modeled water movement in the vadose zone is vertically downward.

Table 4-8 Saturated Zone Material Properties DOE used in PORFLOW Modeling

Saturated Sediments	Saturated Effective Diffusion Coefficient (cm^2/s)	Effective Porosity (%)	Effective Dry Bulk Density (g/cc)	Effective Particle Density (g/cc)
Sandy Clayey	5.3×10^{-6} 4.0×10^{-6}	25	1.04	1.39

Note: Multiply by 0.16 to convert to in^2/s

K_d values used for radionuclide transport through the sandy saturated zone are the same as values assigned in the PORFLOW vadose zone modeling for undisturbed vadose zone material. The entire UTR aquifer [i.e., the UTR-UZ, UTR-LZ, and TCCZ] is assigned sandy K_d values (CC-FF-4 response, [DOE, 2011]). K_d values used for radionuclide transport through the clayey saturated zone are the same as values assigned in the PORFLOW vadose zone modeling for backfill. Clayey K_d values are only assigned to the GCU (CC-FF-4 response, (DOE, 2011)).

Hydrodynamic dispersion in the local FTF/PORFLOW transport model is represented by longitudinal and transverse dispersivities of 10 m (33 ft) and 1 m (3 ft), respectively, which are 10 and 1 percent of a nominal 100 m (330 ft) plume travel distance. In response to RAI-FF-3, DOE provided additional information that showed numerical dispersion was at an acceptable level for the level of physical dispersion assumed in the far-field model. DOE also provided additional information and references to support its assumed dispersivity, namely, an F-Area and H-Area seepage basin modeling study with dispersivities calibrated to contaminant breakthrough data that supported the rule of thumb longitudinal dispersivity used in the FTF model.

4.2.10.4 Model Verification, Calibration and Validation

PORFLOW acceptance testing on a previous version confirmed that the code conserves mass and satisfies Darcy's Law (WSRC-TR-2004-00106). A software quality assurance plan was developed for the version of PORFLOW used for the FTF calculations (WSRC-SQP-A-00028, G-TR-G-00002).

As mentioned above, the initial hydraulic conductivity fields of the finite-difference GSA/PORFLOW saturated zone model were based upon the calibrated hydraulic conductivity fields of the predecessor finite-element GSA/FACT saturated zone model (WSRC-TR-2004-00106). The initial fields for the GSA/FACT model were based upon field and laboratory data, and were modified to final values during model calibration to match measured hydraulic heads and stream baseflow estimates from USGS gauge stations on the UTR creek and FMB (WSRC-TR-96-0399). Hydraulic conductivity was modified in the GSA/FACT model first globally within each hydrostratigraphic unit, and then locally by the following steps:

- Multipliers were used to rescale the conductivity field of each hydrostratigraphic unit.
- Minimum and maximum value clipping parameters were used to eliminate extreme values.
- Affine corrections were used to smooth and thereby lessen the variability of laboratory-derived, upscaled hydraulic conductivities.

- Vertical conductivity was modified with a maximum K_h:K_v ratio parameter used to increase K_v while holding K_h constant.

- Vertical conductivities of cells adjacent UTR creek, FMB, and McQueen Branch were increased based upon known channel incisement and backfilling with permeable sediment (WSRC-TR-96-0399).

The initial hydraulic conductivity fields of the GSA/PORFLOW model were modified to the final calibrated parameter set (e.g., Figures 4-9 and 4-10) by again matching model results with measured hydraulic heads and stream baseflow estimates (WSRC-TR-2004-00106). The following additional modifications to the GSA/FACT calibrated values were made:

- Horizontal hydraulic conductivity in the UTR-UZ was increased 25 percent
- Vertical hydraulic conductivity in the TCCZ was decreased 50 percent
- Horizontal hydraulic conductivity in the UTR-LZ was increased 35 percent

To finalize GSA/PORFLOW calibration, Flach (WSRC-TR-2004-00106) performed particle tracking simulations to compare groundwater travel times. Only one particle track was seeded near the FTF (WSRC-TR-2004-00106, Figure 4-6). Flach (WSRC-TR-2004-00106) found that GSA/PORFLOW travel times were typically longer than GSA/FACT travel times, so the maximum recharge rate was increased from 0.12 cm/day (18 in/yr) in GSA/FACT to 0.13 cm/day (19 in/yr) in GSA/PORFLOW as a final model calibration step. Flach (WSRC-TR-2004-00106) also compared GSA/PORFLOW tritium transport simulations for E-Area against the GSA/FACT simulations of (WSRC-TR-2003-00432); Flach found the peak concentrations were similar, but individual runs varied on the order of ±25 percent. He concluded the particle tracking and transport comparison indicate the velocity field of the two models is similar.

Hydraulic head as a calibration target has a maximum uncertainty of 1 m (3 ft) (WSRC-TR-96-0399). Flach and Harris (WSRC-TR-96-0399) set a calibration goal of 1 m (3 ft) for the root-mean-square hydraulic head residual for the regional GSA/PORFLOW flow model. Furthermore, Flach and Harris (WSRC-TR-96-0399) defined a calibration goal of 5 to 10 percent of the total hydraulic head variation in a given aquifer for the maximum head residual. Total hydraulic head variation within the Gordon aquifer is given as ~24 m (~80 ft), suggesting the maximum head residual would be acceptable at a level of 1.20 to 2.40 m (4 to 8 ft) (WSRC-TR-96-0399). Total head variation within the UTR aquifer is given as ~49 m (~160 ft), suggesting the maximum head residual would be acceptable at a level of 2.40 to 4.80 m (8 to 16 ft) (WSRC-TR-96-0399). The maximum residual in the UTR-LZ of 8.20 m (27 ft) (WSRC-TR-2004-00106) greatly exceeds this calibration goal.

Flach and Harris (WSRC-TR-96-0399) discuss excellent agreement between measured and modeled hydraulic heads within the Gordon aquifer, which they attribute to this confined aquifer having a very simple head variation, unaffected by streams other than the UTR. The UTR-LZ exhibits large residuals where the water table drops sharply to the UTR creek and in the vicinity of H-Area (WSRC-TR-96-0399), but the well locations having the largest residuals are not in the vicinity of FTF (WSRC-TR-2004-00106, Figure 3-4(b)). High residuals west and north of F-Area meet the calibration goal of 5 to 10 percent of the total hydraulic head variation in the aquifer (WSRC-TR-96-0399). Flach and Harris (WSRC-TR-96-0399) found excellent agreement between measured and modeled hydraulic heads within the UTR-UZ. Flach (WSRC-TR-2004-00106) suggests that more extensive calibration efforts would likely improve the hydraulic head residuals of the GSA/PORFLOW model.

DOE used characterization and monitoring field data from E-Area to validate aspects of the regional GSA/PORFLOW saturated zone model. These data included seepline information (FTF PA), which is compared to seepline discharge predicted by the model. Flach and Harris (WSRC-TR-96-0399) suggest that agreement between measured and modeled baseflow is acceptable within an uncertainty range of ±20 to 50 percent. Simulated baseflow is generally consistent with baseflow estimates developed from field observations (see Table 4-9).

Table 4-9 Comparison of Estimated and GSA/PORFLOW-Modeled Baseflows (WSRC-TR-2004-00106)

Stream	Estimated Baseflow m^3/s (ft^3/s)	Modeled Baseflow m^3/s (ft^3/s)
UTR & Tributaries	0.52 (18.2)	0.32 (11.4)
Fourmile Branch	0.07 (2.6)	0.11 (3.8)
McQueen Branch	0.04 (1.5)	0.07 (2.4)
Crouch Branch	0.05 (1.8)	0.05 (1.7)

DOE also compares modeled streamlines with E-Area tritium plume trajectory information in an effort to validate that the regional GSA/PORFLOW saturated zone model which reproduces known plume trajectories in map view (FTF PA). Other important points of comparison, such as travel time and concentration, are unaddressed in this comparison. Flach (WSRC-TR-2004-00106) suggests the model is validated with respect to hydraulic conductivity because 83 percent of the calibrated GSA/PORFLOW hydraulic conductivity field agrees or is neutral with respect to measured hydraulic conductivity data (WSRC-TR-2004-00106).

4.2.10.5 Compliance Point

NRC guidance found in NUREG-1854 (NRC, 2007) indicates that after the end of the institutional control period, the receptor evaluated to demonstrate compliance with the 10 CFR 61.41 performance objective is assumed to be located at the point of maximum exposure located outside of the disposal site that consists of the area encompassing the disposal units (or tanks in the case of FTF) and outside a buffer zone outside the disposal units or area. An appropriate buffer zone is expected to extend approximately 100 m (330 ft) beyond the disposal units in most cases (NRC, 2007). NUREG-1854 also indicates that in some cases the point of maximum exposure might occur beyond the 100 m (330 ft) buffer zone. For example, the point of maximum exposure might occur beyond the 100 m (330 ft) buffer zone in complex hydrogeologial systems.

DOE calculates potential groundwater exposure concentrations at a point located 100 m (330 ft) from the FTF boundary using the local FTF/PORFLOW model (Figure 4-7). However, because (i) the transport paths away from FTF sources do not occur along a straight line, (ii) the transport paths traverse up to three aquifers, and (iii) sources are located at a variety of distances from the downgradient edge of the disposal site, the actual travel distances to reach a 100 m (330 ft) location beyond the FTF site are, in fact, greater than 100 m (330 ft) for many FTF sources. Table 5.2-2 of DOE's PA shows the approximate travel distances for constituents released from the FTF tanks to the 100 m (330 ft) boundary (SRS-REG-2007-00002, Rev. 1).

As stated above, DOE assumes the groundwater concentrations 100 m (330 ft) from the FTF site are the highest concentrations in the area 100 m (330 ft) or farther from the site. DOE indicates that this assumption is supported by Figures 5.2-3 and 5.2-4 in DOE's PA (SRS-REG-

2007-00002, Rev. 1), which presents the plume (in plan view and cross-sectional maps) that would result from a continuous and conservative (no decay or sorption) tracer. Peak concentration is observed to decrease monotonically with travel distance from the source zone, as a result of hydrodynamic dispersion. DOE indicates that no physical mechanism exists to concentrate contamination beyond the source zone in the fully three-dimensional PORFLOW simulations. Therefore, DOE concludes that monitoring wells assumed to be located at 100 m (330 ft) from the disposal site are adequate to capture the peak concentration that can occur at or beyond 100 m (330 ft) (SRS-REG-2007-00002, Rev. 1).

DOE calculates 100 m (330 ft) concentrations for five sectors (Sectors A – E) in the local FTF/PORFLOW model, as shown on Figure 5.2-5 of DOE's PA (SRS-REG-2007-00002, Rev. 1). The peak concentration values for the 100 m (330 ft) results are recorded for the three aquifers of concern (i.e., UTR-UZ, UTR-LZ, and Gordon aquifers). The concentration for each aquifer represents peak concentration in any vertical computational mesh within the aquifer. The vertical thicknesses of the computational grid are less than 3 m (10 ft) in the UTR-UZ, and less than 4.60 m (15 ft) in the UTR-LZ. Well screen averaging was not used to determine the concentrations for dose calculations because the typical well screen length of 6.10 m (20 ft) is similar enough to the thickness of the computational grid. Dividing the results into sectors was necessary to allow the large amount of concentration data from PORFLOW to be stored and used by the GoldSim dose calculator model, and to allow evaluation of variability in peak concentration for different source areas of the FTF. The five sectors are analyzed for each radionuclide and chemical to find the maximum groundwater concentrations 100 m (330 ft) from the FTF. DOE also calculates 1m (3 ft) concentrations for four sectors (Sector 1A–1D), as shown in Figure 5.2-5 of DOE's PA (SRS-REG-2007-00002, Rev. 1). Using the sectors to determine the highest groundwater concentrations causes the calculated peak doses to be higher than they actually would be, because the peak concentrations are determined for each radionuclide independent of the location within the sector.

4.2.10.6 Data and Model Uncertainty

DOE couples several PORFLOW models (e.g., near-field flow and transport models for each individual FTF source, regional far-field flow and transport models, and local far-field flow and transport models) to assess the cumulative contributions of various sources and radionuclides to potential dose at the point of compliance. Thus, one area of key concern for NRC staff was the coupling and linkages between all of the various PORFLOW models that comprise the base case, Configuration A, that is heavily relied on by DOE for its compliance demonstration. DOE uses output from its near-field model for each individual source to compute the flux of radionuclides to the local far-field transport model. However, differences in the discretization of the near-field and far-field transport models could lead to large changes in center-line plume concentrations along flow paths away from FTF sources that may not mimic the actual behavior of potential releases from the FTF facility.

As described in NRC RAI-FF-1, another source of uncertainty in DOE's far-field model is the treatment of the calcareous zone located in the lower portion of the UTR aquifer. While calcareous zone impacts on stability of various facilities has been widely studied, considerably less attention has been placed on the properties of this zone that might impact contaminant flow and transport. DOE argues that particles released from the FTF will not traverse this zone to a significant extent and that monitoring data to date has not revealed any noticeable impacts on hydraulic heads or contaminant transport. Discussions with DOE revealed that mapping of surface water seeps along UTR and FMB have not focused on surface seeps or other features associated with these zones, although future work in this area could be conducted. When

questioned by NRC staff, DOE also indicated that in at least one location significantly downstream of FTF, cave-like features have been observed.

4.2.11 NRC Evaluation of Hydrology and Far-Field Transport

4.2.11.1 NRC Evaluation of Model Construction Including Boundary Conditions

Model construction uncertainty is discussed in this section. It is commendable that DOE developed their hydraulic conductivity parameter set from field and laboratory characterization data (WSRC-TR-96-0399), rather than starting with a uniform parameter set and adding variation as needed to match measured hydraulic heads and estimated stream baseflows. However, as discussed more in Section 4.2.11.3, there is a lack of transparency in this process due to the intermediate step of using a prior GSA FACT calibrated model prior to migrating to the PORFLOW model used in DOE's FTF PA (WSRC-TR-2004-00106).

With regard to grid discretization, in response to RAI-FF-3 (DOE, 2011) DOE presented information to support the finite element size used in the FTF/PORFLOW model to ensure that the level of numerical dispersion in the FTF PORFLOW model is at an acceptable level. In the response to RAI-FF-3 (DOE, 2011), DOE also presented information that shows that additional grid refinement may be necessary to reduce numerical dispersion in cases of very low to no assumed physical dispersion. For example, if no physical dispersion is assumed, then the peak concentrations associated with a pulse release of a conservative tracer are shown to be a factor of around three to four times higher with a grid refined by a factor of two in each dimension (or a factor of 8 times more elements). Thus, NRC staff's review focused on evaluating the basis for the assumed level of physical dispersion in DOE's FTF PORFLOW model as described below.

4.2.11.2 NRC Evaluation of Material Properties and Parameters

This section discusses NRC staff's evaluation of material property assignments, including transport parameters. As discussed above, DOE satisfactorily demonstrated that physical dispersion dominates numerical dispersion in the local FTF/PORFLOW saturated zone model and that numerical dispersion is at an acceptable level given the level of physical dispersion assumed (DOE, 2011, Figure RAI-FF-3.2). The results presented in response to RAI-FF-3 (DOE, 2011) also indicate that physical and numerical dispersion combined in the FTF/PORFLOW model account for significantly lower concentrations compared to modeling simulations where little to no physical dispersion is assumed and/or less numerical dispersion is simulated over the range of grid resolutions studied. Thus, the assumed level of dispersion of contaminant releases and grid construction can have a significant effect on PA results.

In RAI-FF-3 (Camper, 2010), NRC staff expressed concern with the specified dispersivity used in the local FTF/PORFLOW model in that it might lead to excessive plume spread. In its response to RAI-FF-3 (DOE, 2011), DOE indicated that the specified dispersivity values they selected from scale-dependent correlation data (Gelhar, 1992) are supported by an optimization study for F-Area and H-Area seepage basins (WSRC-TR-2002-00291) and that this study had reliable source term data and was the most pertinent to FTF. However, NRC continues to have concerns with the basis for the assumed dispersivities in FTF/PORFLOW modeling. Because the optimization study was based on (1) calibration to contaminant breakthrough data (less reliable than calibration to plume distributions), and (2) preferential flow pathways away from the seepage basins in the UTR-UZ are thought to exist that may not be reflective of flow and transport away from FTF sources (see RAI-FF-1, path forward item 6, in (Camper, 2010)), it is not clear that the F-Area and H-Area seepage basin modeling effort should be considered the

most reliable study of field-scale dispersivity for the FTF facility. Lower dispersivities (e.g., α_L = 1.50 m [5 ft], α_{Th} = 0.10 m [0.33 ft], α_{Tv} = 0.01 m [0.03 ft]) from the tritium optimization study at E-Area Old Burial Ground, although considered less certain, may be more representative of dispersion along flow paths away from the FTF. In general, (Gelhar, 1992) supports the conclusion that the most reliable field data are typically associated with relatively low field-scale dispersivities, which would tend to lessen plume spread and result in higher doses at the point of compliance. A number of modeling studies for the larger SRS site, also support lower dispersivities based on calibration to three-dimensional plume spread. Dispersivities derived from calibration to three-dimensional plume distribution data are expected to be more reliable than calibration to breakthrough curve data (Gelhar, 1992).

Gelhar (1992) also notes that very low vertical transverse dispersivities have been observed at several sites and are typically two orders of magnitude lower than longitudinal dispersivities. Because the version of PORFLOW used in the FTF assessment does not allow separate specification of transverse horizontal and vertical dispersivity, vertical dispersion may be overestimated at one-tenth of the longitudinal dispersivity or 1 m (3 ft) rather than 1/100 of the longitudinal dispersivity or 0.10 m (0.33 ft) in the FTF/PORFLOW model. Furthermore, excessive vertical dispersion may have also been exacerbated by the significant vertical gradient that exists near the groundwater divide that leads to longitudinal dispersivities being applied in the vertical direction. Because longitudinal dispersivities are significantly higher than transverse dispersivities, dispersion in the vertical direction may be exaggerated. DOE also evaluated the impact of lower assumed vertical dispersion and indirectly the impact of strong vertical gradients in its response to RAI-FF-3 (DOE, 2011) and found more modest but still significant impacts on concentration.

With regard to K_ds, DOE's response to RAI-FF-4 (DOE, 2011) indicates that it will only pursue follow-up work related to soft zone sorption (e.g., Pu sorption to soft zone material in the presence of potentially high carbonate concentrations) if soft zones are determined to be a risk-driver. Among NRC staff's concerns were potential effects of pH and aqueous carbonate concentrations on sorption properties, particularly for actinides. A key point in DOE's response was that "...soft zones and carbonates are generally represented by very small and infrequent pockets that do not continuously run the length of the flow path of the plume" (DOE, 2011). NRC staff agrees that, if soft zones do not strongly affect flow paths, then the effect on sorption may be small. However, the role of soft zones on contaminant transport is uncertain, and more work is required to evaluate potential impacts of these zones on contaminant transport. DOE progress in this area will be evaluated during the monitoring phase. Evaluation of the larger set of K_ds values adopted by DOE for the saturated zone model is discussed in Section 4.2.9.4, because the same values were used in the vadose zone transport model.

4.2.11.3 NRC Evaluation of Model Calibration

With respect to GSA/PORFLOW model calibration, the maximum hydraulic head residual in the UTR-LZ of 8.2 m (27 ft) (WSRC-TR-2004-00106) greatly exceeds the calibration goal of no more than 5 to 10 percent of the total hydraulic head variation in the UTR aquifer for the maximum head residual. The UTR-LZ exhibits large residuals where the water table drops sharply to the UTR creek and in the vicinity of H-Area (WSRC-TR-96-0399; WSRC-TR-2004-00106), but the well locations having the largest residuals are not in the vicinity of FTF (WSRC-TR-2004-00106, Figure 3-4(b)). Additional work may be needed to better define the hydraulic conductivity field in the vicinity of H-Area (WSRC-TR-96-0399; WSRC-TR-2004-00106), but relatively poor H-Area calibration should not affect the flow field and transport times from the FTF for the purposes of this PA. High residuals west and north of F-Area meet the calibration

goal of 5 to 10 percent of the total hydraulic head variation in the aquifer (WSRC-TR-96-0399). Flach (WSRC-TR-2004-00106) suggests that more extensive calibration efforts would likely improve the hydraulic head residuals of the GSA/PORFLOW model. Given that the final calibrated hydraulic conductivity field was not generated directly from modification of an initial set of field and laboratory data, but rather from a calibrated predecessor GSA/FACT model parameter distribution, more extensive calibration is desirable both for transparency and model performance. It should be noted that hydraulic head residuals of the predecessor GSA/FACT model were less than those of the GSA/PORFLOW model used to support this PA (WSRC-TR-2004-00106). The maximum residuals of the predecessor GSA/FACT model did not exceed the stated calibration goals (WSRC-TR-2004-00106), unlike the GSA/PORFLOW model used to support this PA.

4.2.11.4 NRC Evaluation of Compliance Point

In response to RAI-FF-5, DOE presents additional PORFLOW simulation results that indicate plume concentrations decrease with distance from the source (DOE, 2011). DOE also indicates that for sources farther upgradient from the 100 m (330 ft) boundary (such as Tank 1), the plume centerline will cross the TCCZ that separates the UTR-UZ from the UTR-LZ inside the 100 m (330 ft) boundary. However, for sources nearer the 100 m (330 ft) boundary (such as Tank 47), DOE indicates that the plume centerline may or may not cross the TCCZ inside the 100 m (330 ft) boundary, suggesting that maximum concentrations for sources closer to the FTF boundary and 100 m (330 ft) compliance point may occur in the UTR-UZ, and maximum concentrations in the UTR-LZ (used in benchmarking—see Section 4.2.18.3) and in the Gordon aquifer (used in probabilistic modeling—see Section 4.2.18.7) may not be fully realized at the 100 m (330 ft) well location extracted from the FTF PORFLOW model.

The point of compliance in the Gordon Aquifer is risk-significant as DOE assigns a relatively high probability that a well will be drilled into this aquifer in the probabilistic analysis (see Table 4-10). Aquifer concentrations at this point in the aquifer system at FTF were used to assign a relative concentration to the Gordon Aquifer (factor of 20 lower than UTR) for use in the GoldSim model. Sensitivity analysis indicates well completion (aquifer) is the single-most important parameter in the probabilistic assessment given its well-defined treatment and impact in the model.

Table 4-10 Aquifer Exposure Probabilities and Relative Concentrations

Aquifer	Exposure Probability	Fraction of UTR-UZ Concentration	Weighted Concentration
UTR-UZ	0.13	1.0	0.13
UTR-LZ	0.44	1.0	0.44
Gordon	0.43	0.05	0.02
Total	1.0	NA	0.59

In an RAI resolution meeting held on July 21, 2011 (ADAMS accession number ML112070958), DOE confirmed NRC's understanding of the RAI-FF-5 response that concentrations in the Gordon Aquifer at the 100 meter (330 ft) boundary would tend to under-predict potential Gordon Aquifer concentration further down-gradient from the site. DOE did indicate that they thought a factor of 20 reduction in concentration in the Gordon Aquifer is sufficiently conservative for locations of higher concentration that may occur beyond the 100 meter (330 ft) boundary. Nonetheless, it is NRC's position—as stated in FTF scoping—that Gordon Aquifer

concentrations should not be used to assess compliance because the point of maximum exposure should not be dependent on the probability of well completion if higher concentrations are observed in a viable aquifer unit that could support the groundwater dependent pathways evaluated in the biosphere modeling (see meeting summary http://www.em.doe.gov/pdfs/Jan%2008%20public%20notes%20_final_.pdf). Thus, while risk information related to well completion in the Gordon Aquifer is informative, information regarding dose at the point of maximum exposure in the UTR aquifer is perhaps more appropriate for comparison against performance objectives. Thus, NRC presents modified results using just those realizations with aquifer completion in the UTR aquifer in Table 4-3 presented above.

4.2.11.5 NRC Evaluation of Data and Model Uncertainty

With regard to linkage between the near-field and far-field models, NRC noted a significant amount of dispersion for HRRs directly below the source tanks. It is not clear if the level of dispersion simulated in the FTF/PORFLOW model is consistent with field observations of dispersion of contaminant plumes in the aquifer. Currently, the vertical cell width exiting the near-field model is typically around a foot while the vertical thickness of the saturated zone cells is a few meters. RAI-FF-3 and FF-6 figures may also suggest a significant amount of upgradient dispersion, although NRC staff is not confident of this observation due to ambiguity with respect to the location of the cross-sections in the figures. In any event, DOE should evaluate the need for additional vertical or horizontal mesh refinement to ensure that contaminant plumes are not artificially dispersed over the volume of the cells in the far-field model in future updates to the PA or justify why the modeling approach is acceptable given the expected level of dispersion in a potential groundwater well. Comparisons of plume spread in the FTF model to actual observations of contaminant plume spread for more mobile and less mobile plumes would also be instructive with respect to the adequacy of the FTF models in predicting contaminant concentrations at a down-gradient well.

With regard to calcareous zones, the GSA and FTF PORFLOW models do not explicitly account for the impact of these zones on contaminant fate and transport (other than considering the impact of mud fraction on hydraulic conductivity). NRC expressed concerns in RAI-FF-1 regarding the potential impact of these zones on contaminant flow and transport (Camper, 2010). For example, transport rates in dual porosity/dual permeability aquifers can be rapid and cover large distances in relatively short periods of time with little dilution or attenuation (Worthington, 2007). Localized fluid piracy along discrete flow paths may occur (cf. Thayer, et al., 1995). Thayer, et al. (1995) indicate that contaminant data for the Burial Ground Complex (located in E-Area, adjacent F-Area at the GSA) and the Chemical, Metals, and Pesticide Pits (located off the GSA) support the hypothesis that the highly porous, high hydraulic conductivity calcareous zones serve as pseudo-karst-like conduits that preferentially transport contaminants. These contaminant data were not provided nor assessed during this evaluation. While DOE suggests that well water levels implicitly reflect the broad influence of soft zones (DOE, 2011, RAI-FF-3 response), NRC staff thinks that low hydraulic head residuals for F-Area do not necessarily imply that the single-domain conceptual model of flow and transport in the UTR-LZ is adequate. Because soft zones are isolated and intermittent, monitoring wells will more often intersect relatively low permeability facies within the Santee Formation than potentially relatively high permeability facies of the soft zones, and contaminant concentrations measured in the low-permeability facies may be lower than concentrations measured in a fast-flow soft zone. The model may be adequately capturing flow behavior within the relatively low permeability sediments of the UTR-LZ, while failing to capture potentially risk-significant flow behavior within highly porous, highly permeable (but relatively low volume) fast flow pathways in the same aquifer.

Ten surface water stations regularly monitored in F-Area do not sample UTR-LZ water (see July 21, 2011, RAI resolution meeting summary, ADAMS accession number ML112580225) and no formal mapping to identify calcareous zone seeps along stream valleys has been conducted. Nonetheless, DOE indicates that a field mapping activity such as this could be incorporated in the future . NRC staff support such an activity to evaluate the impact of potentially highly porous and conductive soft zones in the UTR-LZ on the FTF facility modeled hydrogeologic system. DOE progress in this area will be evaluated during the monitoring period. If calcareous zone seeps are identified, tracer studies in the F-Area UTR-LZ, using innocuous tracers that are commonly used to understand preferential flow and transport could be conducted to better understand the effect of these zones on contaminant flow and transport. Results from any such tracer studies would be evaluated by NRC during the monitoring period.

4.2.11.6 Far-field Review Results and Recommendations

NRC staff notes that DOE's far-field model presents an acceptable framework to facilitate decision-making regarding FTF closure. However, NRC staff note a number of areas where DOE could potentially make improvements to its far-field model to reduce the uncertainty in dose modeling predictions.

NRC staff key review results related to far-field modeling are as follows:

1. Far-field model construction and calibration are generally acceptable; however, the calibration process could be improved and made more transparent in future PA updates.

2. Dispersion may be over-represented in DOE's far-field model.

3. NRC staff is convinced that large voids do not currently exist in the subsurface along FTF flow paths to the 100 m (330 ft) point of compliance. Calcareous zones that have undergone dissolution may still represent high permeability zones that could have a significant affect on contaminant flow and transport. Additional information could be collected during the monitoring period to support DOE's modeling treatment of the calcareous zones in the lower portion of the UTR aquifer.

NRC staff key recommendations related to far-field modeling are as follows:

4. DOE should consider how it might improve far-field model calibration and transparency in future updates to its PA. (Medium Risk-Significance, Intermediate Term)

5. DOE should continue to evaluate the appropriateness of selected transport parameters (e.g., dispersivities and K_ds, particularly for calcareous zones) and selection of sorption models (see discussion in Section 4.2.9.4 on Pu transport) during the monitoring period. (Medium Risk-Significance [Pu sorption High Risk-Significance], Intermediate Term [Pu sorption Short Term])

6. DOE should consider additional data collection related to calcareous zone outcrop locations and tracer tests to provide further support for the adequacy of its modeling treatment of the UTR-LZ aquifer. (Medium Risk-Significance, Short to Intermediate Term)

7. Finally, DOE should evaluate the need for additional vertical or horizontal mesh refinement to ensure that contaminant plumes are not artificially dispersed over the volume of the cells in the far-field model. Comparisons of plume spread in the FTF model to actual observations of contaminant plumes for more mobile and less mobile plumes would be instructive with respect to the adequacy of the FTF models in predicting contaminant concentrations at a down-gradient well. (Medium Risk-Significance, Intermediate Term)

4.2.12 Dose Methodology

The dose methodology used by DOE in the FTF PA process was the application of dose conversion factors to an all-pathways exposure scenario. This methodology is widely used in PAs and consists of multiplying the radionuclide concentration in air, water, or soil (that a receptor might be exposed to through any of the various pathways) by the dose conversion factor specific to that ingestion or inhalation process and radionuclide.

The calculation process and the dose factors used for the all-pathways exposure PA are described in DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1). The exposure pathways include drinking water dose from groundwater, all-pathways dose from groundwater (including drinking water dose), air dispersion pathways, and intruder pathways. The primary mechanism for transport of radionuclides from the FTF to a human receptor is expected to be leaching (after degradation of the grouted tanks and vaults) to the groundwater and human consumption and use of well water for domestic purposes. The exposure pathways considered in the PA involving contaminated well water include direct ingestion, ingestion of milk and meat from dairy and beef cattle consuming contaminated well water, and ingestion of plants and animals grown and raised in areas irrigated with contaminated well water.

In the analysis performed by DOE for the PA, radionuclide dose conversion factors from the Federal Guidance Reports (FGR) developed by the EPA and International Commission on Radiological Protection (ICRP) were used. Ingestion and inhalation (internal) dose conversion factors were taken from ICRP-72 (ICRP, 1995), and external dose conversion factors were taken from FGR 12 (EPA, 1993). The dose conversion factors were included in the PA document Table 4.7-1 (SRS-REG-2007-00002, Rev. 1).

4.2.13 NRC Evaluation of Dose Methodology

The dose methodology implementation of the FTF PA is well supported and suited for the purpose. Numerous NRC guidance documents provide recommendations on the approach and use of the specific dose conversion factors used in the FTF PA process. These include NUREG–1573 (NRC, 2000), which provides guidance on the use of pathway dose conversion factors for calculating doses via the potential exposure pathways, and NUREG–1757 (NRC, 2003, Volume 2, Appendix I), which provides guidance on the use of specific dose conversion factors such as those developed by the EPA and published in FGR Nos. 11 and 12 (EPA, 1988, 1993), as well as those developed by the ICRP and published in ICRP-72 (ICRP, 1995).

4.2.14 Protection of the Public

To demonstrate compliance with 10 CFR 61.41, DOE compares the peak all-pathways dose estimated in the PA to 0.25 mSv (25 mrem) Total Effective Dose Equivalent (TEDE). The PA uses a hybrid deterministic-probabilistic approach to estimate the dose to a hypothetical future member of the public. The member of the public is assumed to be located at the boundary of

DOE's controlled area until the end of the active institutional control period (i.e., 100 years after FTF closure). After the active institutional control period has elapsed, the member of the public is assumed to move to the point of maximum exposure at or outside the FTF 100 m (330 ft) buffer zone. The all-pathways dose estimated in the PA combines water-dependent pathways originating from contaminated groundwater and air pathways originating from the release of volatile radionuclides from the waste forms. DOE also performed probabilistic analyses to assess the impacts of variability and uncertainty on a demonstration of compliance with 10 CFR 61.41.

The peak all-pathways TEDE to a member of the public was estimated in the deterministic analysis (from Configuration A) to be 0.03 mSv/yr (2.5 mrem/yr) occurring at year 10,000 using the highest 100 m (330 ft) groundwater pathway dose results during the 10,000-year performance period in combination with the air pathway results. The water ingestion pathway contributed 59 percent of the peak dose with the principal radionuclide being Ra-226. The second ranked primary pathway was vegetable ingestion with 26 percent of the peak dose and the same principal radionuclide. DOE also evaluated for time periods out to 20,000 years to assess longer-term impacts and out to even longer periods to assess uncertainty and sensitivity in the PA models. Results from these evaluations are presented in Table 4-3 and discussed in detail in sections of this report on uncertainty and sensitivity analyses that follow.

Using the probabilistic model, DOE estimates a peak of the mean all-pathways dose of 0.05 mSv/yr (4.8 mrem/yr) to a member of the public within 10,000 years for Configuration A. Including the alternate configurations (excluding Configuration G), DOE estimates a peak of the mean all-pathways dose of 0.14 mSv/yr (14 mrem/yr) to a member of the public within 10,000 years. In response to RAI-PA-1, DOE developed Configuration G to address NRC staff RAIs regarding model support for the performance of risk-significant barriers. Configuration G assesses the impacts when select barriers of concern are modified simultaneously. The peak groundwater pathway dose for Configuration G is 1.25 mSv (125 mrem/yr) to a member of the public within 10,000 years.

4.2.15 NRC Evaluation of Protection of the Public

NRC staff considers DOE's use of 0.25 mSv (25 mrem) TEDE to demonstrate compliance with 10 CFR 61.41 without specific consideration of individual organ doses acceptable for incidental waste determinations. DOE's PA indicates that compliance with 10 CFR 61.41 for protection of the public can be demonstrated. However, NRC staff provides several recommendations in this document where enhanced model support would be needed prior to tank closure to provide reasonable assurance that the closure activities would comply with requirements for the protection of the public at 10 CFR 61.41. It is important to note that DOE modeling suggests that if assumptions regarding the timing of the degradation of the waste tanks and the release of radionuclides from the CZ within 10,000 years prove invalid, then the peak doses that currently occur outside the period of performance in DOE's PA may exceed 0.25 mSv (25 mrem) TEDE within 10,000 years.

4.2.16 Protection of Intruders

In order to demonstrate compliance with 10 CFR 61.42, DOE quantitatively evaluated potential intruder scenarios. DOE considered the bounding scenarios to be an acute intruder-drilling scenario and a chronic intruder-agricultural scenario. While a numerical performance objective is not provided in 10 CFR 61.42, a limit of 5 mSv/yr (500 mrem/yr) was applied by DOE as

suggested in the EIS for the 10 CFR Part 61 rulemaking, for development of waste classification requirements of 10 CFR 61.55, and in NUREG-1854.

The acute intruder drilling scenario assumes that an intruder drills a well into a transfer line at the end of institutional controls after 100 years. The intruder is externally exposed to drill cuttings as well as inhalation and inadvertent ingestion of the drill cuttings. The peak dose for the acute intruder-drilling scenario was 0.02 mSv (1.6 mrem) at year 100, primarily due to direct exposure to drill cuttings from the principal radionuclide Cs-137/Ba-137m.

The chronic intruder-agricultural scenario assumes that at the end of institutional controls after 100 years, a farmer lives near the well drilled as part of the acute intruder-drilling scenario (i.e., waste from a transfer line) and consumes food crops, meat, and milk from animals raised there. These food crops are grown in a plot that is contaminated by drill cuttings as well as irrigated with contaminated well water. The animals raised by the farmer drink water from the contaminated well and eat fodder from a pasture irrigated with well water. The intruder-agricultural is also exposed by ingestion and inhalation of well water and garden soil and by direct exposure (plume shine) from the contaminated garden soil. The peak dose within 10,000 years for the chronic intruder scenario was 0.73 mSv/yr (73 mrem/yr), with the most important pathway being the ingestion of vegetables contaminated with drill cuttings at the time of intrusion at 100 years. The principal radionuclides by dose were Sr-90/Y-90 (56 percent) and Cs-137/Ba-137m (44 percent) for the vegetable ingestion pathway. The peak dose within 20,000 years was slightly higher at 0.75 mSv/yr (75 mrem/yr) due to groundwater dependent pathways with most of the dose from vegetable and water ingestion from Np-237. Other important radionuclides from the groundwater pathway include Th-229 and U-233, although these radionuclides were eliminated from the list of HRRs based on the fact that the inventories of these radionuclides were expected to be insignificant (inventories were overestimated in the Rev. 1 PA).

DOE considered a number of additional intruder scenarios but found them (i) to not be applicable, or (ii) to be bounded by the two scenarios that were evaluated. The intruder-construction, intruder-discovery, and bio-intrusion scenarios were not considered applicable because of the depth and form of the waste. The contamination will be from 3.05 to 12.19 m (10 to 40 feet) below the FTF closure cap and will be contained in protective barriers that are assumed to deter intrusions and help to clearly distinguish the waste from the surrounding soil. The chronic intruder-resident scenario and the chronic intruder-recreational hunting/fishing scenario were considered to be bounded by the intruder agriculture scenario. The pathways associated with the intruder constructing a house over the containments were considered insignificant because of the shielding provided by the disposal depth of the contaminants, as well as the waste tanks and ancillary equipment containment.

DOE provided additional probabilistic analyses in the FTF PA to investigate uncertainty and sensitivity in the intruder dose results. The uncertainty analyses were intended to provide information regarding how uncertain model input parameters would be propagated through the various model results. The sensitivity analyses were intended to provide information on how the various individual stochastic input parameters affect the dose results.

Results of the probabilistic intruder analysis indicated peak of the mean doses of 6.39 mSv/yr (639 mrem/yr) for the chronic intruder (and median doses of 1.12 mSv/yr [112 mrem/yr]) associated with groundwater dependent pathways and waste release from the tanks. Important parameters related to dose include Pu inventory, Pu and Tc solubility limits, Pu sand K_d; steel

liner failure times, and aquifer thickness (as it impacts aquifer dilution). These results are consistent with the 10 CFR 61.41 probabilistic sensitivity analysis.

DOE evaluated two sensitivity cases: (i) chronic exposure to drilling into a 10 cm (4 in) transfer line and (ii) acute exposure to drilling into a tank (Tank 18 was the only tank evaluated), although the former is considered unlikely (only 0.24 percent of the transfer lines are 10 cm [4 in] lines) and the latter is not considered credible due to the presence of engineered barriers whose long-term strength and durability would preclude intrusion, as well as providing a recognizable waste form. The sensitivity results indicate doses of 1.25 mSv/yr (125 mrem/yr) for chronic exposure from drilling into a large (10 cm [4 in] versus 8 cm [3 in]) transfer line. A potential receptor could get a dose of 0.07 mSv/yr (7 mrem/yr) from acute exposure (inhalation, incidental soil ingestion, and external dose) while drilling a well through Tank 18. Chronic exposure from intrusion into Tank 18 following the drilling event was not evaluated in DOE sensitivity analysis. All of the deterministic intruder scenario doses are less than 5 mSv (500 mrem) per year all-pathways TEDE. The peak of the mean of the probabilistic assessment is greater than 5 mSv/yr (500 mrem/yr) at 6 mSv/yr (639 mrem/yr) associated with releases from the tanks and groundwater-dependent pathways.

4.2.17 NRC Evaluation of Protection of Intruders

While DOE discussed a number of conservatisms in the intruder analysis and the use of many site-specific, stochastic biosphere parameters to evaluate dose to an inadvertent intruder, NRC staff noted exceptions. First, the plant transfer factors (bioaccumulation factors) used to calculate plant uptake of radionuclides in the vegetable pathway dose calculation were not stochastic (i.e., single expected values) and were developed from the literature for root vegetables only. The use of a single expected value removes these uncertain and variable values from uncertainty and sensitivity analyses in the probabilistic pathway dose model. The use of root vegetable values only for a resident farmer scenario in the SRS area is difficult to justify technically. Through RAI questions and responses (see RAI-IT-1 and RAI-IT-2 (DOE, 2011)), DOE attempted to provide information as to the low risk-significance of variable and leafy component plant transfer factors and specifically for the intruder dose calculation. However, uncertainties in system performance and key risk drivers may invalidate conclusions with respect to the risk-significance of plant transfer factor uncertainty and inclusion of different classes of vegetables in the dose calculations. Furthermore, in the interest of transparency and traceability, as well as given the importance of the vegetable ingestion pathway to the member of the public dose calculation discussed previously, it is recommended that future updates to the FTF PA and other PAs at SRS provide a more technically defensible selection of plant transfer factors and consideration of the impact of variability and uncertainty of the dose modeling results to this important parameter.

Second, DOE used an expected value of 337 L/yr (0.92 L/day) for drinking water ingestion rate with a maximum value of 730 L/yr (2 L/day) for the water ingestion pathway dose calculation (see RAI-IT-3 (DOE, 2011)). This pathway is generally the largest contributor to total dose in a resident farmer scenario. A number of references, in particular Estimated Per Capita Water Ingestion and Body Weight in the United States--An Update EPA-822-R-00-001 (October, 2004) would suggest that 2 L/day is a more defensible and conservative expected value. Taking the site-specific nature of the warmer climate at SRS and the resident farmer scenario whose water comes from an onsite well, the maximum value of this important uncertain parameter could be above 2 L/day. Again, it is recommended that future updates to the FTF PA and other PAs at SRS provide a more technically defensible selection of water ingestion rate. The staff notes that water ingestion has a linear relationship to water ingestion pathway dose, so a doubling of water

ingestion would tend to double the dose from this pathway. For the intruder calculation, this would provide dose estimates which are still well below the 5 mSv/yr (500 mrem/yr) performance objective.

DOE developed reasonable inadvertent intruder scenarios to evaluate protection of inadvertent intruders and demonstrated that performance objectives found in 10 CFR 61.42 ("Protection of individuals from inadvertent intrusion") could be achieved. All intruder doses are calculated to be less than 5 mSv/yr (500 mrem/yr), except under what is considered by DOE to be very pessimistic conditions discussed in the Intruder Uncertainty/Sensitivity Analysis Section 6.5 of DOE's PA. Nonetheless, given the uncertainties identified in the 10 CFR 61.41 evaluation that spill over into the 10 CFR 61.42 evaluation (i.e., uncertainty in dose predictions associated with groundwater-dependent pathways), NRC's evaluation of 10 CFR 61.42 is tied to the 10 CFR 61.41 evaluation. As issues are closed with respect to the 61.41 evaluation, then so will issues be resolved for the 61.42 evaluation. It is important to note that considering the higher allowable dose of 5 mSv/yr (500 mrem/yr) used for the intruder assessment (factor of 20 higher than the dose limit for the 10 CFR 61.41 analysis) offset by the expected dilution factors between the 1 meter (3 ft) (intruder) and 100 meter (330 ft) (member of the public under 61.41) points of assessment (e.g., about a factor of 5 for Pu 239), potential exceedance of the 10 CFR 61.42 performance objective would generally be less significant than the potential exceedance of the 61.41 performance objective.

4.2.18 Uncertainty/Sensitivity

DOE included an uncertainty and sensitivity analysis in its PA documentation. DOE used the PORFLOW and GoldSim modeling codes to perform the analysis. While PORFLOW was primarily used to support the deterministic analysis with single, best estimate values of parameters, additional single parameter sensitivity analyses were also conducted using PORFLOW. Details regarding PORFLOW model construction are found in Sections 4.2.8 (near-field model) and Section 4.2.10 (far-field model). DOE used the software program GoldSim, described in more detail below, to construct a probabilistic model that was used to evaluate both conceptual model and parameter uncertainty. Conceptual model uncertainty rests, for example, with our lack of knowledge regarding how a system should be represented or how it might evolve over time.

DOE considers different states of degradation of engineered barriers when evaluating conceptual model uncertainty like early tank grout failure, or the bypassing of concrete materials underneath the tank that would otherwise tend to attenuate or reduce the impacts from the tanks. Conceptual model uncertainty is evaluated by running alternative simulations in GoldSim and averaging the results together based on the expected likelihood of occurrence of these alternative realities. Parameter uncertainty is also evaluated in GoldSim. Parameter uncertainty rests with our lack of knowledge regarding what value a particular variable should be assigned. Sometimes the uncertainty being evaluated is more correctly attributable to variability such as the variability in corrosion rates for particular components or subcomponents, for example, or spatial variability in material properties such as permeability or porosity. GoldSim was also used to perform deterministic sensitivity analysis. In many respects, the GoldSim model can be considered an abstraction or simplification of the more complex, multi-dimensional PORFLOW models, although GoldSim was also used to supplement analyses that could not be conducted in the PORFLOW code (e.g., biosphere or dose modeling calculations). A gradient boosting algorithm was also used to interrogate the GoldSim probabilistic modeling results to identify important PA model sensitivities. DOE's approach to performing the

uncertainty and sensitivity analysis for the FTF is discussed in more detail in the sections that follow.

4.2.18.1 GoldSim Model Overview

The GoldSim code developed by the GoldSim Technology Group was used to perform a probabilistic uncertainty and sensitivity analysis to support the FTF PA and basis document. GoldSim is a modeling platform that includes many built-in functions and tools to allow users to perform a wide range of probabilistic simulations related to topics as diverse as financial planning and contaminant transport modeling. NRC recognizes that the resource demand associated with execution of the probabilistic analysis using the deterministic PORFLOW model is likely prohibitive given the (i) size of (number of elements in) the near- and far-field models, (ii) number of radionuclide chains being simulated, and (iii) the number of scenarios being considered in DOE's PA. The computational burden of executing a probabilistic analysis for a complex problem the scope of the FTF facility is somewhat alleviated through use of a simpler model such as GoldSim, which is perhaps more efficient at exploring important model sensitivities and predictive uncertainties. Nonetheless, use of a simpler model such as GoldSim to perform the probabilistic analysis is not without its own limitations and disadvantages. For example, GoldSim is not equipped to solve the set of differential equations implementing Darcy's law to compute hydraulic heads and flows. Instead, an abstraction or simplification of the two- and three-dimensional flow fields produced in the deterministic PORFLOW models was developed to produce one-dimensional flows (e.g., vertical velocities in the unsaturated zone and horizontal velocities in the saturated zone) as data inputs to the probabilistic GoldSim model. Additionally, the inherent differences between the multi-dimensional, PORFLOW models and the one-dimensional, analytical (GoldSim) model make it difficult to compare modeling results. This difficulty is partially alleviated through the benchmarking process discussed in detail in Section 4.2.18.2. Finally, while GoldSim is a valuable tool for performing probabilistic assessments, like any model, the usefulness of the probabilistic model is dependent on the model inputs. And while many parameters were varied in the analysis (nearly 2,000 stochastics were included [see page 658 of DOE's PA]), many others were not (SRS-REG-2007-00002, Rev. 1). Most notably, flow rates into the contaminated zone within individual configurations and many biosphere parameters such as plant transfer factors were not varied in the analysis.

In addition to evaluation of parameter uncertainty, one of the primary advantages of the probabilistic assessment is its consideration of alternative configurations or scenarios related to engineered barrier performance. As indicated in Section 4.2.8, DOE considers alternative scenarios primarily related to potential variability in engineered barrier performance and waste release. In addition to presenting PORFLOW deterministic and sensitivity analysis results for Configuration A (base case) and D (an alternative fast flow case), DOE also simulates and provides results for all alternative configurations (B–F) in its probabilistic assessment. Table 4-11 provides summary information regarding the alternative configurations.

4.2.18.2 Probabilistic Model Flow Velocity Abstraction

As mentioned above, the GoldSim model does not compute flow. The major hydraulic affect of the engineered closure cap and tank grout is to initially reduce flow into the tank system. Therefore, these barriers do not need to be explicitly modeled as hydraulic barriers in GoldSim. Instead, the expected flow through the contaminated zone is imposed as an upper boundary condition on the GoldSim model. Five flow rates, extracted 0.5 m (1.6 ft) underneath the basemat for each (i) tank type, (ii) configuration and (iii) time period (40 stress or time periods

over which flow is assumed to gradually change) from the PORFLOW output were geometrically averaged and entered as a deterministic flow rate applied to an element representing the contaminated zone in GoldSim (Taylor, 2009). Average PORFLOW velocities were also used as input to the GoldSim model for the saturated zone, varying by tank type (e.g., to account for differences on either side of the groundwater flow divide). Time markers for a conservative tracer were included in the PORFLOW output to deduce the travel time to the 100 m (330 ft) point of assessment considering both horizontal and vertical flow. The PORFLOW groundwater velocities were adjusted to reach agreement on expected travel times.

With regard to discretization of the model, the vadose zone is represented as a few discrete elements in the PORFLOW model for each tank type and 10 cells in the GoldSim model to represent the distance from the bottom of the tanks to the water table. With respect to the saturated zone, the length of the elements in the PORFLOW model were 15 m (50 ft), while 50 cells in total were used to simulate the saturated zone in the GoldSim model (10 cells were used to represent the tank farm footprint and an additional 40 cells were used to represent the remainder of the distance to the 100 m (330 ft) compliance point). Table 4-12 summarizes the range of cell thicknesses in the PORFLOW and GoldSim models graphically illustrates the conceptualization of the GoldSim model.

4.2.18.3 Probabilistic Model Benchmarking

As discussed above, the purpose of the benchmarking process was to align the PORFLOW and GoldSim models to facilitate comparisons between the deterministic and probabilistic modeling results. Two modeling outputs from PORFLOW and GoldSim were used as metrics by which to gauge the goodness of agreement between the two independent models (Taylor, 2009) and to better align GoldSim to PORFLOW:

1. Contaminant flux to the saturated zone

2. Contaminant concentration or dose at the 100 m (330 ft) point of assessment

Although the benchmarking process is similar to a calibration or optimization process, it should not be confused with calibration because data are not used to optimize model parameters. Rather, only predictive model results are used to make adjustments to the models. An underlying assumption in the benchmarking process, therefore, is that the more complex PORFLOW model adequately represents the FTF system such that adjustments should only be applied to the simpler, more conservative GoldSim model to produce comparable results. NRC staff performs a separate evaluation of the adequacy of the PORFLOW models in Sections 4.2.9 and 4.2.11. This is not to imply that no adjustments were made to the PORFLOW model during benchmarking. In fact, GoldSim results apparently informed PORFLOW and some changes were made to the PORFLOW models during PA development; however, these changes are not explicitly discussed in the PA and benchmarking factors are only discussed in the context of changes to the GoldSim models.

While flux comparisons revealed differences between the PORFLOW and GoldSim representations of the real system, no adjustments were made to the GoldSim model based upon these differences. Some notable differences between the models include the following:

1. GoldSim considers the mass of all isotopes of the same element (that are simulated) when determining whether solubility limits are exceeded. Because each decay chain is run separately in PORFLOW, solubility is not strictly evaluated based on elemental

mass. Rather isotopic mass is considered. Thus, PORFLOW results may show slightly higher peak fluxes compared to GoldSim results for constituents such as U, Th, and Pu.

2. GoldSim does not account for complex in-tank hydraulics and specifically does not account for attenuation in the grout above the contaminated zone. The PORFLOW vadose zone model considers upward mobility of constituents from the contaminated zone into the tank grout, allowing for sequestering of contaminants in the tank system for longer periods of time.

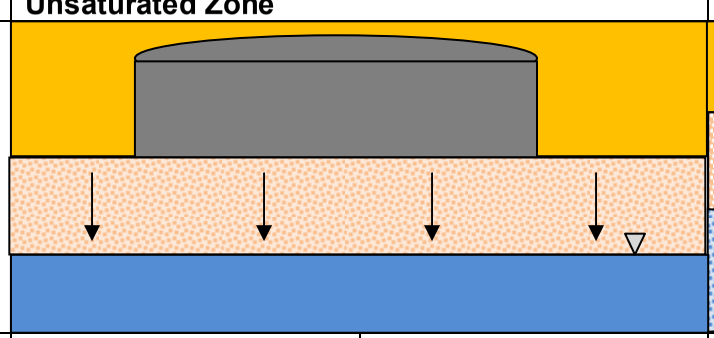
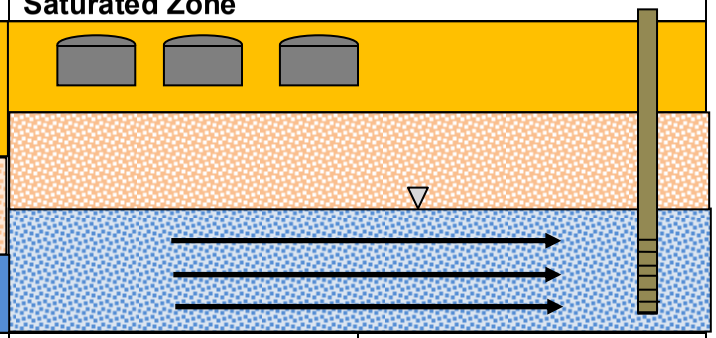
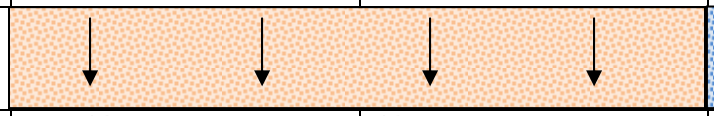
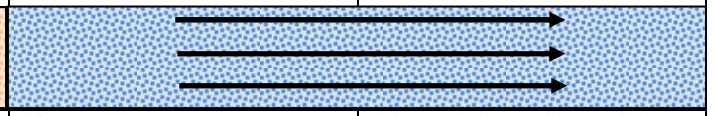
3. GoldSim accounts for the basemat fraction in Configuration D as a portion of flow that bypasses the attenuating properties of the basemat. The PORFLOW vadose zone model explicitly simulates a portion of the basemat with distinctly different material properties. Differences in PORFLOW peak fluxes tend to be more pronounced for certain constituents such as Pu, with large basemat K_ds.

Table 4-11 Summary of Configurations Evaluated in Probabilistic Analysis

	A	B	C	D	E	F	G
Description (Successive Degradation)	Base case	Configuration A with rapid cement degradation at 500 years	Configuration B with fast flow through tank	Configuration C with fast flow through tank and basemat	Configuration B with water table above contaminated zone	Configuration B with soils only cover	Configuration D with faster times to chemical transition
Tank Grout-- Hydraulic Failure	Variable by tank type (13,000 yrs [Type I] to 64,000 yrs [Type IV])	Completely degraded 500 years	Completely degraded 500 years	Completely degraded 500 years	Completely degraded 500 years	Completely degraded 500 years	As in base case
Cement or CZ-- Chemical Failure	Variable by tank type and radionuclide	Similar to base case for all but Type IV tanks, which is faster due to earlier grout degradation	Earlier failure then base case due to earlier steel liner failure	Earlier failure then base case due to earlier steel liner failure	Earlier failure then base case due to earlier steel liner failure	Similar to base case for all but Type IV tanks, which is faster due to earlier grout degradation	Chemical transitions based on base case pore volumes but occur rapidly through CZ.
Steel Liner Failure	Variable by tank type (13,000 yrs [median, Type I and III] (3600 yrs [median, Type IV])	Same as base case	More rapid failure then base case	More rapid Failure then base case	More rapid Failure then base case	Same as base case	More rapid failure than C-E for Type I, III/IIIA (500 yrs) and same time as C-E for Type IV (75 yrs)
Probability Type I, III/IIIA, and IV	0.5825 0.60 0.54	0.28 0.30 0.26	0.05 0.025 0.0125	0.025 0.0125 0.025	0.05 0.05 0.15	0.0125 0.0125 0.0125	* 0.0 0.0 0.0

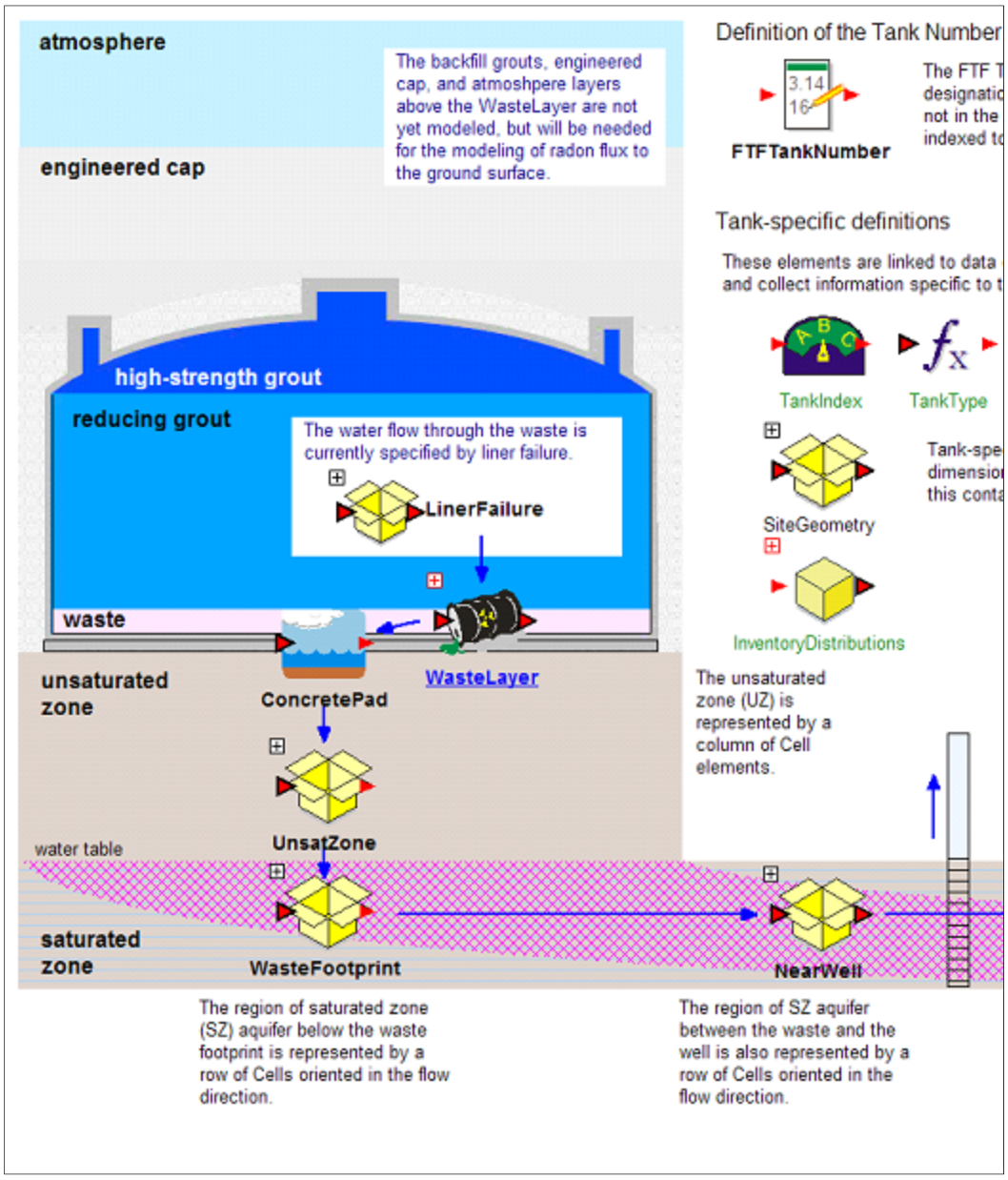
*Additional configuration evaluated in RAI response (DOE, 2011) and not considered in the FTF PA (SRS-REG-2007-00002, Rev. 1).
Note: The red text in the table above indicates the difference in the level of degradation beyond a previous configuration.

Table 4-12 Summary Information on PORFLOW and GoldSim Models

	Unsaturated Zone		Saturated Zone	
				
	PORFLOW	**GOLDSIM**	**PORFLOW**	**GOLDSIM**
Hydrogeologic Unit Being Described				
Number of Cells or Elements Along Primary Flow Direction	4 to 12	10	Up to 60	10 + 40 (or 50) = 50 (or 60) tank footprint + distance from tank boundary to compliance well*
Length of Primary Flow Path	0.5 to 6 m	0.4 to 6 m	104 to 310 m	104 to 310 m#
Cell or Element Length Along Primary Flow Direction	0.1 to 0.5 m%	0.04 to 0.6 m	15 m	2.6 to 6.2 m
Tank/Type Numbers Cell Dimensions Reported Above Pertain Too	Type IV to Type IIIA tank	Tank 20 to Tank 46	All Tank Types	Tank 45 to Tank 8

Multiply by 3.28 to convert meters to feet
*Forty cells are used for eastern and fifty cells used for western tanks.
#The distance reported is from the edge of the tank to the 100 m (330 ft) well.
%Note the vertical discretization for Type IV tanks in the saturated zone model is about 2.4 m (7.87 ft) (compared to 0.10 m [0.33 ft] in the near-field model).

Figure 4-11 GoldSim Conceptual Model (Probabilistic Assessment)
(Image Credit: SRR-REG-2007-0002, Rev. 1, Figure 4.4-42)



With respect to the second metric (i.e., concentrations at the point of assessment), DOE ultimately applied two benchmarking factors to the GoldSim model to reduce contaminant concentrations at the 100 m (330 ft) boundary (see Table 4-13). These factors are (i) a plume correction and benchmarking factor and (ii) a clayey fraction factor to account for clayey material in the saturated zone. Both factors served to lower the GoldSim concentrations to more closely match FTF/PORFLOW saturated zone model results. The plume correction factor accounts for transverse horizontal dispersion at the 100 m (330 ft) boundary by assigning a fraction of contaminant flux to each of the 100 m (330 ft) points of assessment. DOE indicates that a benchmarking factor is applied to the plume correction factor to better reflect flow data

inherent in the GSA database. NRC received clarification in RAI-FF-6 response that the benchmarking factor was used to account for differences primarily in the transverse vertical direction (DOE, 2011). The clayey fraction was used to account for differences between the two codes in timing and magnitude of peak dose due to the presence of clay in the FTF/PORFLOW saturated zone model.

NRC also received clarification in the RAI responses (DOE, 2011) that the GoldSim model simulated flow and transport in the UTR aquifer generally without regard to what aquifer zone the streamlines traversed and that benchmarking was performed using concentrations in the UTR-LZ rather than in the UTR-UZ. It was further explained in RAI response that only the GCU was modeled as having the properties of clay in the FTF/PORFLOW saturated zone model and that the GCU affected transport in the UTR. The RAI response further clarified that the clayey fraction was only assigned to the western sources and that the clayey fractions reported in the PA were not the final values used during benchmarking. NRC expressed two primary concerns with the benchmarking process in its RAIs (Camper, 2010): (i) that the benchmarking process needed to adjust GoldSim to more closely match PORFLOW modeling results might be an indicator of larger systematic deficiencies in the PORLFOW model, and (ii) that the benchmarking process would artificially bias the GoldSim modeling results low. DOE resolved many of NRC staff's questions in the RAI-FF-5 response (DOE, 2011), but new questions emerged.

Table 4-13 Final Benchmarking Factors

Benchmarking Factor	Benchmarking Factor Value	Area of Applicability
Clayey Soil	0.0*	Eastern Region
	0.25	Western Region
Final benchmarking factor	0.35	Type I tanks
	0.30	Type III/IIIA/IV tanks

*Note the value of 0.13 listed in DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) for the clayey fraction (eastern region) was stated to be incorrect in RAI-FF-5 response (DOE, 2011)

4.2.18.4 Probabilistic Sensitivity Analysis and Results

To identify those most risk-significant parameters and processes affecting FTF PA results, DOE conducted a sensitivity analysis using modeling results from the probabilistic assessment. The results of the model output were analyzed using Gradient Boosting Models (GBM). GBM partition the PA results into smaller and smaller parts to find relationships between model inputs and outputs. Thus, the GBM approach attempts to both identify those parameters and processes having the most impact on the results and the ranges over which the impacts are strongest. Of course, model sensitivities are dependent on the model output being evaluated which would be outputs such as the peak dose or peak groundwater concentrations within the compliance period or beyond. Because different radionuclides dominate the dose at different times and have their own unique sensitivities, the selection of time period would have a significant effect on the results of the analysis.

Once a GBM model is constructed, each of the model inputs can be assigned a sensitivity index that can be interpreted as the percentage of variability explained in the model by a given model input. The sum of sensitivity indices will approximately equal the R^2 of the linear regression of the model output versus the GBM predictions. The R^2 values for the FTF model are typically high, indicating the high degree of predictive power of the GBM in fitting the GoldSim model.

Two GoldSim models were run for the sensitivity analysis, one for Configuration A and one for Configuration D. Both were run for a 20,000 year time period with 5,000 model realizations. DOE provides partial dependence plots to show the relationship between the model outputs and inputs. These plots show the input parameter distributions in the background and model outputs plotted against the model input values. Because a parameter may only be influential to the model output over part of its range, only that portion of the model output range that the parameter is influential is depicted. These figures are useful in illustrating important model sensitivities. Tabular results are also presented that indicate the most important parameters affecting certain outputs of interest (i.e., peak dose within certain time periods and peak concentrations of certain radionuclides). See Section 5.6.6 in DOE's Rev. 1 PA for additional details (SRS-REG-2007-00002, Rev. 1).

Table 4-14 Summary of DOE's Sensitivity Analysis Results

Model Input Parameter	End Point of Interest
Case A	
Aquifer Well Completion Type IV Steel Liner Failure (Configurations A and B)	Peak Dose Within 10,000 years
Aquifer Well Completion Pu K_ds in Clay Pu K_ds in Soil	Peak Dose Within 20,000 years
Case D	
Time to Final Chemical Transition (generally to higher solubility) Aquifer Well Completion Pu solubility (Oxidized Region II)	Peak Dose Within 10,000 years
Aquifer Well Completion Tc Solubility (Oxidized Region II) Type I steel liner failure (Configurations C, D, and E)	Peak Dose Within 20,000 years
Input parameters are listed in order or rank (highest rank is listed first). Sensitivity indexes (SIs) are not provided. However, Model Input Parameters in red identify those SIs that account for a significant portion of the variability in the model response (SI>0.20). No input parameters listed above accounted for less than 0.05 percent of the endpoint variability.	

Sensitivity analysis results show the importance of steel liner failure times, Tc and Pu solubility (or chemical transition times), Pu K_ds and the exposure aquifer as the most important parameters impacting the peak dose within 20,000 years. These results indicate that (i) steel liner failure times, (ii) key radionuclide solubility (Pu and Tc)[9], and (iii) key radionuclide aquifer sorption (Pu) are most important to DOE's compliance demonstration. For what is deemed by NRC staff to be of lesser but well-defined risk-significance compared to other parameters, the

[9] Key radionuclide solubility is important as evidenced indirectly by (i) the relative importance of the timing of transition to higher solubility input parameters or directly by (ii) the relative importance of Pu and Tc solubility input parameters with respect to the magnitude of the peak dose.

assumed aquifer that the well is located in is also of importance to the compliance demonstration. It is important to note that NRC evaluates compliance with protection of the general population under 10 CFR 61.41 at the point of maximum exposure at or beyond a 100 m (330 ft) buffer zone no matter where that exposure takes place or the probability of that exposure. Therefore, while DOE evaluates aquifer well completion as an uncertain parameter in its probabilistic assessment, compliance with the performance objectives should be evaluated at the point of maximum exposure beyond the 100 m (330 ft) buffer zone for any viable aquifer/unit. However, the impact of treating well completion as an uncertain parameter is modest—accounting for only about a 40 percent decrease in the peak dose. It is important to note, however, that the effect can be quite pronounced when considering certain statistics associated with the probabilistic analysis. For example, the median dose in DOE PA Figure 5.6-38 (SRS-REG-2007-00002, Rev. 1) is dominated by the forty-percent of the realizations that are a factor of 20 times less than what the dose would have been for the point of maximum exposure in the UTR. Additionally, when viewing peak dose versus time of peak dose on a figure such as Figure CC-UA-3.9 (DOE, 2011) also color coded and illustrated in Figure 4-12 in this TER, the differences in results for the UTR and Gordon aquifers are striking (i.e., two distinct data sets consistent in timing but dissimilar in magnitude are discernible in this figure).

4.2.18.5 Deterministic Sensitivity Analysis and Results

In addition to the probabilistic assessment, DOE also performs deterministic sensitivity analysis runs. The following sensitivity analyses were conducted.

Table 4-15 Summary of DOE's Deterministic Sensitivity Analysis Results

Sensitivity Analysis	**Tanks or Materials**	**Results**
Inventory 1.5X base case 0.5X base case	Tank 5 Tank 6 Tank 18 Tank 19 C-14 Cs-135 I-129 Np-237 Pa-231 Pu-239 Pu-240 Ra-226 Tc-99 Th-229 U-233 U-234	Dose corresponds linearly to inventory for many radionuclides that are not solubility limited or whose mass is low enough not be solubility controlled (e.g., C-14, Cs-135, I-129, Pa-231, Ra-226, Th-229, U-233, U-234). For those radionuclides that are solubility limited either there is no impact of inventory on dose (e.g., Tc-99, Pu-240) or there is a mass depletion effect on doses that results in doses higher or lower than the base case.

K_d
If K_d>1000 L/kg, then
5X or 0.20X base case
If K_d<1000 L/kg, then
2X or 0.5X base case
0.5X base case
	<u>Tanks</u>
	Tank 5
	Tank 34
	Tank 18
	
	<u>Materials</u>
	Basemat
	Soil
	
	Pu-239
	Tc-99
		The results of the K_d sensitivity runs show that the Tc-99 flux is relatively unaffected by K_d changes, while the Pu-239 flux can be significantly influenced especially for Type I and III tanks with thicker basemats.

4.2.18.6 Barrier Analysis

As stated in Section 5.6.7.3 of the FTF PA, the purpose of the barrier analyses was to use PORFLOW to identify barriers to waste migration and to evaluate the capabilities of each barrier. The barrier analyses assessed the following barriers:

1. Engineered Closure Cap
2. Grout (hydraulic and chemical as it impacts CZ [no change in grout K_ds])
3. Contaminated zone (chemical)
4. Tank liner (hydraulic)
5. Waste tank concrete (hydraulic and chemical [basemat only])

There are several limitations with the barrier analysis including the following:

1. Natural system performance is not evaluated.
2. Intruder barriers are not specifically evaluated.
3. Inconsistencies with respect to barrier performance exist within a single case (e.g., contaminated zone and tank grout barriers affect solubility of constituents in the waste zone but were not always consistent, leading to ambiguities).
4. Barriers are lumped together making it difficult to evaluate the impact of an individual barrier (e.g., vault walls and roof were lumped in with the basemat, while the basemat primarily affects transport of constituents out of the tanks).
5. Results are presented in terms of changes to flux over 10,000 years. This presents two problems: (i) the significance of the barrier to dose is unclear as fluxes (not doses) are provided and large changes in fluxes can reflect changes at very low doses that are inconsequential, and (ii) it is difficult to determine barriers to timing versus barriers to magnitude. Had the absolute peak flux (or dose) over time been reported, information on impact of magnitude could have been gleaned from the analysis, which is currently not possible. Had results for both the absolute peak flux (or dose) and the peak flux (or

dose) within 10,000 and 20,000 years been reported, information on impact of the barrier on timing could also have been gleaned from the analysis.

A description of the barrier analysis cases is provided in Table 5.6-22 (page 697) and 5.6-23 (page 698) of DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1). The barrier analysis supported sensitivity analysis conclusions including the importance of the steel liner, chemical barriers in the grout and contaminated zone, as well as the cover in limited cases. The barrier analysis results also indicated that Np-237 in particular was at risk of being over the compliance limits within a 10,000 year performance period if barriers did not perform as well as assumed in DOE's PA (SRS-REG-2007-00002, Rev. 1).

The following specific observations were noted by NRC staff:

1. Type IV tanks degrade early enough to get doses in 10,000 years, so they are less sensitive to the degraded conditions simulated. Early chemical barrier failure in the CZ is generally needed to realize higher doses within 10,000 years.

2. The engineered cover can be important in certain cases with other failures.

3. In limited cases, the basemat seems to have very large impact on lowering dose with other failures occurring (e.g., Type I tanks and Tc-99).

4. The steel liner can compensate for other barrier failures attributes when other barriers are failed.

5. Early liner failure can lead to significantly higher doses especially for Type I and III/IIIA tanks. In limited cases, radionuclide doses can go down (e.g., Ra-226 due to more limited time for in-growth).

6. Tank grout can be a significant, independent barrier when other barriers have failed.

7. Faster times to chemical failure (as would occur in a Condition 2 or Configuration G scenario) can have a large impact on dose.

8. Faster times to transition to Oxidized Region III can have a significant impact on dose.

4.2.18.7 Probabilistic Uncertainty Analysis and Results

DOE performed a probabilistic uncertainty analysis using the GoldSim modeling platform as described above. DOE ran several probabilistic models including the following listed below:

Table 4-16 List of DOE's Probabilistic Modeling Simulations

Description	Details	File Name
Configuration A—Base case	5000 realizations	SRS FTF v 2.4 20ky Case A r5000 s1
Configuration D—Preferential Pathway Case	5000 realizations	SRS FTF v2.4 20ky Case D r5000 s1
All Configurations (weighted by probabilities listed in Table	1000 realizations	SRS FTF v2.4 100k All Cases r1000s1

4-11)		
Configuration E	500 realizations	SRS FTF v 2.4 20ky Case E r500
Configuration F	500 realizations	SRS FTF v 2.4 20ky Case F r500

Sections 4.4.4.2, 5.6.1, and 5.6.3-5.6.5 in DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) provide additional details regarding DOE's probabilistic assessment. Figure 5.6-38 in DOE's PA provides statistics for the All Cases probabilistic model. Tables 5.6-10 and 5.6-11 in DOE's PA provides information on the mean of the peak and mean of the peaks as summarized in Table 4-3 in this TER. Additional information was also provided by DOE in response to NRC's RAI-UA-4 and CC-UA-3 with respect to Configuration E and F results, as well as additional information regarding the peak dose and timing of the peak dose. It is significant to note, that the average peak dose (or mean of the peaks) does not appear to vary significantly between Configurations in DOE's probabilistic analysis, although changes to the timing of the peak dose is more significant between Configurations. Perhaps, this is due in part to the peak dose being dominated by Tc and Pu in a large fraction of the realizations and Tc eventually being released with no solubility control and Pu being released at a risk-significant solubility upon transition to oxidized Region III. While Tc can be depleted prior to transition to no solubility control and this result happens in roughly 50 percent of the realizations, owing to a higher assumed solubility limit in Oxidized Region II for an unspecified phase (see Table 4.2-14 in DOE's Rev. 1 PA [SRS-REG-2007-00002, Rev. 1]), this phenomenon happens in all Configurations (i.e; not configuration dependent) apparently resulting in roughly half of the peak doses being attributable to Tc and half attributable to Pu. Similarly, trends in the magnitude of the Pu peak dose are mainly attributable to changes in the solubility limiting phases and natural system K_ds; however, these parameters are also not Configuration dependent leading to variability in the timing and magnitude of the peak dose irrespective of Configuration.

On the other hand, significant differences do exist between the timing of steel liner failure between two sets of Configurations. Configurations A, B, and F are lumped together with relatively longer times to steel liner failure, while configurations C, D, and E are lumped together with relatively shorter times to steel liner failure. Thus, there is some variability due to the timing of steel liner failure. However, the other key parameter distributions affecting timing of peak dose, the chemical barrier and chemical transition times, are tightly constrained. Because the relative contribution of this barrier to the delay in the timing of peak dose is larger than that of the steel liner, the full effect of uncertainty in engineered barrier longevity may not be fully propagated in the PA model. From a risk dilution perspective, the narrow parameter distributions surrounding expected or central values makes it relatively easy to see trends in the results. At the same time, peak doses rarely occur within the time frames of interest in DOE's PA results.

As discussed above, another important feature observed in Figure 4-12 provided by DOE following the June 28, 2011 RAI resolution meeting and discussed at the July 19, 2011 RAI resolution meeting (see Figure 1a in the attachment to the meeting summary found at ADAMS accession number ML112200656), is the appearance of two distinctly different data sets that are similar in shape and spread but only varying by roughly an order of magnitude. This result is due to a parameter assignment related to well completion. In the case of a well completed in the Gordon aquifer, DOE assumes a simple ratio between the UTR concentration and dose actually being simulated in the GoldSim and the concentration and dose in the Gordon aquifer.

The Gordon aquifer concentration and dose is not actually being calculated by GoldSim, but rather, is assumed to be a factor of 20 less than the UTR concentration. The probability of well completion in the Gordon aquifer is 40 percent. Thus, the peak doses for individual realizations depicted in Figure 4-12 occur in two distinct data sets separated by roughly a factor of 20 with realizations generally not overlapping between the two data sets over the same time intervals. Over all time, the two data sets overlap with uncertainty in the magnitude of the peak dose spanning roughly two orders of magnitude.

4.2.19 NRC Evaluation of Uncertainty/Sensitivity Analysis

With respect to the level of analysis performed by DOE to study the importance of key barriers and parameters in DOE's PA model, NRC staff thinks DOE's efforts in this area are commendable. DOE performs deterministic and probabilistic modeling to evaluate sensitivity and uncertainty in its deterministic and probabilistic assessments, and provides a detailed barrier analysis to study the impact of various combinations of engineered barrier failure or the importance of certain barriers as independent barriers. The information provided by DOE in its PA and RAI responses is exceptional and greatly assists NRC with gaining a better understanding of overall system performance to help focus its review.

With respect to the implementation of DOE's probabilistic model, NRC notes a few concerns in the paragraphs that follow. DOE has chosen to use a hybrid modeling approach (using deterministic and probabilistic analysis) to evaluate compliance with the performance objectives in 10 CFR Part 61, Subpart C with the compliance demonstration heavily weighted towards the base case configuration or Configuration A as explained in more detail below. NRC staff expressed concern with the level of support for key modeling assumptions in DOE's base case Configuration A that NRC staff indicated in a number of RAIs (Camper, 2010) could be overly optimistic. Much of the discussion in the preceding sections provides additional details regarding NRC's specific concerns. As shown in Table 4-3 and Figure 4-12, the exceedance of the performance objective by one or two orders of magnitude at some point in the future seems fairly certain in DOE's deterministic and probabilistic analyses, although the time of the peak dose seems fairly uncertain spanning from 20,000 to 60,000 years in DOE's probabilistic analysis (see Figure 4-12). NRC staff's primary concern is that the timing of peak dose could be shifted into the period of performance and that the magnitude of the peak dose is not clearly communicated in DOE's PA.

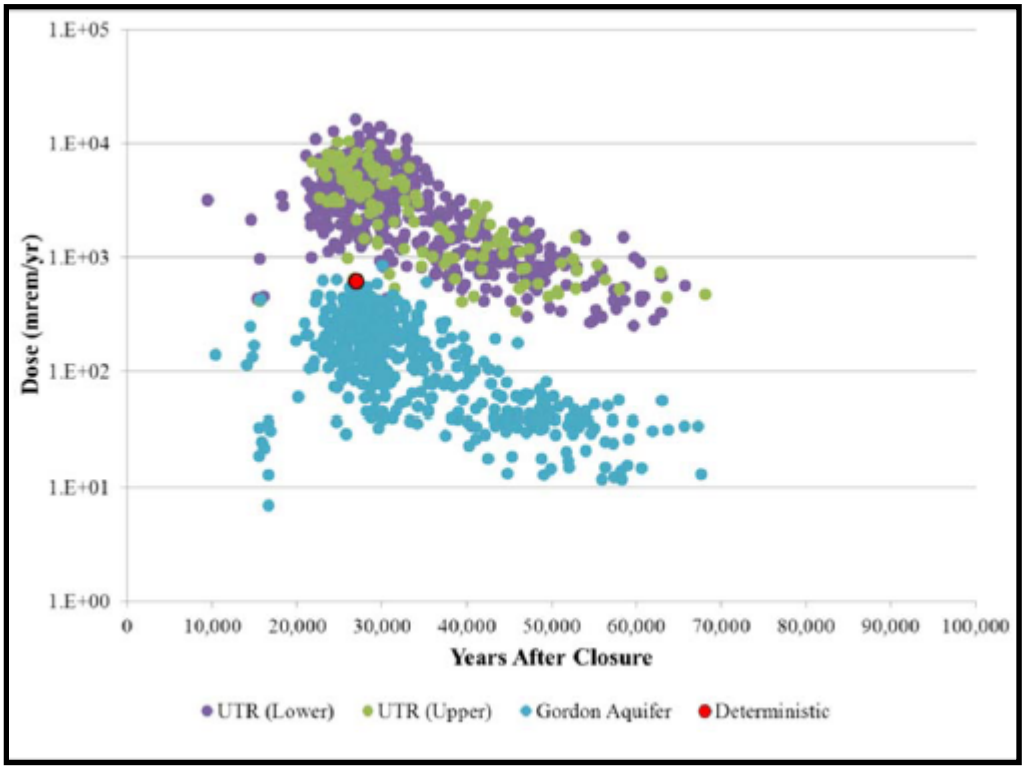


Figure 4-12 Probabilistic Modeling Results (Peak Dose vs. Time, see July 19, 2011 RAI resolution meeting summary, ADAMS accession number ML112200656)

4.2.19.1 NRC Evaluation of DOE's Treatment of Uncertainty Related to Timing of Peak Dose

Most of the realizations with peak doses dominated by Tc fall narrowly around 26,000 years (the time to failure of Type I tanks that contain the largest inventories of Tc in the FTF in DOE's base case). The spread in timing of the peak dose attributable to Pu is wider, spanning tens of thousands of years owing to the uncertainty surrounding the K_ds values assigned to Pu in DOE's probabilistic assessment. As communicated in a number of NRC RAIs, staff's largest concern with DOE's PA rests with the lack of support for key modeling assumptions and barriers namely (i) steel liner performance and (ii) chemical barriers that serve to delay the timing of peak dose. In the simplest of terms, if the performance of engineered barriers that collectively serve to delay the timing of peak dose (represented by steel liner failure times and chemical transition times) were off by a factor of roughly 3 to 4, then the peak dose associated with Tc or Pu could occur within the period of compliance of 10,000 years. NRC thinks that an uncertainty of this magnitude should not be described as "highly unlikely" or "incredible" and a meaningful discussion regarding the peak doses in DOE's probabilistic analysis should be communicated in the PA and basis document.

Although the probabilistic analysis considers faster times to steel liner failure and faster transition times, the collective uncertainty in these two parameter values is constrained. We will first discuss the treatment of uncertainty in steel liner failure times. NRC staff thinks that the steel liner failure times are overly optimistic with longer times to failure represented by the base case Configuration A, or slight variants of this case, heavily weighted in the compliance demonstration. While faster steel liner failure times can be realized in the probabilistic analysis (Configurations C, D, and E), these scenarios account for approximately 18 percent or less of

the realizations in DOE's probabilistic assessment depending on tank type. Thus, the faster steel liner failure times are swamped by Configurations A, B, and F that have relatively longer steel liner failure times (e.g., 99.5 percent of the realizations for Configurations A, B, and F have failure times beyond 10,000 years for Type I and III/IIIA tanks).

Additionally, NRC staff thinks that the level of uncertainty in the performance of the chemical barrier transitions times may be understated. DOE uses factors of 30 and 50 percent to represent uncertainty in the E_h and pH chemical transition times, respectively. As stated above, due to the lack of support for the geochemical modeling used to predict chemical transitions times and uncertainty in the conceptual model used to predict times to transition to higher solubility limiting phases (see Section 4.2.9 above), NRC staff thinks that the uncertainty in chemical transitions is likely under-stated. DOE indicates that the values of 30 and 50 percent were selected so as not to swamp the results of the probabilistic analysis; no clear and transparent approach is used to develop the parameter distributions and apparently only partial consideration was given to small set of factors that might affect this key modeling assumption (RAI-NF-10, [DOE, 2011]). The parameter distributions appear to be overly constrained. The end result of potentially (1) overly optimistic steel liner failure times, (2) prolonged chemical transition times, and (3) biases in or overly constrained parameter distributions accounting for uncertainties in liner and chemical barrier failure times is peak doses that occur much later in time (e.g., 99.9 percent of the realizations in DOE's All Cases probabilistic model have peak doses that occur beyond 10,000 years and 97.5 percent after 20,000 years). Thus, although DOE provides detailed analysis of the doses that occur within 10,000 and sometimes 20,000 years, considerably less attention it paid on the peak doses that almost always occur without exception later in time.

4.2.19.2 NRC Evaluation of DOE's Hybrid Modeling Approach

With regard to the hybrid modeling approach, NRC staff thinks that additional information needs to be provided by DOE regarding why the base case is the best estimate or most likely scenario. This information would ideally be in the form of model support for key modeling assumptions and parameters associated with the base case. Without support for the base case or Configuration A, it would be difficult to argue the relative likelihood of one alternative scenario over another. In the face of large uncertainty with respect to the most likely configuration, compliance scenarios should be conservatively selected, in that they err on the side of higher doses or higher risk with respect to the compliance demonstration. Furthermore, it is not clear that all plausible, high risk configurations were considered in the Rev. 1 PA. For example, RAI-NF-15 requested additional information regarding one high risk scenario agreed to in FTF scoping that was not presented in DOE's Rev. 1 PA. Condition 2 (as described on page 264 of DOE's PA, SRS-REG-2007-00002, Rev. 1) was described in the Rev. 0 PA but results were actually presented for a different scenario with significantly lower risk within the 10,000 year compliance period. In a related RAI (RAI-PA-1), NRC requested DOE evaluate the appropriateness of the compliance case in the face of great uncertainty. In response to RAI-NF-15 and RAI-PA-1, DOE modeled an additional configuration named Configuration or Case G. The results of this scenario, which are described as bounding or highly unlikely by DOE, indicates that doses of 1.25 mSv/yr (125 mrem/yr) could be achieved within 10,000 years and doses in excess of 5 mSv/yr (500 mrem/yr) could be achieved in 20,000 years. While DOE considers Configuration G an unlikely or bounding scenario, it is important for DOE to provide additional support for key assumptions in its compliance case to provide a firmer basis for its assessment that base case results provide a best-estimate of future predicted dose.

NRC indicated in an RAI resolution meeting that as communicated in the cover letter to the RAIs, that it was not its expectation that DOE would have sufficient time to fully address all of NRC's concerns prior to completion of NRC's TER. NRC attempts to identify data gaps in DOE's PA in this TER, and has indicated areas where additional support is needed for key modeling assumptions, as well as information (recommendations) regarding how NRC thinks DOE could get the additional support. NRC continues to indicate its intent to assess compliance during the monitoring period using a graded approach. NRC thinks that sufficient time is available to obtain additional support for key modeling assumptions given that DOE is in the early stages of FTF closure.

4.2.19.3 NRC Evaluation of Benchmarking

DOE's RAI-FF-6 response provides a well-thought out and well-explained process by which the PORFLOW and GoldSim models could be benchmarked and aligned to produce comparable results (DOE, 2011) greatly assisting NRC with understanding and documenting the benchmarking process in this TER. However, it is NRC staff's understanding that the benchmarking process described in the PA contained inaccuracies with respect to what concentrations were being benchmarked, what aquifer zones were being simulated in the GoldSim model, and other benchmarking details. For example, NRC staff understands DOE to indicate that the GoldSim transport model solves for radionuclide transport not only exclusively within the UTR-UZ, but also through the TCCZ and into the UTR-LZ, contrary to information provided in DOE's Rev. 1 PA (SRS-REG-2007-00002, Rev. 1) with concentrations in the UTR-LZ being used for benchmarking.

NRC staff expressed concerns with the use of the clay fraction in the benchmarking process. DOE explains that the physical basis for the clay fraction used in the GoldSim model is to account for the influences of the GCU located at the base of the UTR aquifer in the local FTF/PORFLOW model. Although DOE provides an explanation that the GGU affects flow and transport in the UTR aquifer, it is not clear why clay located below the UTR-LZ would significantly affect travel times of the plume mass that remains in the UTR-LZ, although flow and contaminant transport into the GCU itself would be expected to slightly affect the magnitude of mass reaching wells in the UTR. Additionally, the fact that a clay fraction is assigned to the western portion of the FTF facility, which has tank sources that would physically encounter less interaction with the GCU (owing to the relatively higher position of their plumes in the UTR aquifer), while eastern tank sources (that are expected to have flow paths deeper in the aquifer with longer travel distances to the point of exposure) are assigned no clay fraction, could not be explained by DOE.

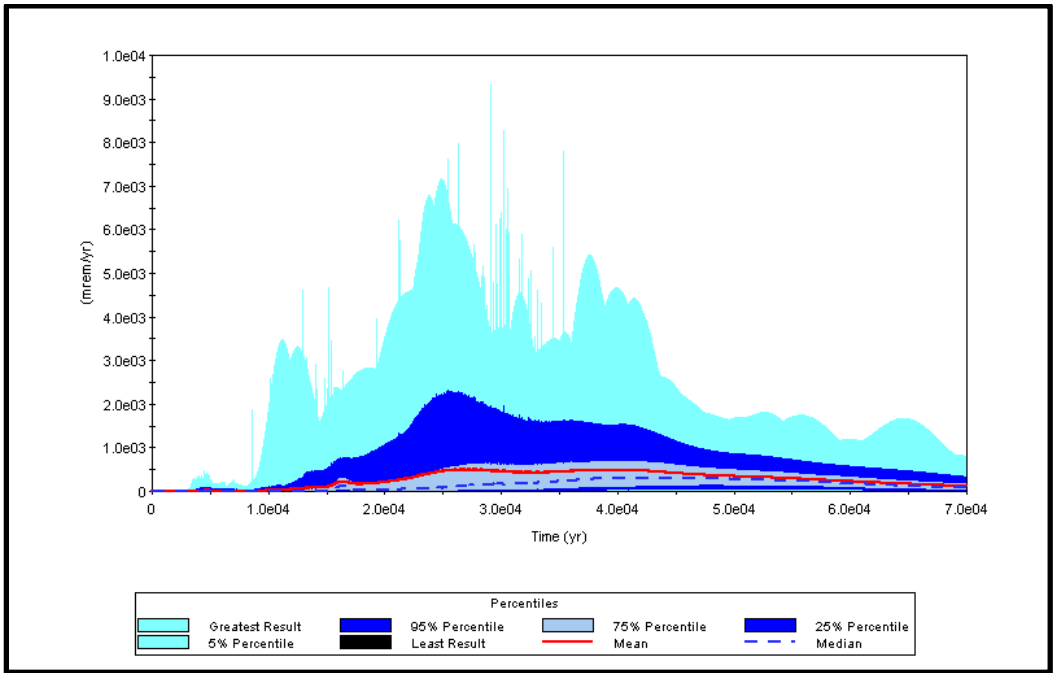
In response to RAI-FF-6, DOE indicates that a value of 1.0×10^{-20} for the porosity of the GoldSim saturated zone sand was erroneously applied to the UTR aquifer (DOE, 2011). DOE indicates the impact of this assumption leads to faster travel times for nonsorbing constituents and modest impacts for sorbing constituents. Given the fact that such a low porosity would lead to very fast transport times, it is entirely plausible that benchmarking factors were needlessly applied simply to compensate for this error. Given the fact that doses are dominated by Tc-99 and Pu-239 from the eastern and western portions of the site, the clay fraction may have been needed to align Pu-239 doses (to compensate for rapid transport to the well) while the relative impact on Tc-99 transport was low given the fact that Tc-99 transport to a well is rapid (less than one time step or 10 years in the Goldsim model) with or without the error in porosity. Nonetheless, corrections in this parameter assignment seem to have modest impact on modeling results.

DOE acknowledged that the flow model abstractions from PORFLOW could have been more accurate eliminating the need for so many benchmarking adjustments that to NRC staff seemed in some cases to be at odds with one another (see July 21, 2011, RAI resolution meeting summary, ADAMS accession number ML112070958). For example, it is not clear why adjustments to the Darcy velocity were used when they seemed to bring the peak times further out of alignment with PORFLOW and required an adjustment to the magnitude of the peaks, and why the number of GoldSim cells in the horizontal direction was increased to make the peak sharper and occur later in time, which is at odds with a Darcy velocity that attempts to speed up transport in the saturated zone at a lower concentration. NRC and DOE agree that improvements to the benchmarking process can be made in the future leading to a more streamlined and elegant process. The same methodological approach used to describe the benchmarking process and to indicate the reason for and effect of each adjustment in the RAI response is recommended in future benchmarking efforts and documentation (see RAI-FF-6 response, [DOE, 2011]).

4.2.19.4 NRC Evaluation of Probabilistic Uncertainty Analysis Results

NRC staff also noted potential inaccuracies in the statistical information being presented in the PA due to the selection of a 1,000 year averaging time for plots in the probabilistic assessment. Furthermore, compliance with the performance objective in 10 CFR 61.41 should be against only those realizations with a well assumed to be completed in the UTR (and not the Gordon aquifer) as discussed in the July 21, 2011 RAI resolution meeting (ADAMS accession number ML112070958). Figure 5.6-38 in the PA (SRS-REG-2007-00002, Rev. 1) was replaced with another figure that considered (i) only those wells completed in the UTR, and (ii) used a plot time of 10 (rather than 1000 years). Figure 4-13 below shows NRC's attempt to provide more appropriate figure to assess compliance as an alternative to the one presented in DOE's PA Figure 5.6-38 (page 625 of SRS-REG-2007-00002, Rev. 1). The results shown in Figure 4-13 indicate a peak of the mean dose of around 0.20 mSv/yr (20 mrem/yr) over a 10,000 year period of performance and an overall peak of the mean dose of 5 mSv/yr (or 500 mrem/yr).

Figure 4-123 NRC Staff Alternative Presentation of DOE's Probabilistic Modeling Results



4.2.19.5 NRC Evaluation of Model Abstraction Uncertainty and Other DOE PA Conservatisms

Although NRC has significant concerns with respect to the modeling treatment of engineered barrier performance, NRC staff acknowledges a number of modeling conservatisms that may mitigate less conservative modeling assumptions. For example, modeling simplifications may lead to over-estimates of the peak dose. In reality, the FTF steel liners may be expected to fail more gradually over time with releases from the tanks being spread out more over time. In contrast, DOE assumes complete, discrete failure of the steel liners which serves to magnify the peak dose. NRC had an RAI related to this modeling simplification, RAI-NF-16, that expressed the thought that although earlier, partial release from the tank might lower the magnitude of the peak dose, releases that occur after 10,000 years do not add to the compliance risk that considers a period of performance of 10,000 years. In other words, although early, partial release may lead to a lower overall peak dose, if less than a 10 percent fraction of the current base case peak dose was realized within the 10,000 year compliance period, the compliance standard could be exceeded using DOE's current deterministic analysis results (and less than 1 percent of the mean of the peak dose from DOE's probabilistic analysis would need to be realized within 10,000 years). Thus, early partial release may represent a scenario that carries greater compliance risk than one in which most of the tank failures (18 of 22 tanks) are assumed to occur beyond the period of performance.

Another mitigating factor limiting the magnitude of the peak dose is related to the Tc inventory assumed for Type I tanks. DOE has evidence that the inventories for Tc may have been significantly over-predicted in the Rev. 1, PA. Since the Rev. 1 PA was completed, DOE has treated Type I Tanks 5 and 6 with oxalic acid strikes and has removed additional inventory through the feed and bleed method. While final inventory estimates have not been provided, DOE thinks that the Tc inventories could have been overestimated by over two orders of magnitude. If the Tc inventories are less than 3.7×10^{10} Bq (1 Ci), consistent with DOE revised

estimates (see RAI-UA-3 response [DOE, 2011] and revised Figure CC-UA-3.9 discussed during the July 21, 2011 RAI resolution meeting, ADAMS accession number ML112070958), then the peak dose of 6 mSv/yr (600 mrem/yr) in the base case or Configuration A analysis, and up to an order of magnitude higher in the probabilistic analysis may not be realized. This would leave Pu-239 in Tank 18 as the single largest risk driver for the FTF, based on DOE's PA analysis, with doses in excess of 3 mSv/yr (300 mrem/yr) in the base case Configuration A and approximately 1000s of mrem/yr in the probabilistic analysis (see Figures 3-5 in the attachment to the July 19, 2011 meeting summary at ADAMS accession number ML112200656).

Uncertainty and sensitivity analysis recommendations should be considered in future PA updates (represent intermediate term recommendations). As execution of the probabilistic analysis may impact the compliance demonstration (if DOE chooses to use the results of the probabilistic analysis to inform the compliance demonstration), the risk-significance of recommendations in this area are apparent.

4.3 Other Performance Objectives Evaluated in the Basis Document

4.3.1 Protection of Individuals During Operations

The performance objective in 10 CFR 61.43 cross-references "the standards for radiation protection in Part 20". DOE's approach to demonstrating protection of individuals during operations (10 CFR 61.43) is to cross-walk the relevant DOE regulation or limit with that provided in 10 CFR Part 20 and demonstrate that DOE's regulation provides an equivalent level of protection. The cross-referenced "standards for radiation protection" in 10 CFR Part 20 (USNRC, 2005) that are considered in detail 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). These dose limits correspond to the dose limits in 10 CFR Part 835 and relevant DOE Orders which establish DOE regulatory and contractual requirements for DOE facilities and activities.

A number of measures will ensure that exposure of individuals during operations are maintained ALARA. These include: (1) a documented RPP, (2) a Documented Safety Analysis (DSA), (3) design of the FTF, (4) regulatory and contractual enforcement mechanisms, and (5) access controls, training, and dosimetry. These measures are described in the waste determination or basis document (DOE/SRS-WD-2010-001, Rev. 0, September 2010).

4.3.2 NRC Evaluation of Protection of Individuals During Operations

DOE has provided adequate information that individuals will be protected during operations. DOE provided a detailed cross-walk of the relevant DOE regulations to those provided in 10 CFR Part 20, which is referenced in the 10 CFR 61.43 performance objective. NRC agrees that an equivalent level of protection is provided by the relevant DOE regulations or limits to the requirements found 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). In addition, a number of measures are applied to ensure that exposure of individuals are maintained as low as reasonably achievable including: (1) a documented RPP, (2) a Documented Safety Analysis (DSA), (3) design of the FTF, (4) regulatory and contractual enforcement mechanisms, and (5) access controls, training, and dosimetry.

In general, the activities at the FTF involve inert materials and common, low-temperature, low-energy industrial operations. The public will be located a significant distance (several kilometers) from the facilities during operations and active security is maintained to prevent inadvertent access to the site. The NRC agrees with DOE that the risk to the public during operations should be minimal, and the relevant regulatory limits can be achieved.

4.3.3 Stability of the Disposal Facility Following Closure

SRS is located in a region with relatively low seismic activity. The largest known earthquake in the vicinity of the site occurred in Charleston, SC in 1886 with a magnitude of 7.3 on the Richter Scale (USGS, 2011). The FTF is located approximately 79 m (260 ft) above mean sea-level (MSL) with the lowest waste tank elevation being 69 m (227 ft) MSL. The calculated 100,000 yr flood elevations for FMB and UTR Branch are 59 and 45 m (194 and 146 ft) MSL, respectively, which are both below the minimum tank elevation of 69 m (227 ft) MSL (WSRC-TR-99-00369); (SRS-REG-2007-00002, Rev. 1).

Once all of the 22 HLW tanks and ancillary equipment are cleaned and grouted, a closure cap will be installed, and a 100 year period of institutional controls will begin. Beyond the institutional control period, DOE's "SRS End State Vision" includes ownership and control of the entire site by the federal government for perpetuity and prohibits residential use of the site (PIT-MISC-0089).

A closure cap will be placed over the grouted HLW tanks that will be designed to limit long-term erosion (WSRC-STI-2007-00184). Static loading induced settlement, seismic induced liquefaction and subsequent settlement, and seismic induced slope instability will be considered in the final design of the closure cap. The closure cap is discussed further in Section 4.2.4.

Grouting of the HLW tanks and annular spaces (for Type I and III/IIIA tanks) is designed to minimize void space and provide a monolithic structure to prevent collapse and differential settlement (T-CLC-F-00421). A stability assessment concluded that the grouted tanks would not be susceptible to cracking due to settlement or seismic loading.

Much of SRS, including FTF, is underlain by calcareous sediment in the Santee formation resulting in the presence of "soft zones" (WSRC-TR-99-4083). The stability of these soft zones has been investigated since the 1950s with the understanding and treatment of these features evolving since preliminary evaluations (WSRC-TR-99-4083) (COE, Geologic Engineering Investigations, 1952). Early approaches to these regions included grouting, however it was determined that these grouting campaigns provided limited benefit in mitigating the potential settlement from the soft zones (WSRC-TR-99-4083). More recently, advances in analytical techniques have been utilized to resolve foundation stability issues without requiring soil remediation. The stability of the soft zones beneath the FTF under closure conditions (i.e., tanks grouted and cover emplaced) has not been directly evaluated (SRR-CWDA-2011-00054). However, analyses for numerous SRS facilities have been conducted that show soft zones to be stable under static conditions and in response to a design basis earthquake. Measurements of settlement at these facilities after construction have indicated conservatism in settlement calculations (WSRC-TR-99-4083). Dynamic stability analyses for a design basis earthquake indicated that soft zones would not be susceptible to collapse (SRR-CWDA-2011-00054). Although soft zone collapse is not anticipated by DOE, a hypothetical collapse of a 2.4 m (8 ft) soft zone near the FTF was calculated to result in a surface settlement of approximately 10 cm

(4 in) (McHood, 1998). The potential for mass removal of carbonate material leading to subsidence within 100 years was dismissed due to the Santee formation being located beneath the water table, in a relatively stable geochemical environment (WSRC-TR-99-4083).

4.3.4 NRC Evaluation of Stability of the Disposal Facility Following Closure

4.3.4.1 NRC Evaluation of Site-Stability

Grouting of the HLW tanks and annular spaces should prevent tank collapse and differential settlement and the closure cap will limit erosion. DOE's stability assessment predicted that the grouted HLW tanks would not crack under seismic loading or potential settlement configurations throughout the 10,000 year compliance period. The grouted HLW tanks were modeled as a uniform monolith with a long-term grout compressive strength of 12.4 MPa (1800 psi) and did not take credit for any reinforcing steel in the vault concrete or the tank itself. Grout degradation mechanisms were determined to only affect the outermost 2.5 to 5 cm (1 to 2 in) and would have a negligible effect on overall structural integrity. The assumed long-term compressive strength of the monolith is not adequately supported and may be optimistic based on observations of vault cracks, discussed further in Section 4.2.9.1. Cracking of the vault concrete and tank grout is not expected to result in significant structural tank collapse. However, the integrity of the vault concrete and tank grout is important to steel liner performance and waste release.

DOE settlement analyses at SRS have indicated conservatism in the calculations and appear adequate on a design basis (WSRC-TR-99-4083). Although dissolution of the calcareous sediment is likely to be a very slow process, it has not been demonstrated that dissolution is insignificant with respect to site-stability throughout the performance period. Calcareous sediment in the Santee formation underlies much of SRS, resulting in a ground surface that may be in a state of unstable equilibrium (COE, Geologic Engineering Investigations, 1952). Evidence of dissolution, based on elevated bicarbonate ion concentrations and relatively high pH values for groundwater samples collected in or near the Santee formation, demonstrates an ongoing evolution (COE, Geologic Engineering Investigations, 1952) (WSRC-RP-92-450). This has created a soil structure that is characterized by arching, under-consolidation, and historic, periodic collapses. The COE (1952) identified 7 depressions believed to be sinks at F Area, with one sink located within the 100 m (330 ft) boundary of the FTF. DOE's calculations do not account for the potential removal of subsurface material which has resulted in subsidence observed at SRS and FTF. DOE should account for these processes and features or demonstrate that reasonably predicted future dissolution of calcareous sediment is insignificant to site performance.

4.3.4.2 NRC Review Results and Recommendations Related to Site-Stability

NRC staff note the following:

1. DOE has provided sufficient information for NRC to perform a preliminary review of site-stability.

2. NRC staff thinks that additional information is needed with respect to the site-stability analysis as noted below. Additional analyses can be conducted during the monitoring period.

NRC staff recommends DOE continue to evaluate the following areas during the monitoring period:

1. Closure cap settlement and stability analyses (Medium Risk-Significance, Intermediate Term) including the following:

2. Site-specific settlement analysis for FTF, that includes the increased overburden from tank grout and the closure cap.

3. Evaluation of vault and grout integrity that is consistent with observations and reasonable expectations of future degradation of cementitious materials.

4. Assessment of the potential subsidence due to ongoing dissolution of calcareous sediment in the Santee formation.

4.4 NRC Evaluation (Criterion 3)

NRC staff's review results related to Criterion 3 are as follows:

- Due to lack of transparency and traceability of DOE's PA documentation, it is difficult to determine whether features, events, and processes (FEPs) are comprehensively evaluated in DOE's PA.

- DOE's approach to developing inventories for FTF sources is reasonable and generally tends to err on the side of conservatism for tanks that have yet to be cleaned.

- Major degradation processes are evaluated in DOE's cementitious material degradation and steel liner corrosion modeling analyses; however, assumptions regarding long-term performance of cementitious materials and liners are highly uncertain and do not appear to be fully supported by site-specific evidence and other observations.

- Near-field modeling simplifications (e.g., assumed discrete failure of tank liners) may lead to overall higher peak dose but lower dose within the compliance period and, consequently, lower compliance risk than more realistic approaches to modeling liner failure.

- Basemat K_ds can be especially risk-significant for certain key radionuclides; additional experimental support could increase confidence in selected values given (1) the wide range of values found in the literature, (2) the potential mis-application of experimental data (sorption versus solubility control), and (3) to ensure K_ds representative of aged concrete are obtained.

- Although DOE's overall approach to modeling waste release is reasonable, DOE assumptions regarding solubility limiting phases, solubility limits, and chemical transition times are particularly risk-significant, and have not been confirmed through waste characterization and experimentation.

- Additional characterization, experimentation, or modeling may be needed to reduce uncertainty in assignment of risk-significant parameters such as basemat and natural system K_ds.

- DOE's PA modeling results indicate potential excessive dispersion in the near-field and far-field models.

- Far-field model construction and calibration are generally acceptable; however, the calibration process could be improved and made more transparent in future PA updates.

- NRC staff is convinced that large voids do not currently exist in the subsurface along FTF flow paths to the 100 m (330 ft) point of compliance. However, calcareous zones that have undergone dissolution may still represent high permeability pathways that could have a significant effect on contaminant flow and transport. Additional information could be collected during the monitoring period to support DOE's modeling treatment of the calcareous zones in the lower portion of the UTR aquifer.

- DOE's dose methodology approach is well supported and adequate for the purposes of demonstrating compliance with the performance objectives in 10 CFR Part 61, Subpart C.

- NRC staff provides several recommendations in this TER where additional model support would be needed prior to tank closure to provide reasonable assurance that closure activities would comply with the requirements in 10 CFR 61.41.

- DOE developed reasonable exposure scenarios to evaluate the inadvertent intruder and demonstrated that performance objectives in 10 CFR 61.42 could be met. However, due to overlap in the groundwater analyses for 10 CFR 61.41 and 10 CFR 61.42, compliance with 10 CFR 61.42 is tied to resolution of technical issues associated with the 10 CFR 61.41 analysis.

- NRC staff noted a few instances where biosphere parameters are not fully supported.

- DOE's effort to study uncertainty in modeling predictions and identify important model sensitivities is commendable.

- Improvements can be made to DOE's parameter distribution assignments[10], hybrid modeling approach, benchmarking process, and evaluation and interpretation of probabilistic modeling results as detailed in Section 4.2.19 of this TER.

- DOE can demonstrate compliance with protection of individuals during operations (10 CFR 61.43).

- NRC staff thinks that additional information is needed to evaluate site-stability under 10 CFR 61.44 as detailed in Section 4.3.4.

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[10] Informed by additional data collection, peer review, or experiments, etc.

NRC staff's primary recommendation related to Criterion 3 is as follows:

- NRC staff recommends that DOE conduct waste release experiments to increase support for key modeling assumptions related to: (i) the evolution of pH and Eh in the grouted tank system over time; (ii) identification of HRR association with solid phases comprising the residual wastes; and (iii) expected solubility of HRRs under a range of environmental or service conditions that the residual wastes in the contaminated zone are expected to be exposed to over time. Implementation of this recommendation is deemed crucial for NRC staff to have reasonable assurance that the performance objectives in 10 CFR Part 61, Subpart C can be met. Given the risk-significance of Tank 18 to the overall PA and the short timeline for closure of this tank, DOE should initiate discussions with NRC staff regarding implementation of this recommendation for Tank 18 as soon as practical. Experiments to address this recommendation should be conducted prior to final closure of this single tank. Results of the Tank 18 residual waste experiments will be evaluated by NRC staff to determine the need for additional data collection, experiments, modeling, etc. for Tank 18, as well as other FTF tanks. Additional information regarding NRC staff's recommendations in this area, including details on the suggested implementation of other recommendations listed below will be provided in NRC staff's plan for monitoring the FTF later in FY2012. (High Risk-Significance, Short and Intermediate Term)

Other key recommendations related to Criterion 3 include:

1. NRC recommends that DOE perform a systematic scenario analysis process in which FEPs are identified, screened, and dispositioned using transparent and traceable documentation or the FEPs considered, the screening arguments, and how FEPs are implemented in the models to support future waste determination efforts. (Medium Risk-Significance, Intermediate Term)

2. NRC supports DOE's commitment to sample each tank following waste retrieval activities and will follow-up with DOE on sampling and analysis of cleaned tanks during the monitoring period. (High to Moderate Risk-Significance, Short and Intermediate Term)

3. DOE should consider how it might improve far-field model calibration and transparency in future updates to its PA. (Medium Risk-Significance, Intermediate Term)

4. DOE should continue to evaluate the appropriateness of selected transport parameters (e.g., dispersivities and K_ds, particularly for calcareous zones) and selection of sorption models (see discussion in Section 4.2.9.4 on Pu transport) during the monitoring period. (Medium Risk-Significance [Pu sorption High Risk-Significance], Intermediate Term [Pu sorption Short Term])

5. DOE should consider additional data collection related to calcareous zone outcrop locations and tracer tests to provide further support for the adequacy of its modeling treatment of the UTR-LZ aquifer. (Medium Risk-Significance, Intermediate Term)

6. Finally, DOE should evaluate the need for additional vertical or horizontal mesh refinement to ensure that contaminant plumes are not artificially dispersed over the volume of the cells in the far-field model and that time discretization is adequate. (Medium Risk-Significance, Intermediate Term)

5. OVERALL NRC REVIEW RESULTS AND RECOMMENDATIONS

As discussed in detail in previous sections, NRC staff has conducted a technical analysis of DOE's waste determination for FTF waste at the SRS. NRC's evaluation is based on information presented in DOE's draft Basis for Section 3116 Determination dated September 30, 2010; DOE's responses to NRC's RAIs; supporting references; and information provided during meetings between NRC and DOE. Regarding Criteria 2 and 3, NRC staff has provided a number of recommendations, the implementation of which will strengthen DOE's basis for concluding stabilized waste in FTF tanks and associated ancillary structures and equipment can meet NDAA criteria at the time of closure. Additionally, NRC staff suggests that at a minimum, implementation of a subset of its recommendations (e.g., high risk, short term recommendations) is needed for NRC staff to have reasonable assurance that performance objectives in 10 CFR Part 61, Subpart C can be met. The NDAA requires NRC, in coordination with the State of South Carolina to monitor disposal actions taken by DOE to assess compliance with the performance objectives in 10 CFR 61, Subpart C. NRC will coordinate with the SCDHEC to develop a plan by which NRC and the state will monitor DOE's disposal actions. NRC's monitoring plan will provide additional details regarding the priority and sequencing of recommendations necessary for NRC to have reasonable assurance that the performance objectives can be met. For example, NRC staff thinks that the need for implementation of many of NRC staff's recommendations under Criterion 3 will be contingent on results of key waste release experiments. It should be noted that NRC staff is providing consultation to DOE as required by the NDAA, and NRC staff is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is HLW, in consultation with NRC. This NRC staff assessment is a site-specific evaluation and is not a precedent for any future decisions regarding non-HLW or incidental waste determinations at SRS or other sites.

6. LIST OF REVIEWERS

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