Technical Evaluation Report

For H-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

α	dispersitivity
ac	acre(s)
ADAMS	Agencywide Documents Access and Management System
ADMP	advanced design mixer pump
ALARA	as low as is reasonably achievable
AMSL	above mean sea level
Bq	becquerels
CAP	Clean Air Act Assessment Package
CFR	Code of Federal Regulations
Ci	curie(s)
cm	centimeter(s)
CNWRA	Center for Nuclear Waste Regulatory Analyses
CSR	chemical sludge removal
CTS	concentrate transfer system
CZ	contaminated zone
DOE	U.S. Department of Energy
DWPF	Defense Waste Processing Facility
E_h	measure of reduction (or oxidation) potential
EPA	U.S. Environmental Protection Agency
FEPs	features, events, and processes
FDB	F-Area Tank Farm diversion box
FFA	Federal Facility Agreement
FGR	Federal Guidance Report
ft	feet
FTF	F-Area Tank Farm
g	gram(s)
gal	gallon(s)
GBM	gradient boosting models
GCL	geosynthetic clay liner
GCU	Gordon Confining Unit
GSA	General Separations Area
HDPE	high-density polyethylene
HELP	Hydrologic Evaluation of Landfill Performance
HLW	high-level waste
HM	H-modified
hp	horsepower
HPP	H-Area Tank Farm pump pit
hr(s)	hour(s)
HRR	highly radioactive radionuclide
HTF	H-Area Tank Farm
IAEA	International Atomic Energy Agency
ICM	integrated conceptual model
ICRP	International Commission on Radiological Protection
IP	inspection port
in	inch(es)

ACRONYMS AND ABBREVIATIONS (continued)

K_d(s)	distribution coefficients
kg	kilogram(s)
K_H	horizontal hydraulic conductivity
km	kilometer(s)
K_V	vertical hydraulic conductivity
lb(s)	pound(s)
L	liter(s)
LLW	low-level waste
m	meter(s)
MCCs	moisture characteristic curves
MEP	maximum extent practical
mi	mile(s)
mol	mole(s)
mrem	millirem(s)
mSv	millisievert(s)
NCRP	National Council on Radiation Protection and Measurements
NDAA	Ronald W.Reagan National Defense Authorization Act for Fiscal Year 2005
NEA	Nuclear Energy Agency
NRC	U.S. Nuclear Regulatory Commission
PA	performance assessment
pH	measure of acidity or alkalinity of a solution
PMP	probable maximum precipitation
ppm	parts per million
psi	pounds per square inch
PUREX	plutonium-uranium redox extraction
R^2	coefficient of determination
RAI	requests for additional information
RMS	root mean squared
s	second(s)
SA	special analysis
SCDHEC	South Carolina Department of Health and Environmental Control
SDF	Saltstone Disposal Facility
SEE	systems engineering evaluation
SLP	standard slurry pump
SMP	submersible mixer pump
SRNL	Savannah River National Laboratory
SRS	Savannah River Site
Sv	sieverts
TCCZ	Tan Clay Confining Zone
TEDE	total effective dose equivalent
TER	technical evaluation report
TRR	technical review report
USGS	U.S. Geological Survey
UTRA	Upper Three Runs Aquifer
UTRA-LZ	Upper Three Runs Aquifer–Lower Zone
UTRA-UZ	Upper Three Runs Aquifer–Upper Zone

ACRONYMS AND ABBREVIATIONS (continued)

WCS	Waste Characterization System
WD	waste determination
wt%	weight percent
yr(s)	year(s)

EXECUTIVE SUMMARY

On February 6, 2013, the U.S. Department of Energy (DOE) submitted the "Draft Basis for Section 3116 Determination for Closure of H-Area Tank Farm (HTF) at the Savannah River Site," DOE/SRS-WD-2013-001, Rev. 0 (draft basis document for the waste determination [WD] for HTF) 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 the 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 for the WD for HTF is to demonstrate that the stabilized residuals in the 29 underground radioactive waste storage tanks in the HTF, as well as related ancillary facilities, are not HLW, and therefore, meet the criteria in NDAA Section 3116.

Consistent with the requirements in Section 3116 of the NDAA, DOE consulted with the NRC on the draft basis document for the WD for HTF. The NRC staff has conducted a review of this draft basis document and related HTF performance assessment (PA), as well as DOE responses to the NRC staff's requests for additional information. Accordingly, the NRC staff offers recommendations to DOE in this Technical Evaluation Report (TER) for closure of the 29 underground radioactive waste storage tanks in the HTF. The NRC staff'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 WD activities. The Secretary of Energy, in consultation with the NRC, is responsible for determining whether the waste streams addressed in the draft basis for the WD for HTF are not HLW, and therefore, satisfy the requirements in Section 3116 of the NDAA.

Once a WD has been made, the NDAA requires the 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 the NRC and the State of South Carolina, DOE will also monitor activities associated with the closure of HTF 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 WD by the Secretary of Energy, DOE will proceed with cleaning and closure of the HTF tanks. HTF 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 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 2015.

The HTF and H-Canyon comprise the H-Area of the SRS. The HTF contains 29 below-grade, carbon steel and reinforced concrete HLW tanks, which store liquid radioactive waste generated primarily from the chemical separations facility in the H-Canyon. As described in Section 1.1 of this TER, the HTF tanks consist of four basic tank types (I, II, III, and IV). 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). Type II

tanks were constructed from 1955–1956. They are 26 m (85 ft) in diameter and 8.3 m (27 ft) high with a nominal operating capacity of 3,914 m^3 (1,030,000 gal). The Type III tanks are the youngest HTF tanks. At a height of 11 m (34.5 ft), a diameter of 26 m (85 ft), and a storage capacity [4,940 m^3 (1,300,000 gal)] of about two times that of the Type I tanks, these tanks were added to the HTF during a 15-year period (1966-1981). Type IV tanks are the largest of the tank types in HTF. Designed and sited in the late 1950s, these tanks have a height of 11 m (34.5 ft), a diameter of 26 m (85 ft), and a storage capacity of 4,940 m^3 (1,300,000 gal.). Type IV tanks in HTF were constructed between 1958 and 1962.

HTF has many similarities to the F-Area Tank Farm (FTF), a site that DOE completed a WD for in March 2012. Tank types, sizes, and supporting infrastructure are largely the same. However, HTF contains four Type II tanks while there are no Type II tanks in FTF. HTF also includes several tanks that are either fully or partially submerged in the ground water. While bottoms of several tanks (i.e., Type IV) in FTF are in the zone of water table fluctuation, no waste tanks have historically been fully submerged. Also, in HTF, there have been several instances where primary containment leakage has resulted in significant contamination reaching tank annuli. In one case, Tank 16, there is evidence that a small amount of waste crested the annulus for a short time, resulting in some contamination in the soil immediately surrounding the tank.

Per the NDAA, both a WD and a state approved closure plan or state-issued permit are required prior to closure of the HTF tanks. To date, two of the 29 HTF tanks have undergone waste removal activities, but none of these tanks have received SCDHEC approval for closure. In addition to removing the waste, HTF 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 the tanks and annuli. For tanks that have undergone waste removal activities, DOE reports that more than 99 percent of the waste volume has been removed. DOE states that cleaning methodologies are expected to collectively remove approximately 99 percent of total waste volume and total activity based on a starting point of the maximum historical radionuclide inventory in the overall HTF, however, individual waste tanks or ancillary structures may not achieve this level of highly radioactive radionuclide (HRR) removal. DOE does not consider the 99 percent estimate to be a removal goal, and the 99 percent removal efficiency is not intended to imply a correlation between 1 percent of the historical maximum activity and acceptable facility risk.

The NDAA provides three criteria to use to determine if certain radioactive waste resulting from the reprocessing of spent fuel can be excluded from the definition of HLW. Criterion 1 calls for a determination that the waste does not require permanent isolation in a deep geologic repository for spent fuel or HLW. Criterion 2 states that the waste has had HRRs removed to the maximum extent practical (MEP). Criterion 3 requires a determination that waste will be disposed of in compliance with 10 CFR Part 61, Subpart C, performance objectives.

Regarding Criterion 1,typically all HLW requires deep geological disposal. 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. The NRC staff notes in Chapter 2 of this TER that it believes that DOE can meet NDAA Criterion 1 provided Criteria 2 and 3 are

demonstrated for the HTF. That is, the cleaned HTF tanks will not require exhumation and disposal in a geologic repository.

Regarding Criterion 2, it should be noted that two HTF tanks have undergone removal activities, but DOE has not submitted final closure documentation to SCDHEC for approval of final closure. The remaining 27 tanks have yet to be cleaned. To assess conformance with Criterion 2, the NRC staff assessed DOE's (1) estimation of radionuclide inventory that is expected to remain at HTF closure, (2) identification of HRRs, (3) selection of treatment technology, and (4) demonstration of removal to the MEP, including the costs and benefits of additional radionuclide removal.

Based on its review, the NRC staff identifies several findings related to whether DOE could demonstrate Criterion 2 would be met once all the tanks are cleaned. The NRC staff's findings related to Criterion 2 are discussed in detail in Chapter 3 of this TER. Also, DOE does not establish quantitative removal goals, therefore, while projected inventories are mainly used in the demonstration of Criterion 3, they are discussed under Criterion 2 because determination of the inventory of radionuclides is relevant to both criteria. Of the findings, the NRC staff identifies five primary findings related to remaining inventory estimations, identification of HRRs, selection of treatment technology, and demonstration of removal to the MEP. The primary findings related to Criterion 2 are summarized as follows:

- DOE's approach to developing the projected inventories (i.e., radionuclide activities) that will remain after cleaning for tanks that have not yet been cleaned relies on waste volume and radionuclide concentration assumptions. The NRC staff finds that the development of projected inventories appears to be generally conservative, in terms of Criterion 3, in that it tends to over- rather than under-project the radionuclide concentrations and remaining waste volumes. However, the NRC staff notes specific exceptions regarding the subsequent inventory adjustments (see Section 3.2.1). The NRC staff will continue to monitor DOE's efforts towards reducing the actual inventory remaining in HTF, as it pertains to Criterion 3, including optimization to maintain exposures to radiation from the waste as low as is reasonably achievable.

- Although DOE had not completed final characterization for any of the HTF tanks as of the development of this TER, the NRC staff reviewed DOE's approach for final characterization and drew upon experience from FTF tanks that have undergone final characterization. In general, DOE's approach to developing final inventories that remain in tanks after cleaning is reasonable, although the approach to managing and quantifying uncertainty in volume estimates and radionuclide sampling could be improved (see Section 3.2.4). The NRC staff reiterates a number of recommendations from FTF reviews regarding final characterization in Sections 3.2.4.1 and 3.2.4.2, which are summarized in ES-2.1 below, regarding uncertainty in volume estimates and radionuclide sampling.

- To identify HRRs that must be removed to the MEP, DOE performed an evaluation that considers risk to workers, the public, and the environment. The NRC staff finds that DOE's process for identification of HRRs is reasonable (see Section 3.4). The NRC staff provides several recommendations in Section 3.4, which are summarized in ES-2.2 and ES-2.3 below, regarding the re-examination of HRRs as additional or unexpected information becomes available during future characterizations of the HTF tanks.

- The NRC staff reviewed DOE's approaches to identify, evaluate, and implement cleaning technologies to remove HRRs to the MEP, as well as the experience gained by DOE from cleaning HTF and FTF tanks. The NRC staff finds that DOE has a reasonable program in place to identify, evaluate, and implement cleaning technologies to remove HRRs to the MEP, but that some of the processes could still benefit from additional detail as DOE gains experience cleaning the tanks (see Sections 3.6 and 3.8). The NRC staff provides several recommendations regarding future technology selection in Section 3.6 of this TER, which are summarized in ES-2.4 and ES-2.5 below. In Section 3.8 of this TER, the NRC staff also provides several recommendations regarding technology implementation and optimization to minimize limitations such as those encountered during previous cleaning experiences. These implementation and optimization recommendations are summarized in ES-2.6 and ES-2.7 below.

- The NRC staff finds that the amount of material that remains in the Tank 16 annulus, as of the development of this TER, is potentially significant as it relates to Criterion 3 (see Sections 3.8.2 and 4.2.9) and that DOE has not appropriately evaluated the risk from this annular waste. To address the potential need for additional removal of waste from the Tank 16 annulus, the NRC staff provides a recommendation in Sections 3.8.2, which is summarized in ES-2.8.

As a result of the primary findings noted above, the NRC staff has identified a number of recommendations related to Criterion 2 that are discussed in detail in Chapter 3 of this TER. The NRC staff's key recommendations[1] related to Criterion 2 are summarized as follows:

ES-2.1 The NRC staff recommends that DOE explore methods to improve the process by which residual waste volumes and associated uncertainty are estimated. The NRC staff also recommends that DOE evaluate and clearly communicate the relative contributions of various forms of uncertainty related to the radionuclide sampling process.

ES-2.2 The NRC staff recommends that DOE continue to evaluate its HRR list and provide sufficient justification for any changes as additional information becomes available. The HRR list should be evalutated especially where it is used to inform decisions, such as the selection of radionuclides characterized in residual waste, selection of treatment technologies, and the screening of radionuclides for the purpose of detailed PA calculations.

ES-2.3 The NRC staff acknowledges DOE's efforts in understanding unexpected final inventory results (e.g., larger than expected inventory of zirconium in FTF Tanks 5 and 6) and recommends that DOE continue to examine the reasons for unexpected results, should they occur, and attempt to trace them back to known

[1] The key recommendations of the NRC staff related to Criterion 2 are summarized here. Other recommendations from the NRC staff related to Criterion 2 are identified and discussed in individual sections of Chapter 3 of this TER.

waste streams or processes that might reveal other radionuclides that could have been underestimated by the projections based on the waste characterization system data. The NRC staff recommends that DOE assess, through future tank residual characterization, the validity of prior assumptions and the resulting impacts to the list of HRRs.

ES-2.4 As practical, the 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.

ES-2.5 The NRC staff recommends that DOE provide more emphasis on removal of HRRs in its technology selection process and provide a clear linkage between the HTF PA results, including information regarding the long-term risks associated with the HTF facility, and the demonstration that HRRs have been removed to the MEP per Criterion 2.

ES-2.6 The NRC staff recommends that DOE consider how it might better assess and optimize the effectiveness of selected technologies (e.g., obtain better baseline information).

ES-2.7 The NRC staff recommends that DOE continue to better define the documented process to be used to demonstrate removal to the MEP to ensure consistent (non-arbitrary) application of the criterion.

ES-2.8 Given the potential risk significance of the waste remaining in the Tank 16 annulus, the NRC staff recommends that DOE more fully evaluate the practicality of additional radionuclide removal from the Tank 16 annulus versus the long-term benefit of reduced risk considering uncertainty in the releases of radionuclides from the Tank 16 annulus. While DOE's HTF PA demonstrates that the risk from waste remaining in the annulus is reasonable, alternative waste release models may lead to higher risk estimates.

Regarding Criterion 3, 10 CFR Part 61, Subpart C establishes requirements for protection of the public, protection of an inadvertent intruder, protection of individuals during operations, and site stability. It is important to note that the NRC staff is not making a conclusion on the ability of the DOE 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, the NRC staff evaluated: (i) the waste classification for tank system components; (ii) the HTF PA results and supporting technical bases (including infiltration, near-field release, far-field transport, dose methodology, and exposure assessment); (iii) the inadvertent intruder analysis; (iv) the radiation protection program for individuals during operations; and (v) the stability of the disposal facility after closure.

The NRC staff evaluated DOE's demonstration of compliance with NDAA Criterion 3, including DOE's approaches regarding classification of the waste remaining at HTF following closure and compliance with the 10 CFR Part 61 performance objectives. Based on its review, the NRC staff identifies several findings related to whether DOE could demonstrate Criterion 3 would be met. The NRC staff's findings related to Criterion 3 are discussed in detail in Chapter 4 of this

TER. These findings are summarized for waste classification and the 10 CFR Part 61 performance objectives as follows:

- DOE's methodology for classifying waste is generally an acceptable application of Category 3 (site specific concentration averaging) in NUREG-1854. The NRC staff notes specific exceptions in Section 4.2.1.

- DOE demonstrates through the HTF PA and supporting references that the performance objective in 10 CFR 61.41 could be met if modeling assumptions are adequate. In reviewing the HTF PA, supporting documentation, and DOE responses to requests for additional information, the NRC staff finds that DOE develops reasonable exposure scenarios to evaluate releases of radioactivity. However, the NRC staff notes uncertainties in the projected releases from HTF and, principally, that DOE has provided limited support for the performance of key barriers that are considered important to demonstrating the performance objectives can be met (see Sections 4.2.3, 4.2.5, 4.2.7, 4.2.9, 4.2.11, 4.2.13, 4.2.15, and 4.2.19). For instance, DOE's assumptions regarding solubility-limiting phases, solubility limits, and chemical transition times are risk significant and have not been confirmed through waste characterization and experimentation (see Section 4.2.9.3). Further, the NRC staff notes that DOE's approach to modeling waste release from the Type I and II tanks does not adequately assess the risk from the contamination located within the annular regions (see Sections 4.2.9.3 and 4.2.9.4). The NRC staff provides a number of recommendations for DOE to develop additional support for the models used in the HTF PA and its demonstration of compliance with the performance objective at 10 CFR 61.41. The NRC staff has prioritized its recommendations related to the performance objective in 10 CFR 61.41. The most significant recommendations are summarized below in ES-3.1 through ES-3.5.

- As described in Sections 4.2.17 and 4.2.19, DOE develops reasonable exposure scenarios to evaluate potential exposures of the inadvertent intruder to radiation from the residual waste and demonstrate that performance objectives in 10 CFR 61.42 could be met. However, the NRC staff notes that the probabilistic analyses for inadvertent intrusion indicate that projected doses could exceed 500 mrem/yr within 10,000 years. Additionally, because the groundwater pathway is a key contributor to demonstrating compliance with both 10 CFR 61.41 and 10 CFR 61.42, a demonstration of compliance with 10 CFR 61.42 is dependent upon resolution of technical issues associated with the 10 CFR 61.41 analysis.

- Because DOE regulations provide an equivalent level of protection as is provided by 10 CFR Part 20, DOE has procedures and processes in place to demonstrate compliance with protection of individuals during operations (10 CFR 61.43), as discussed in Section 4.3.2.

- While the NRC staff finds that DOE has provided sufficient information to perform a preliminary review of site stability (10 CFR 61.44), additional information is needed and can be conducted during the monitoring period. The NRC staff provides recommendations for the additional information pertaining to site stability analyses that will be needed in Section 4.3.4, which are summarized in ES-3.6 below.

The NRC staff provides several recommendations in Chapter 4 of this TER along with a rating of risk significance and priority[2] where additional model support would be needed prior to tank closure to provide reasonable assurance that closure activities would comply with the 10 CFR Part 61 performance objectives. The NRC staff's primary recommendation related to Criterion 3 is similar to the primary recommendation that was identified by the NRC staff in the FTF TER (Camper, 2011 [ADAMS[3] Accession No. ML112371751]). Other key recommendations with moderate or greater risk significance[4] related to Criterion 3 are also summarized below:

ES-3.1 As its primary recommendation, the NRC staff reiterates its FTF recommendation that DOE conduct waste release experiments. The NRC staff began monitoring activities related to this recommendation for FTF as part of its monitoring responsibilities. The NRC staff will continue to monitor DOE's activities to address this recommendation when it combines monitoring activities for both FTF and HTF (High Risk Significance, Short and Intermediate Term).

ES-3.2 DOE should conduct a more comprehensive analysis of contaminant release from the annular regions of Type I and II tanks (Medium-to-High Risk Significance, Short and Intermediate Term).

ES-3.3 DOE should continue to sample each tank following waste retrieval activities and will follow-up with the NRC staff on sampling and analysis of cleaned tanks during the monitoring period (High-to-Medium Risk Significance, Short and Intermediate Term).

ES-3.4 DOE should continue to evaluate the appropriateness of selected transport parameters (e.g., cementitious material and soil K_ds) and the selection of sorption models during the monitoring period (Medium Risk Significance, Intermediate Term).

ES-3.5 DOE should improve the calibration of the far-field model, particularly local to H-Area and continue to study uncertainty in calibration targets (Medium Risk Significance, Intermediate Term).

ES-3.6 DOE should perform a closure cap settlement and stability analysis during the monitoring period (Medium Risk Significance, Intermediate Term).

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[2] 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.

[3] The Agencywide Documents Access and Management System (ADAMS) is the official recordkeeping system through which the NRC provides access to its collections of publicly available documents. ADAMS can be accessed at online at http://www.nrc.gov/reading-rm/adams.html.

[4] The NRC staff's recommendations with lower risk significance related to Criterion 3 are identified and discussed in individual sections of Chapter 4 of this TER.

Statements and recommendations contained in this TER are based on the NRC staff's review of DOE's Draft Basis for Section 3116 Determination for Closure of HTF at SRS dated February 6, 2013, DOE responses to the 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 the NRC staff 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. As stated above, the Secretary of Energy, in consultation with the NRC, is responsible for determining whether the waste is HLW. This assessment by the NRC staff is a site-specific evaluation and is not a precedent for any future decisions regarding non-HLW or incidental WDs at SRS or other DOE sites.

Because DOE is still in the early stage of HTF cleaning and significant technical uncertainties remain regarding DOEs ability to meet the performance objectives, it would be premature for the NRC to draw conclusions in that regard now. However, the NRC is confident that, if DOE implements the recommendations contained in this TER, the likelihood of meeting the 10 CFR Part 61 performance objectives is enhanced.

1. INTRODUCTION

1.1 Facility and Site Description

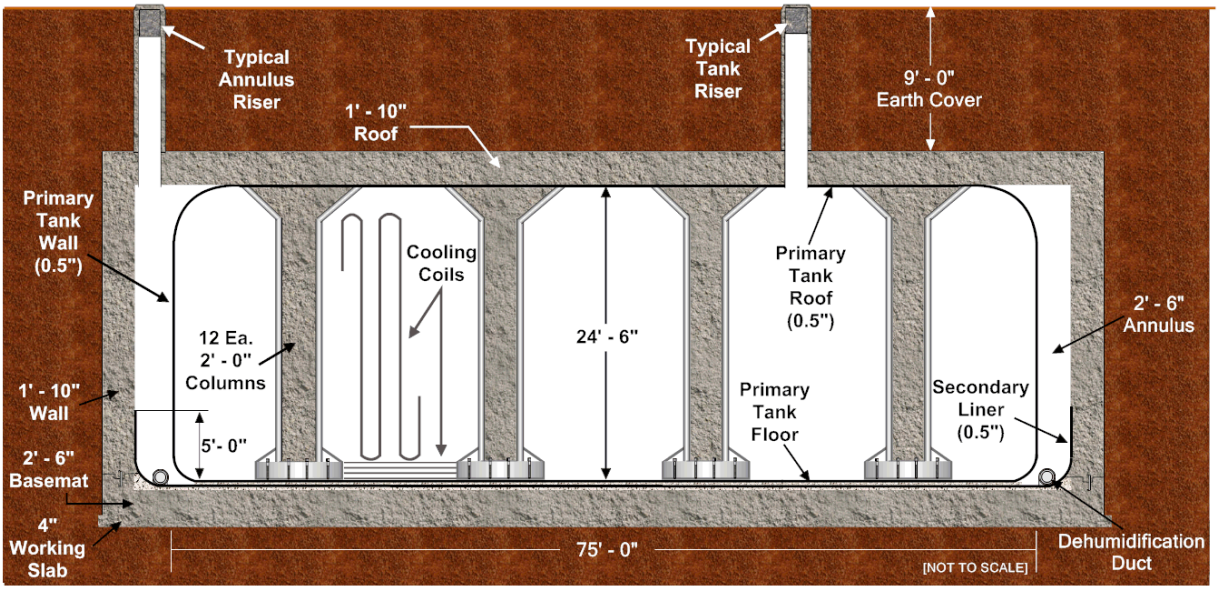
The Savannah River Site (SRS) is an 800 km^2 (310 mi^2) site, located in the upper coastal plain of South Carolina, adjacent to the Savannah River. The site is approximately 19 km (12 mi) south of Aiken, South Carolina, and 24 km (15 mi) southeast of Augusta, Georgia. It is owned by the U.S. Department 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 liquid 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 tank farms associated with two separations areas: H-Area and F-Area. H-Area Tank Farm (HTF) contains 29 tanks and F-Area Tank Farm (FTF) houses the remaining 22 tanks. Of the 51 tanks, four tanks in FTF have been closed (i.e., Tanks 17-20) and of the 47 remaining tanks that have not been closed, 29 are in the HTF while 18 are in the FTF.

DOE employed a variety of tank designs that varied in capacity, containment, and internal support features. Twenty-seven tanks were designed and built with secondary containment for the full height of the primary tank and also included leak detection systems. These tanks meet the current U.S. Environmental Protection Agency (EPA) requirements for underground storage tanks. The remaining 24 tanks do not have secondary containment for the full height of the primary tank and do not meet current EPA requirements. Twelve of the tanks that do not have full-height secondary containment are located in HTF with the remainder in FTF. Several of the tanks without secondary containment the full height of the primary tank have leaked, but sufficient waste has been removed from these tanks so that there are currently no active leak sites. At the time of the writing of this Technical Evaluation Report (TER), HTF Tanks 9 through 16 have leaked waste from the primary containment.

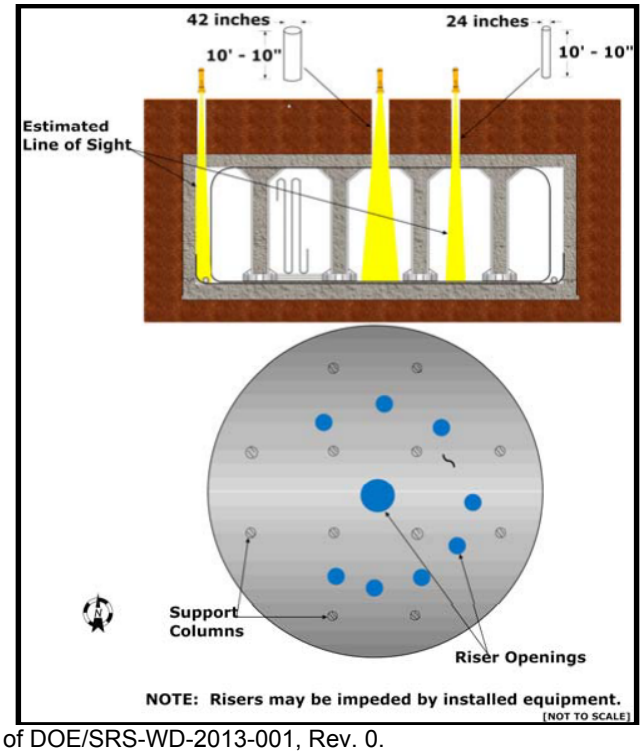
The HTF occupies approximately 0.18 km^2 (45 ac) of the 1.6 km^2 (395 ac) H-Area within the General Separations Area (GSA) near the center of the SRS. The HTF contains 29 carbon steel waste tanks of five designs (Figures 1-1 through 1-5), four with a nominal capacity of 2,850 m^3 (750,000 gal) per tank (Type I); four with a nominal capacity of 3,900 m^3 (1,030,000 gal) per tank (Type II) and 21 with a nominal capacity of 4,940 m^3 (1,300,000 gal) per tank (Types III, IIIA, and IV). Most of the tank waste in HTF originated in the H-Canyon chemical separations facility. The NRC staff evaluates DOE's approach for treating and disposing of HTF waste tanks, ancillary equipment, and residual radioactivity in this HTF TER.

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



Adapted from Figure 2.1-29 of DOE/SRS-WD-2013-001, Rev. 0.

(a)

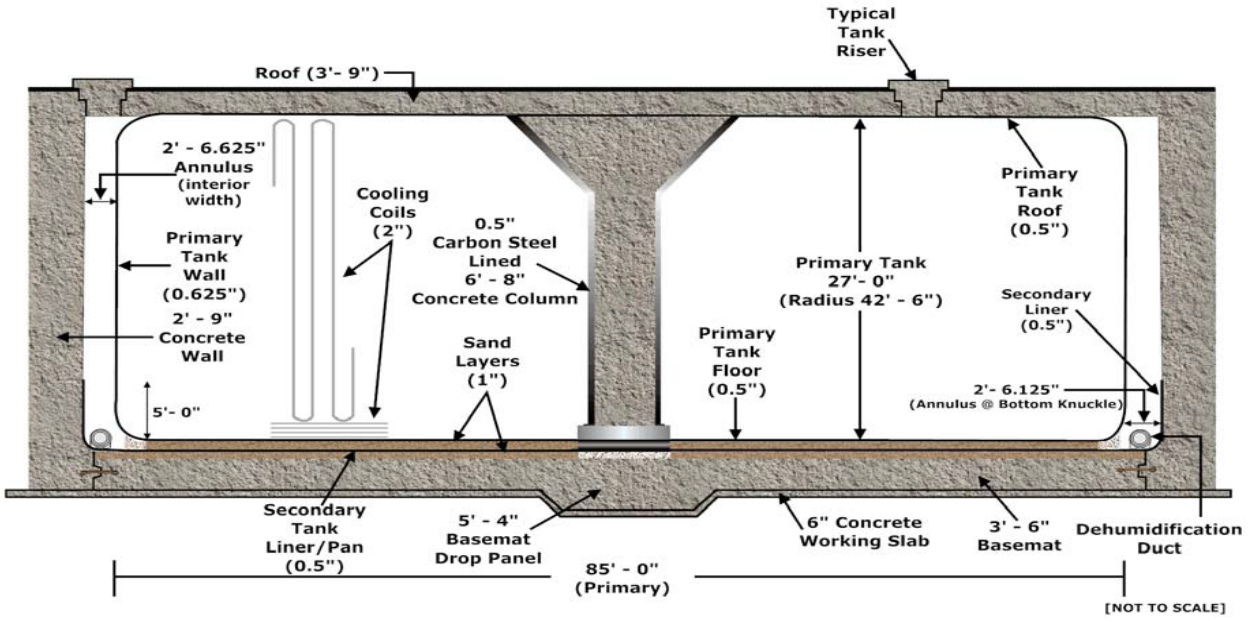


Adapted from Figure 2.1-34 of DOE/SRS-WD-2013-001, Rev. 0.

(b)

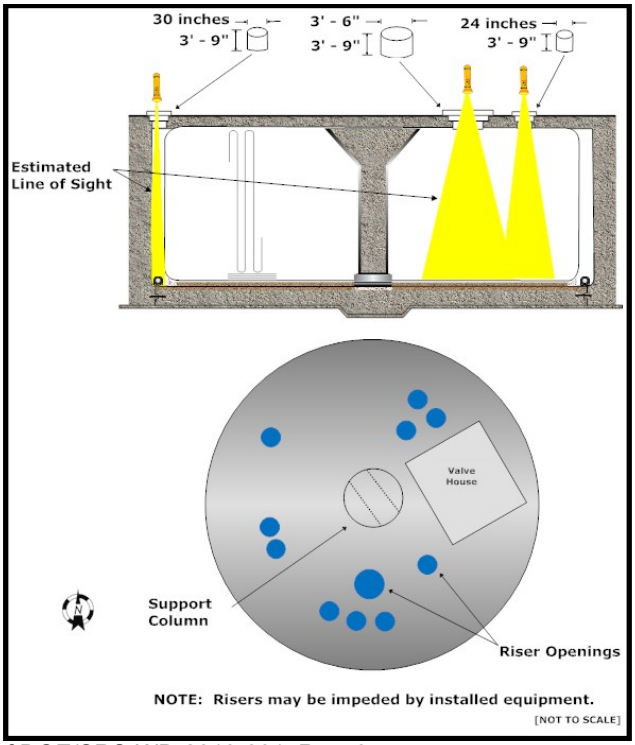
To convert feet to meters, multiply by 0.3048.
To convert inches to centimeters, multiply by 2.54.

Figure 1-2 Typical Type II Tank Cross Sections (a) and Plan View of Access Ports (b)



Adapted from Figure 2.1-35 of DOE/SRS-WD-2013-001, Rev. 0.

(a)

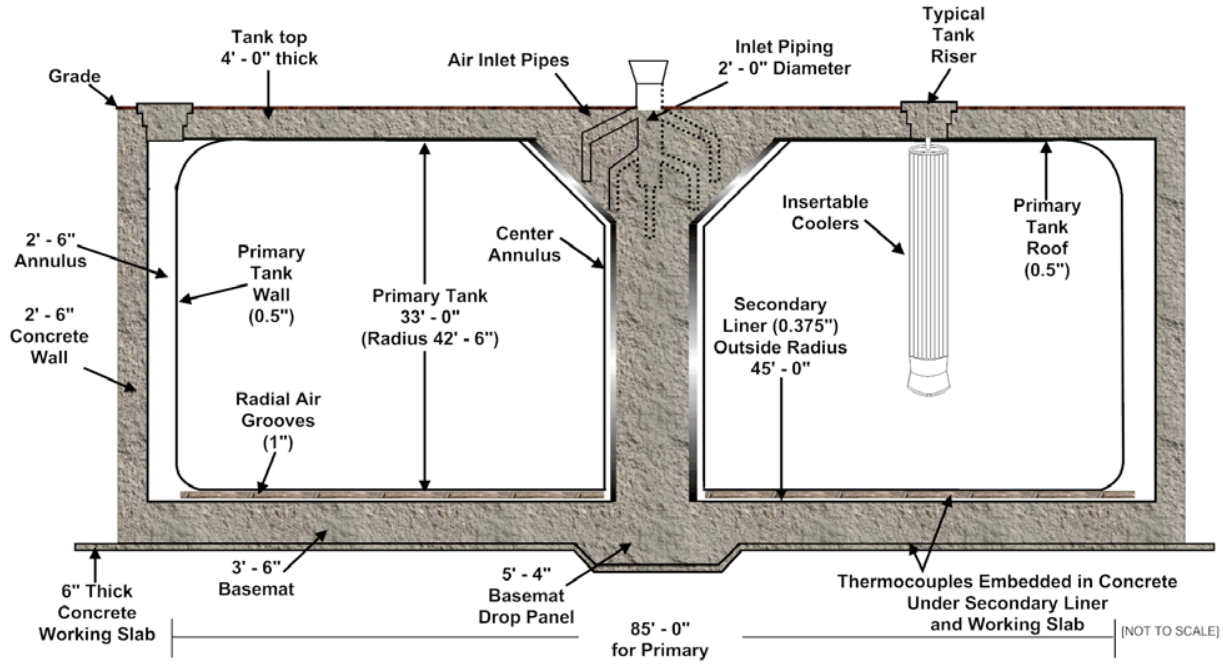


Adapted from Figure 2.1-41 of DOE/SRS-WD-2013-001, Rev. 0.

(b)

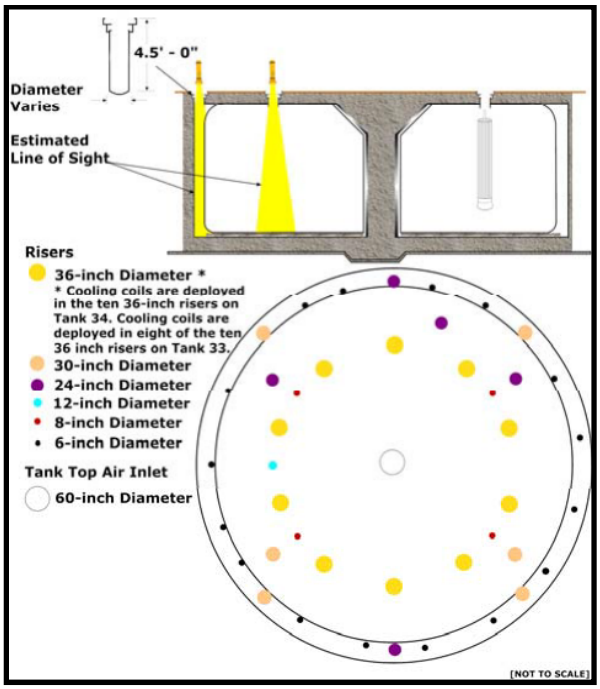
To convert feet to meters, multiply by 0.3048.
To convert inches to centimeters, multiply by 2.54.

Figure 1-3 Typical Type III Tank Cross Sections (a) and Plan View of Access Ports (b)



Adapted from Figure 2.1-42 of DOE/SRS-WD-2013-001, Rev. 0.

(a)

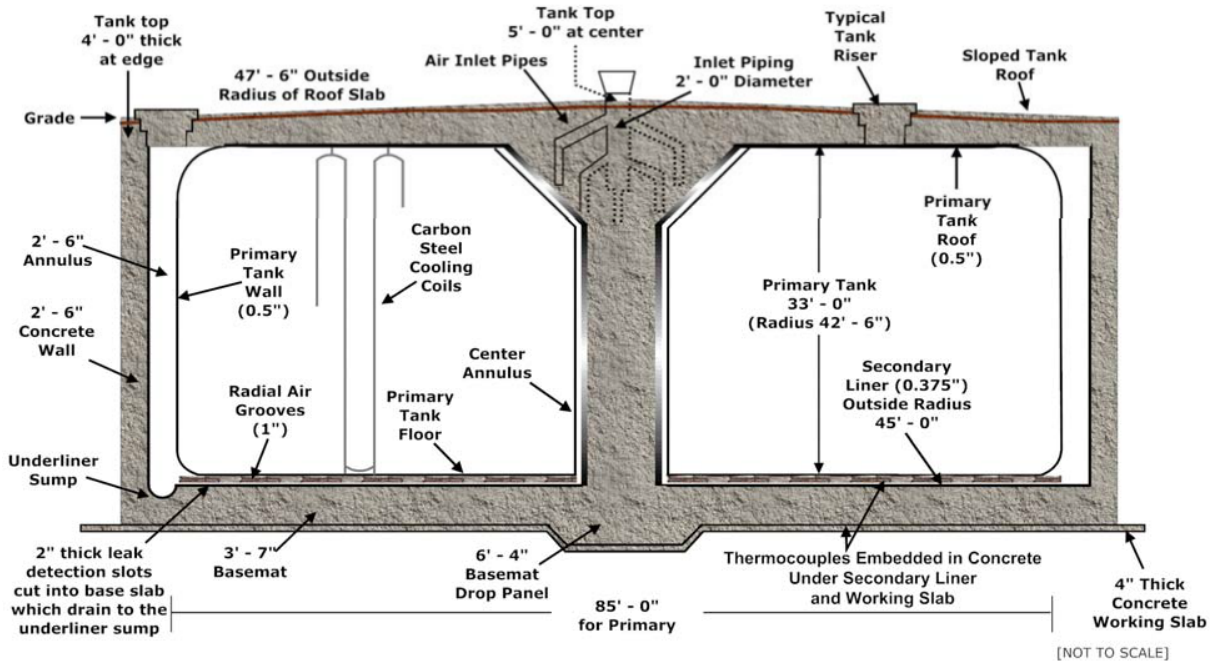


Adapted from Figure 2.1-49 of DOE/SRS-WD-2013-001, Rev. 0.

(b)

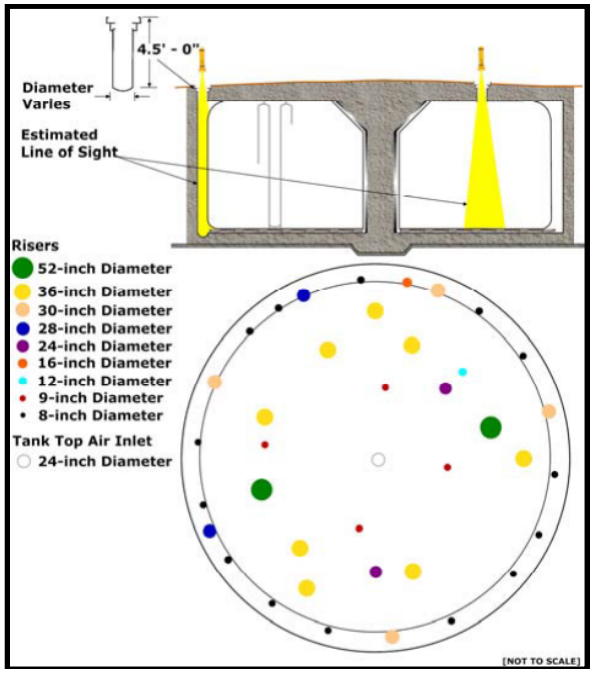
To convert feet to meters, multiply by 0.3048.
To convert inches to centimeters, multiply by 2.54.

Figure 1-4 Typical Type IIIA Tank Cross Sections (a) and Plan View of Access Ports (b)



Adapted from Figure 2.1-42 of DOE/SRS-WD-2013-001, Rev. 0.

(a)

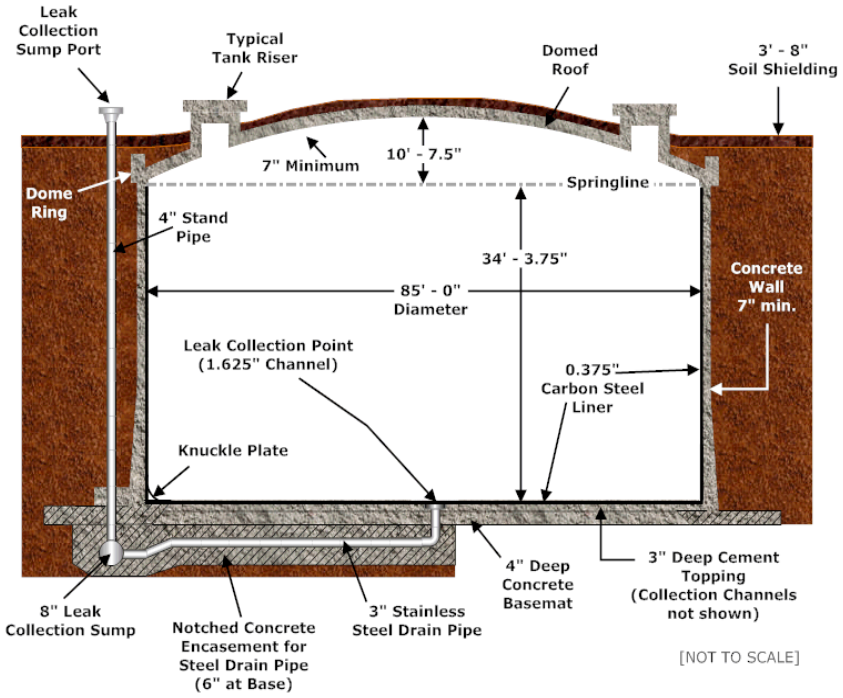


Adapted from Figure 2.1-49 of DOE/SRS-WD-2013-001, Rev. 0.

(b)

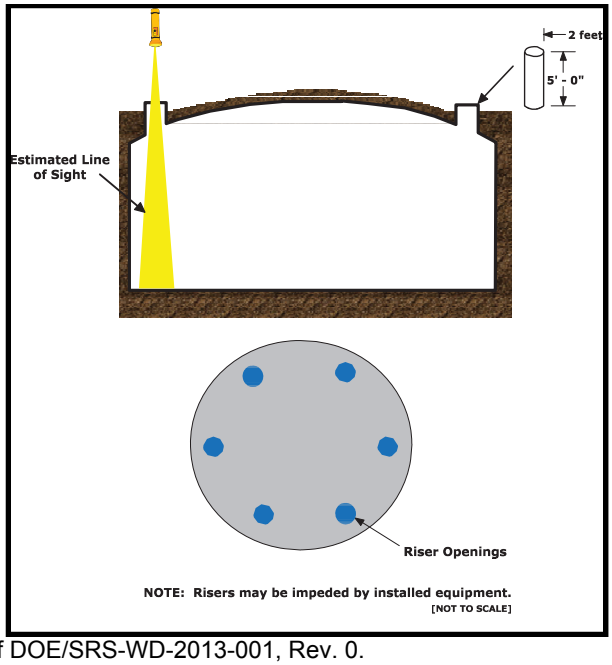
To convert feet to meters, multiply by 0.3048.
To convert inches to centimeters, multiply by 2.54.

Figure 1-5 Typical Type IV Tank Cross Section (a) and Plan View Map of Access Ports (b)



Adapted from Figure 2.1-51 of DOE/SRS-WD-2013-001, Rev. 0.

(a)



Adapted from Figure 2.1-55 of DOE/SRS-WD-2013-001, Rev. 0.

(b)

To convert feet to meters, multiply by 0.3048.
To convert inches to centimeters, multiply by 2.54.

1.1.1 Land and Water Use

More than 90 percent of the SRS's land area consists of natural forests and managed pine plantations with the remainder used as aquatic wildlife habitats or developed for industrial purposes.. Approximately five 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 HTF 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 H-Area.

Because the SRS is controlled by the Federal Government, the public does not have unrestricted access to any site facility. DOE expects restrictions on public access and land use to continue based on the following assumptions about land ownership and use that are articulated in the SRS End State Vision (PIT-MISC-0089):

- The entire site will be owned and controlled by the Federal Government, in perpetuity.
- The property will be used only for industrial purposes.
- Site boundaries will remain unchanged.
- Residential use will not be allowed onsite.

The Savannah River bounds the SRS to the southwest for approximately 32 km (20 mi); the site is 257 river km (160 river mi) 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 (a clear, cold, spring-fed stream), Fourmile Branch, 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. According to DOE/SRS-0303, Rev. 0, the Savannah River and streams on the SRS are classified by the South Carolina Department of Health and Environmental Control (SCDHEC) as freshwaters, which are defined as (1) surface water suitable for primary and secondary contact recreation, (2) drinking water (after treatment), (3) fishing, (4) survival and propagation of an indigenous aquatic community of fauna and flora, and (5) industrial and agricultural uses (cf. SCDHEC Regulations 61.68, "Water Classifications & Standards," and 61.69, "Classified Waters"). Prominent surface water features at the SRS include Par Pond and L Lake, which 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 HTF.

Upper Three Runs, its unnamed tributaries, Crouch Branch, and Four Mile Branch and its tributaries are perennial streams near the HTF. U.S. Geological Survey (USGS) gauging stations monitor flow in Upper Three Runs and Four Mile Branch near H-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 including these streams and tributaries. Continuous monitoring of water flowing through streams also 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

The Carolina bays on the 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. Slightly elevated, well-drained areas are characterized by a mixture of oak species, as well as red maple, sweetgum, and other hardwood species. Low-lying areas that are perennially flooded are dominated by second-growth bald cypress and water tupelo. Nearer the H-Area, the Four Mile Branch and Upper Three Runs 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 Four Mile Branch are tag alder, willow, sweetgum, and wax myrtle. The smooth purple coneflower is an endangered plant species occurring on the SRS (DOE/EIS-0303). In addition to the bottomland hardwood forests, several regions surrounding the H-Area consist of pine plantations managed for timber production by the U.S. Forest Service.

The 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 SRS include 55 mammalian species (e.g., white-tailed deer, feral hog, raccoon, beaver, otter, gray fox, opossum, bobcat, gray squirrel, fox squirrel); macroinvertebrates and fish observed in Upper Three Runs (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 (e.g., salamanders, frogs and toads, the American alligator, turtles, lizards, and snakes) (WSRC-TR-2005-00201).

The HTF 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.

1.1.3 Local Meteorology and Climatology

Climate at the 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; 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

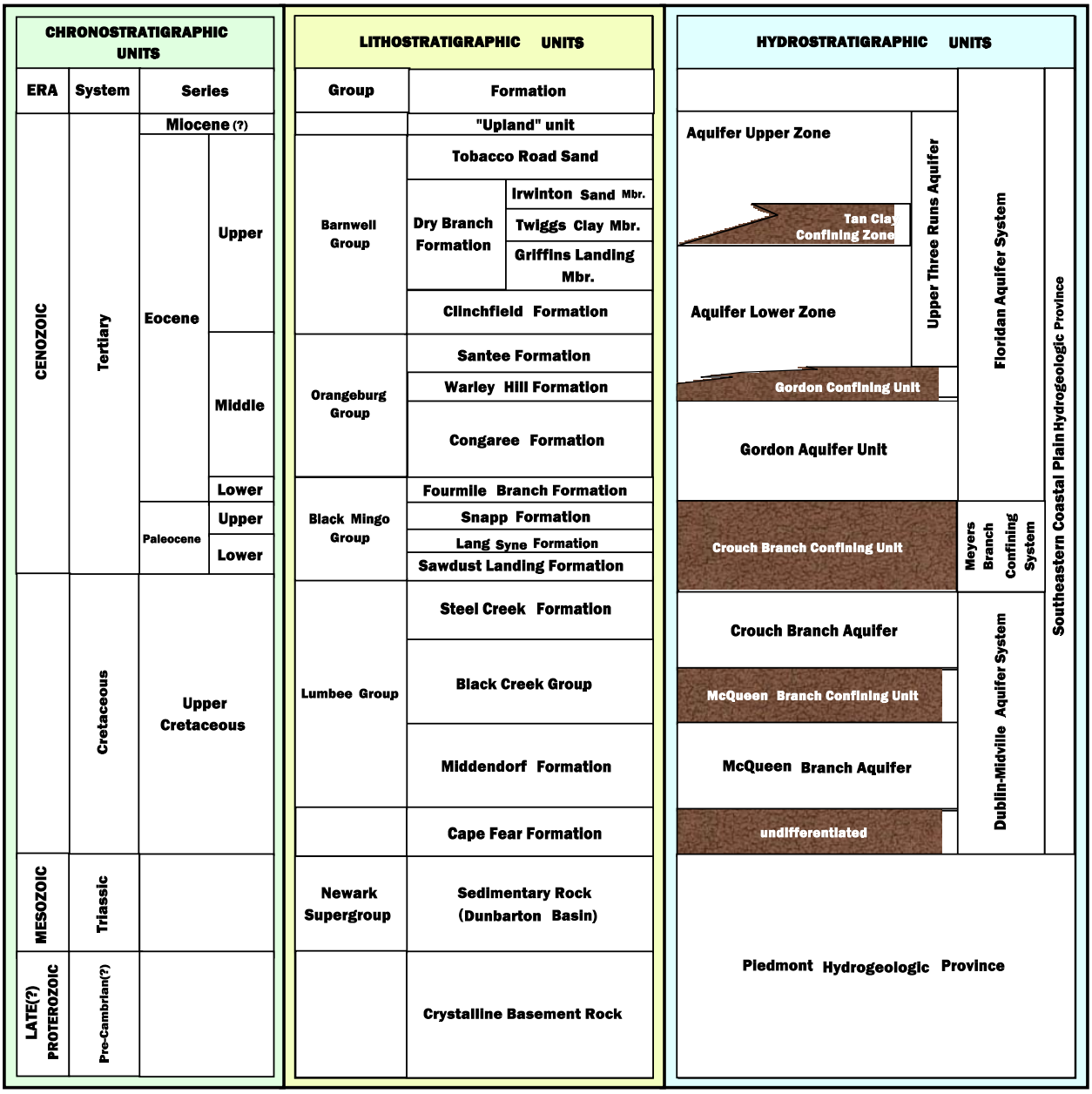
The SRS lies on the Atlantic Coastal Plain, above a southeast-dipping wedge of unconsolidated and semi-consolidated sediments forming the Upper Coastal Plain sedimentary sequence. Sediments range in age from Late Cretaceous to Recent and include sands, clays, limestones and gravels. At the SRS, this sedimentary sequence ranges in thickness from 210 m (700 ft) at the northwestern boundary to 370 m (1,200 ft) at the southeastern boundary (WSRC-TR-95-0046).

The Coastal Plain sedimentary sequence forms a series of aquifers and confining or semi-confining units. Aquifer systems below the SRS include the Floridan 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 (Figure 1-6).

Upper Three Runs is a large, long, blackwater stream that drains an area greater than 500 km^2 (195 mi^2) as it flows to the southeast. The stream is approximately 40 km (25 mi) long, and has its lower 27 km (17 mi) within SRS boundaries. Upper Three Runs receives more water from groundwater sources than other SRS streams; it is located approximately 2.4 km (1.5 mi) to the northwest of the HTF. The Upper Three Runs 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. 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 Upper Three Runs.

Four Mile Branch, a smaller perennial blackwater stream, has its headwaters near the center of the SRS. It drains a 57 km^2 (22 mi^2) area (including much of H-Area) as it flows to the southeast, and traverses 24 km (15 mi) before discharging at the SRS boundary to the Savannah River or Beaver Dam Creek. The V-shaped valley of Four Mile Branch has a floodplain ≤305 m (≤1,000 ft) in width with steep to gently sloping sides. Four Mile Branch is located approximately 1.2 km (0.7 mi) to the southwest of the HTF. The calculated 100-yr,

Figure 1-6 Chronostratigraphic, Lithostratigraphic, and Hydrostratigraphic Units at SRS



Adapted from Figure 2.1-15 of DOE/SRS-WD-2013-001, Rev. 0.

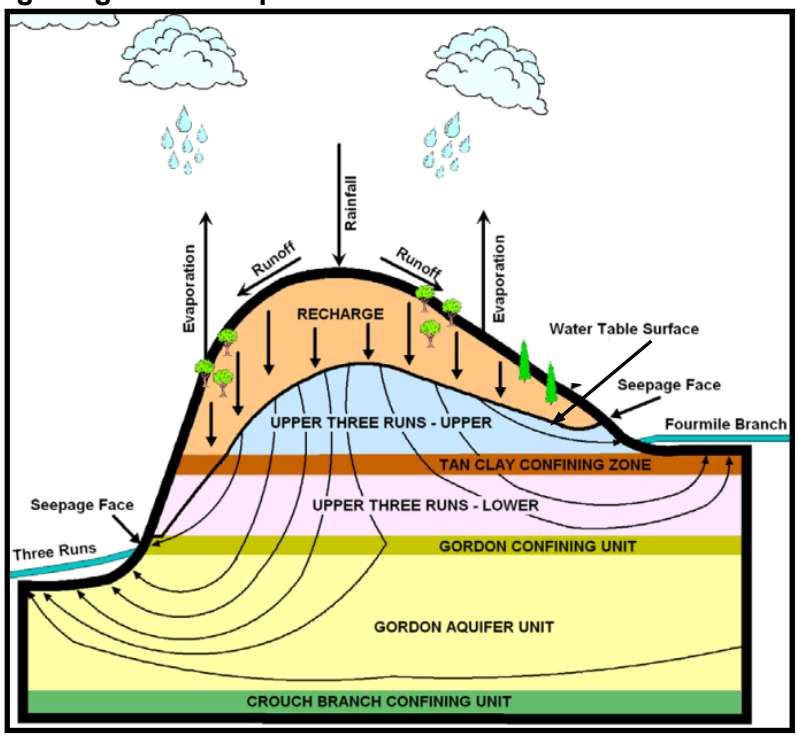
1,000-yr, 10,000-yr and 100,000-yr flood elevations for Four Mile Branch Basin near H-Area are 71.4, 71.7, 71.9 and 72.2 m (234.3, 235.2, 235.8 and 236.8 ft) above mean sea level (AMSL), respectively (WSRC-TR-99-00369), which is at least 0.9 m (3.1 ft) below the lowest waste tank bottom (73.1 m [239.9 ft]) based on elevations provided in DOE's HTF PA, Rev. 1, Table 3.2-1 (SRR-CWDA-2010-00128, Rev. 1). The lowest elevation of the lower foundation layer at the bottom of the side slope of the engineered closure cap is approximately 85 m (280 ft) AMSL, which is approximately 13.5 m (44 ft) above the 10,000-yr floodplain level (SRR-CWDA-2010-00128, Rev. 1). Because the HTF is not located within a floodplain and has adequate topographic relief for runoff, DOE does not consider flooding to be an issue of concern.

The Floridan Aquifer System is of primary importance for estimating the ultimate flow and transport of residual HTF contaminants. At the SRS, the Floridan Aquifer System is composed of the upper and lower zones of the Upper Three Runs Aquifer (UTRA) above, the Gordon Aquifer below, and their associated confining units: (1) the Tan Clay (semi)-Confining Zone (TCCZ) within UTRA and (2) the Gordon Confining Unit (GCU) separating the UTRA from the Gordon Aquifer (Figure 1-6). The saturated thickness of the upper zone of the UTRA (UTRA-UZ) ranges from 12 to 43 m (40 to 141 ft) with an average of 24 m (80 ft). The thickness of the lower zone (UTRA-LZ) ranges from 16- to 24-m (55- to 80-ft) thick with an average of 18 m (60 ft) (SRNL-STI-2010-00148, Rev. 0). The TCCZ is comprised of interbedded clay, sandy clay, and clayey sand with subordinate, thinly bedded sands. The TCCZ varies from 2.1 to 4.6 m (7 to 15 ft) in thickness, but its semi-confining clay layers are discontinuous, varying from 0.9 m (3 ft) thick to absent (SRNL-STI-2010-00148, Rev. 0). The HTF sits on an interfluvial ridge, above a groundwater divide between the Four Mile Branch and Upper Three Runs watersheds. DOE's conceptual model for flow in affected aquifers on the GSA is presented in Figure 1-7. Although groundwater flow in the UTRA is predominantly lateral, vertical flow at the groundwater divide is significant (SRNL-STI-2012-00465). DOE expects that contaminants will ultimately discharge from the UTRA-LZ to one or both streams, depending on source location (SRS-REG-2007-00002, Rev. 1), or cross the GCU into the Gordon Aquifer. Near H-Area, the GCU is comprised of thin clay beds, sandy mud, and sandy clay beds intercalated with lenses of quartz sand, gravelly sand and calcareous mud; the GCU, which is a significant barrier to downward flow, is 2.1-m (7-ft) thick on average (SRNL-STI-2010-00148, Rev. 0). Underlying the GCU, the Gordon Aquifer ranges in thickness from 17 to 21 m (55 to 70 ft; SRNL-STI-2010-00148, Rev. 0) and all flow discharges to the Upper Three Runs.

The vadose zone at the HTF is composed of the Altamaha Formation (i.e., the "Upland Unit") and the underlying Tobacco Road Sand Formation. The Altamaha Formation represents a complex depositional environment, dominated by fluvial processes but having occasional transitional marine influences. Its material composition is thus highly variable, consisting of poorly sorted coarse- to fine-grained quartz sand, pebbly-to-cobbly quartz sand lenses, and beds of clayey to silty sand and sandy clay. Despite its lateral and vertical heterogeneity, SRNL-STI-2010-00148, Rev. 0 simply describes the Altamaha Formation as a silty clay. Vertically upward through the unit, grain sorting generally becomes poorer, clay beds generally become more abundant and thicker, and sands generally become more argillaceous and indurated. Its lower surface is an irregular erosional contact. The Tobacco Road Sand Formation is a moderately to poorly sorted clayey quartz sand deposited under alternating fluvial and transitional marine conditions. It is extensively burrowed by Ophiomorpha and contains wispy clay laminae of kaolinite and illite (e.g., weathered feldspar and muscovite) (SRNL-STI-2010-00148, Rev. 0).

The median vadose zone thicknesses between the concrete slabs underlying the waste tanks and the UTRA-UZ water table range from −10.8 m to −2 m (−35.5 ft to −6.6 ft), for tanks either completely or partially submerged beneath the water table, to no more than 5.5 m (18.2 ft) based on thicknesses provided in SRNL-STI-2010-00148. SRNL-STI-2010-00148, Rev. 0 shows graphically that the water table elevation at HTF has fluctuated as much as

Figure 1-7 Hydrogeological Conceptual Model Near HTF



Adapted from Figure 2.1-23 of DOE/SRS-WD-2013-001, Rev. 0.

±4.6 m (±15 ft) during the documented period (a precipitous decline and recovery of this magnitude was observed at some tanks during the 1985–1987 timeframe); nevertheless, water table elevations at HTF today are similar to the median elevations documented at the site since the 1970s.

H-Area soils may contain as much as 20 to 40 percent clay (DOE/EIS-0303). The predominant general soil association at the H-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.

1.1.5 Geology—Soft Zones, Seismicity, and Volcanism

1.1.1.1 Soft Zones

At the GSA, much of the sediment wedge consists of porous media that is not thought to support significant preferential flow. There are soft zones, however, due to sediment dissolution in a portion of the UTRA-LZ within the middle-Eocene-age, McBean Member (shallow marine shoreface/shelf facies) of the Tinker/Santee Formation. 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). WSRC-RP-94-54 indicates that

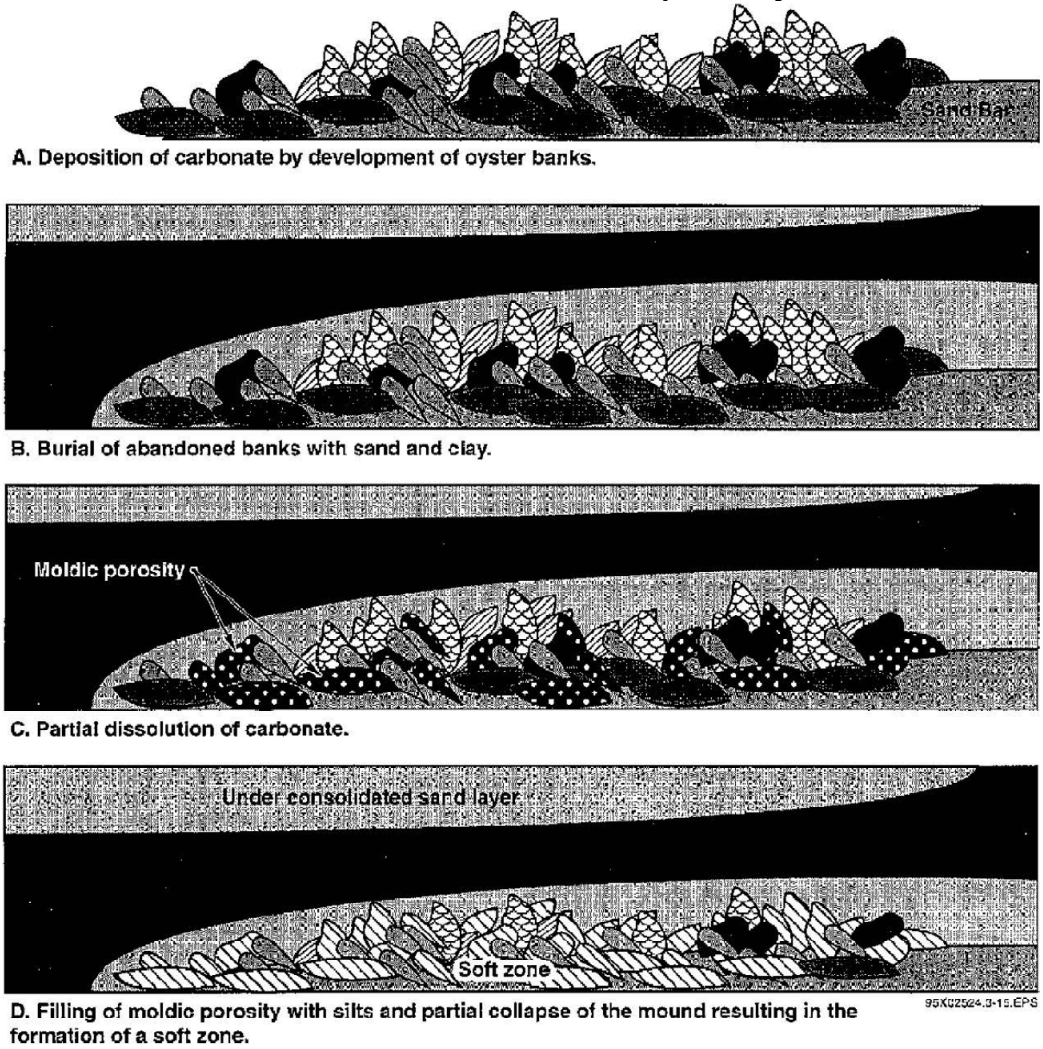
litho- and hydrostratigraphic positions of soft zones were originally documented by isopach maps based on well data within an application for a hazardous waste postclosure permit for H-Area (DOE, 1990). This analysis included information gleaned from two wells in H-Area (cf. WSRC-RP-93-00287), but neither were noted to intersect carbonate. However, HTF Tank 38 may be located above or near the edge of a known sinkhole in the UTRA-LZ (SRNL-TR-2012-00160, Rev. 0). Soft zones found within exploratory boreholes were filled with grout to provide stability to critical infrastructure. Sixteen percent of boreholes drilled in the HTF were grouted due to soft zone intersection. Calcareous rock was observed in almost 40 percent of boreholes drilled near Building 241-H (SRNL-TR-2012-00160, Rev. 0).

Soft zones and calcareous material (e.g., calcarenitic and shelly limestone, marl, calcareous quartz sand, and sandy shell hash) (Siple, 1967; Evans, 1995) are generally more pervasive down-dip toward the southeast (SRNL-STI-2010-00148, Rev. 0; RAI-FF-1 Response, SRR-CWDA-2011-00054, Rev. 1). SRNL-STI-2010-00148, Rev. 0 indicates that below H-Area at the far eastern edge of the tank farm (e.g., below In-Tank Precipitation Tanks 48 and 49) is a highly variable, interfingering transition zone of the UTRA-LZ. In this transition zone, the poorly cemented calcareous sediments of the Santee Limestone to the south and east laterally grade into the silica-cemented, clayey-to-silty sands of the Tinker Formation to the north and west. Dense, well-cemented, low hydraulic conductivity micritic limestone at the base of the McBean Member is not of concern because it acts to strengthen the integrity of the underlying GCU (Evans, 1995). In contrast, soft zones composed of highly porous, high hydraulic conductivity shelly limestone and coarse-grained shell hash that occur in the center of the McBean Member (Evans, 1995) are of concern (Figure 1-8).

Where soft zone facies are present in the UTRA-LZ (e.g., WSRC-RP-93-00287), they likely cause preferential flow that dominates the flow field (Evans, 1995). Grouting operations assessed at K-Area support this conceptual model, in that “post-mortem” drilling revealed that injected grout had traveled along thin preferential pathways, no more than 5 cm (2 in) thick (SRNL-TR-2012-00160, Rev. 0), but presumably many meters wide and long, given the large volumes of grout injected. Grouting operations ceased, given that grout emplacement geometries of this nature would not likely affect settlement potential; however, these grouting operations demonstrated evidence of fast flow pathways in the UTRA-LZ through fluid losses and grout uptake. The geotechnical engineering definition of an SRS soft zone requires a continuous vertical thickness of at least 0.3 m (1 ft) (SRNL-TR-2012-00160, Rev. 0). However, this geotechnical definition is immaterial to the hydrogeologic concerns of fast flow and transport through soft zones in the UTRZ-LZ because fast flow and transport may occur through thinner soft zones than those identified according to the definition.

Dissolution of mixed carbonate/siliciclastic sediments by meteoric water under shallow vadose conditions within the units that now comprise the UTRA-LZ is thought to have caused (1) soft zone formation (Response to RAI-FF-1; SRR-CWDA-2011-00054, Rev. 1; cf. May and Warne, 1999), and (2) sediment volume loss and localized ground subsidence that formed sinkholes and Carolina bays at the land surface (Johnson, 1942; Siple, 1967). More recently, SRNL-TR-2012-00160, Rev. 0 indicates that Carolina bays are only of aeolian or lacustrine origin rather than a result of sinkhole development; however, this report provides no supporting evidence or reference for this conclusion. In contrast, siliciclastic-to-carbonate pseudokarst-to-karst

Figure 1-8 Mechanism of Soft Zone Formation Proposed by WSRC-TR-92-42-004



A. Deposition of carbonate by development of oyster banks.

B. Burial of abandoned banks with sand and clay.

C. Partial dissolution of carbonate.

D. Filling of moldic porosity with silts and partial collapse of the mound resulting in the formation of a soft zone.

development resulting in sinkhole formation has long been invoked as one of the most likely formation mechanisms of Carolina bays (e.g., Johnson, 1942; May and Warne, 1999). Once initiated through subsurface desilicification/dissolution processes, the resulting surficial Carolina bays are thought to have undergone modification by aeolian and lacustrine surficial processes. Johnson (1942) referred to these processes as the solution–lacustrine–aeolian hypothesis. During extensive field surveys, Johnson observed that every possible stage of transition from typical sinkhole to typical Carolina bay is evident in the region, often in close association.

Seismicity

Seismicity in the southeastern United States has been recorded for 300 years. Regionally, the record is dominated by the Charleston, South Carolina, earthquake on August 31, 1886 (estimated magnitude 7.0). Scientists collected pre-network, mostly qualitative seismic data from 1698 to 1974, after which earthquakes began being instrumentally recorded. Most seismic activity within a 50-km (30-mi) radius of the site is located to the east and southeast; reservoir-induced seismicity occurs northwest of the site (i.e., at Lakes Jocassee, Keowee, and Clarks Hill/Strom Thurmond) (WSRC-TR-95-0046). The most recent earthquakes within an 80-km

(50-mi) radius of the SRS were of magnitudes 2.8, 4.1, and 3.0 occurring on April 26, 2013 (USGS, 2013), February 15, 2014, and February 16, 2014 (USGS, 2014a; USGS, 2014b), respectively. The two greater than magnitude 3.0 (>M3) earthquakes in 2014 occurred approximately 72 km (45 mi) northwest of Aiken. Three other >M3 earthquakes occurred on August 14, 1972, October 10, 1974, and August 8, 1993 (WSRC-TR-95-0046). The August 8, 1993 earthquake (magnitude 3.2) had an epicenter approximately 12 km (7.4 mi) northeast of Aiken, South Carolina, and was most strongly felt near Couchton, South Carolina (WSRC-TR-95-0046). Four recorded earthquakes with epicenter locations within the SRS have occurred: (1) June 9, 1985 (magnitude 2.6); (2) August 5, 1988 (magnitude 2.0); (3) May 17, 1997 (magnitude 2.5); and (4) October 8, 2001 (magnitude 2.6). Strong motion accelerometers were not triggered as a result of these earthquakes.

1.1.1.2 Volcanism

SRS is not located within a region of active plate tectonics characterized by volcanism, thus, it is not an issue of concern.

1.2 HTF Closure Strategy

The HTF has 29 waste tanks that will be cleaned and closed in accordance with the SRS Federal Facility Agreement (FFA), a formal agreement between DOE, Region 4 of the EPA, and the SCDHEC. The FFA (WSRC-OS-94-42) establishes that, among other things, waste tanks that do not meet secondary containment standards (i.e., Type I, II, and IV tanks at HTF) must be removed from service according to the FFA schedule. The current FFA schedule calls for staggered operational closure of the 12 HTF and 10 FTF waste tanks that do not meet secondary containment standards by September 2022. In this regard, the current FFA requires the operational closure of four additional waste tanks by September 2015, which likely will be split between HTF and FTF. DOE addresses closure of the remaining HTF waste tanks (Type III/IIIA) and ancillary structures in the SRS *Liquid Waste Plan* (SRR-LWP-2009-00001, Rev. 17).

Tank cleaning involves stepwise processes of bulk waste removal, sludge removal, and heel removal using a series of mechanical and chemical techniques (DOE/SRS-WD-2013-001, Rev. 0). Tanks can be accessed 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. DOE will remove residual waste in tank annuli of Type I and II tanks using a baseline method of mechanical and wall crawler devices with washing capabilities (V-ESR-G-00003, Rev. 1). DOE anticipates that tanks with large deposits in the annulus (e.g., Tank 16) may require other methods of removal, if determined practical. DOE will continue efforts to sample and characterize waste remaining in the tanks after cleaning. Furthermore, they will continue to refine mapping techniques that allow accurate estimates of residual waste volume. DOE will flush and grout cooling coils and abandoned equipment in tanks 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. DOE will install an engineered closure cap over the HTF 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.

1.3 NRC Review Approach

The purpose of the DOE's HTF Performance Assessment (PA) (SRR-CWDA-2010-00128, Rev. 1) is to assess the long-term fate and transport of residual contamination in the environment and provide reasonable assurance that the closure of the HTF will protect human health and the environment into the future. The HTF PA is intended to support multiple decision documents, including the draft basis for a Section 3116 determination for closure of the HTF (DOE/SRS-WD-2013-001) per the Ronald Reagan National Defense Authorization Act of Fiscal Year 2005 (NDAA). There are three NDAA criteria that the Secretary of Energy must demonstrate are met in order to determine that the waste remaining in the tanks is not high-level waste (HLW). DOE's demonstration of Criterion 1, 2, and 3 and the NRC staff's evaluation of DOE's demonstration are discussed in detail in Chapters 2 through 4 of this TER, respectively. After the NRC staff completes their consultation on the draft basis for a Section 3116 waste determination (WD) for the HTF, DOE will complete their final basis for a Section 3116 determination for the HTF that the Secretary of Energy will use to develop the final WD for the HTF.

Prior to the NRC staff's review of DOE's draft basis document for the WD for HTF (DOE/SRS-WD-2013-001, Rev. 0) and its supporting references, the NRC staff, SCDHEC, and EPA representatives participated in a public scoping meeting (Lowman, 2010 [ML100970781]) with DOE regarding its development of the HTF PA. In developing the HTF PA, DOE incorporated improvements to account for insights gained from scoping and from previous PA developments, in particular, lessons learned from the NDAA Section 3116 consultation regarding the FTF WD and the accompanying FTF PA.

The NRC technical staff has conducted a detailed review of both the HTF PA (SRR-CWDA-2010-00128, Rev. 1) as well as DOE's draft basis document for the WD for HTF (DOE/SRS-WD-2013-001, Rev. 0). The NRC staff leveraged its previous review of the DOE FTF PA (SRS-REG-2007-00002, Rev. 1) and FTF draft basis document (DOE/SRS-WD-2010-001, Rev. 0), as documented in the FTF TER (Camper, 2011 [ML112371751]), to inform its review of the HTF PA and draft basis document for the WD for HTF. During the review for HTF, the NRC staff engaged DOE in a series of technical exchanges and public meetings from April through July 2013 (Shaffner, 2013a–f [ML13106A338], [ML13126A127], [ML13154A327], [ML13193A072], [ML13183A410], [ML13199A413]) to clarify DOE approaches and rationales documented in DOE's draft basis document for the WD for HTF (DOE/SRS-WD-2013-001, Rev. 0) and the HTF PA (SRR-CWDA-2010-00128, Rev. 1). The review also has been supplemented by confirmatory computations of various results using analytical software (e.g., PORFLOW™, Goldsim™, HELP). The NRC staff review process resulted in a number of requests for additional information (RAIs). Topics addressed in the RAIs are discussed elsewhere in this TER. RAIs were presented to DOE in July 2013 (Mohseni, 2013a [ML13196A135]). A public meeting took place in August 2013 that provided DOE an opportunity to better understand the purpose and intent of the NRC staff's RAIs (Shaffner, 2013e [ML13183A410]). DOE responded to the NRC staff's RAIs in November 2013 (SRR-CWDA-2013-00106, Rev. 1). As part of the process for completing this TER, on November 21, 2013, the NRC staff requested clarifications to gain a better understanding of some of DOE's RAI responses Shaffner, 2013g [ML14016A391]). DOE provided clarifications to the NRC in late January 2014 (SRR-CWDA-2013-00144, Rev. 0).

1.4 Prior Reviews

Prior to this review, the NRC staff assessed several draft basis documents. These assessments included evaluations of the methodologies DOE used to determine if residual tank waste had been properly classified as incidental. DOE used the draft basis documents to support WDs that residual tank waste had been properly classified as incidental. The following are summaries illustrating the NRC's involvement in these reviews:

SRS HLW Tank Closure: Classification of Residual Waste as Incidental (1999)

In December 1999, the NRC staff reviewed DOE's tank closure methodology to determine if residual waste (after cleaning) in HLW tanks at the SRS could be classified as incidental waste. The NRC staff determined that the waste could be considered incidental waste, in keeping with the criteria approved by the Commission in SRM-SECY-92-391, "SECY-92-391 - 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 WD.

As a result of the assessment, the NRC staff determined that two of the three criteria for classification of waste as incidental in the HLW tanks were satisfied. The requirements in Criterion 2, which state that waste in a solid form should not exceed the limits for Class C low-level waste (LLW), were not met. While many of the HLW 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 more stringent methods such as oxalic acid cleaning, to ensure that the residual waste did not exceed the Class C limitations. For this reason, in lieu of Criterion 2, DOE relied on an alternative classification consideration, 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 NRC staff concluded that this method was an acceptable means of meeting the concentration limitations for Class C waste as set forth in 10 CFR 61.55 and encouraged DOE to develop a site-specific alternative to Criterion 2, notwithstanding the requirements set forth in Criteria 1 and 3.

The NRC 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, the NRC staff, in response to a request from DOE, conducted a review to of DOE's draft basis document for the WD for salt waste disposal (DOE/SRS-WD-2005-001a) in accordance with the requirements in the NDAA. The NRC staff 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 Faciity (SDF). In the Salt Waste Disposal TER (Camper, 2005 [ML053010225]), the NRC staff presented information on DOE's salt waste processing strategy, DOE's review criteria and review approach, as well as the NRC staff's assessment 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 basis for the WD for salt waste disposal, the NRC staff concluded that there was reasonable assurance that the applicable criteria of the NDAA can be met, provided certain assumptions made in DOE's analyses are verified via monitoring. The NRC staff identified factors important to assessing compliance with 10 CFR Part 61, Subpart C, including improvements in future modeling and the associated support to justify the modeling predictions. Based on concerns identified in the assessment, the NRC staff also emphasized to DOE the importance of verifying the assumptions that DOE made in assessing whether the performance objectives of 10 CFR Part 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. The NRC staff went on to document, in a monitoring plan, the actions that the NRC staff would take to observe activities DOE would undertake to fulfill its responsibilities under the NDAA.

NRC Staff Review of the PA for the FTF Tanks 18 and 19 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 the NRC's review, per the requirements in the NDAA. As a result of the NRC staff's review, several issues were identified in the FTF PA, which prompted the staff to seek clarification from DOE through RAIs (Flanders, 2006 [ML060800295]).

Two years later, DOE requested that the NRC discontinue its review and consultation on this draft basis document (Marcinowski, 2007 [ML080090405]). DOE cited its decision to pursue a promising new technology for potentially removing additional waste from the tanks as its reason for requesting that the NRC discontinue its review. Additionally, DOE informed the NRC that it would develop a new PA for the entire FTF prior to closure of Tanks 18 and 19 at SRS. In this letter, DOE also notified the NRC that while it would not formally respond to the 50 open RAIs from the NRC staff that were submitted in March 2006; it would, however, consider the RAIs in developing the new basis document, as appropriate. The new PA for the FTF was submitted for the NRC staff's review in September 2010, as discussed below.

Technical Evaluation Report for the Draft Waste Determination for the FTF at the Savannah River Site (2011)

In September 2010, DOE submitted the *Draft Basis for Section 3116 Determination for Closure of F-Tank Farm at the Savannah River Site* for the NRC's review as required by the NDAA (DOE/SRS-WD-2010-001, Rev 0). As a result of the NRC staff's review and consultation with DOE, the NRC staff provided DOE a TER which identified several findings and recommendations (Camper, 2011 [ML112371751]) to address the limited amount of information regarding important factors influencing facility performance. Once addressed by DOE, the NRC staff expected those comments and recommendations would improve DOE's demonstration of compliance with the 10 CFR Part 61 performance objectives. After considering the NRC staff's recommendations and findings, DOE issued a final FTF basis document (DOE/SRS-WD-2012-001, Rev. 0) in March 2012. In accordance with the NDAA, following issuance of the final FTF basis document by DOE, the NRC will assess FTF compliance with the performance objectives in 10 CFR Part 61, Subpart C. The NRC staff developed a monitoring plan (Camper, 2013a [ML12345A322]) that details the NRC's path forward for assessing DOE's compliance with each

of the performance objectives for residual waste remaining in the waste tanks at the time of facility closure.

Technical Evaluation Report for the Revised Performance Assessment for the Saltstone Disposal Facility at the Savannah River Site (2012)

In November 2009, DOE submitted a PA for the SDF (SRR-CWDA-2009-00017, Rev. 0) for the NRC staff's review, as part of the NRC's monitoring role under the NDAA. The 2009 SDF PA is an update to DOE's PA approach supporting the 2006 basis document for salt waste disposal (DOE/SRS-WD-2005-001b) that includes new information about issues addressed in the 2005 SDF TER (Camper, 2005 [ML053010225]) and information about changes to the disposal cell design. DOE developed the revised SDF PA to demonstrate that its waste disposal strategy for the SDF remains in compliance with the performance objectives specified in 10 CFR Part 61, Subpart C. The NRC staff conducted a review and confirmatory analysis of the 2009 SDF PA and documented its results in a TER (Camper, 2012 [ML121020140]).

The NRC staff concluded in the SDF TER that it had reasonable assurance that waste disposal at the SDF meets the 10 CFR Part 61 performance objectives for protection of individuals against intrusion (10 CFR 61.42), protection of individuals during operations (10 CFR 61.43), and site stability (10 CFR 61.44). However, based on its evaluation of DOE's results and independent sensitivity analyses conducted with DOE's models, the NRC staff indicated that it no longer had reasonable assurance that DOE's disposal activities at the SDF meet the performance objective for protection of the general population from releases of radioactivity (10 CFR 61.41). Although the NRC staff could not conclude that the performance objective in 10 CFR 61.41 was met, based on DOE's results and the NRC staff's own independent analyses, the NRC staff indicated that the potential dose to an off-site member of the public from DOE's disposal actions was still expected to be relatively low (i.e., approximately 1 mSv/yr [100 mrem/yr], or equivalent to the public dose limit in 10 CFR 20.1301).

Following completion of the TER for the SDF, the NRC staff revised its monitoring plan (Camper, 2013b [ML13100A076]) for the SDF to include new monitoring factors resulting from its review findings. This monitoring plan will be used by the NRC staff to continue to assess DOE compliance with the performance objectives in 10 CFR Part 61, Subpart C, in fulfillment of the NRC monitoring responsibilities under the NDAA.

2. CRITERION 1

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

2.1 Tank Waste Disposal

Criterion 1 allows HLW to be disposed of in accordance with the risk posed by the waste 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 1for the waste at HTF.

3. CRITERION 2

The waste has had Highly Radioactive Radionuclides removed to the Maximum Extent Practical.

The NRC staff evaluated Criterion 2 by analyzing DOE's (1) methodology for developing radionuclide inventories for the tanks and auxiliary equipment; (2) process for identifying highly radioactive radionuclides (HRRs); (3) selection of waste treatment technology; and (4) demonstration of removal to the maximum extent practical (MEP), including analysis of the costs and benefits of additional radionuclide removal. For the purpose of reviewing the DOE basis documents for the WDs, the NRC's guidance defines HRRs as those radionuclides that contribute most significantly to risk to the public, workers, and the environment (NUREG-1854).

3.1 Waste Inventory

This section describes DOE's approach to determine radionuclide inventories for the tanks (including tank walls and in-tank components) and ancillary equipment. The inventory in each of the tanks: (1i) is used to demonstrate that the waste has had HRRs removed to the MEP (see Section 3.7), (2) determines whether the waste is greater than Class C (see Section 4.1), and (3) is used to develop the source term in the HTF 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 also 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 have been or will be developed for use in the HTF PA including (1) projected inventories for all tanks, and (2) final inventories for tanks once they have been cleaned and sampled. DOE considers the projected inventories to be risk-informed with integrated conservatisms, therefore, these projected inventories are not necessarily a reflection of what DOE expects will actually remain in the tanks. This section reviews DOE's basis for inventory projections. This section also details the approach DOE uses to estimate final inventories for tanks that have been cleaned (e.g., Tank 16) including DOE's sampling and analysis plan, methods used to determine residual waste volumes, and consideration of uncertainty in final inventory characterization.

3.1.1 Tank Projected Inventory

3.1.1.1 Screening Methodology for Radionuclides

DOE attempted to develop inventories that were both bounding and reasonable for waste tanks following operational closure. An initial screening approach (described in CBU-PIT-2005-00228, Rev . 0) narrowed the list of radionuclides to support characterization efforts and develop an initial inventory for use in the HTF PA modeling. This approach resulted in the screening of an initial list of 849 radionuclides down to 159 using information on the physical properties of each radionuclides such as half-life and decay mechanism, waste production and processing information, and screening factors for ground disposal of radionuclides developed in National Council on Radiation Protection and Measurements (NCRP)-123. The report SRR-CWDA-2010-00023, Rev. 3 summarizes the screening approach.

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

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

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

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

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

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

5. DOE eliminated those radionuclides 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 radionuclides 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 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 during the institutional control period) based on the age of the waste and the time to closure. This analysis considered decay and ingrowth 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 radionuclides following this initial screening, DOE eliminated another 105 radionuclides due to their decay behavior or because their half-life was less than 5 years. The details of why certain radionuclides were eliminated are discussed in the NRC staff's evaluation of the screening methodology in Section 3.2.1.1. Following the second screening, 54 radionuclides remained. The final list of 54 radionuclides is reproduced in Table 3-1.

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 HTF PA (SRR-CWDA-2010-00128, Rev. 1). In most cases, DOE made adjustments to the inventory based on tank type or radionuclide.

Table 3-1 Contaminants With an Initial Inventory

Ac-227	Cl-36	Eu-152	Pa-231	Ra-226	Th-232
Al-26	Cm-243	Eu-154	Pd-107	Ra-228	U-232
Am-241	Cm-244	H-3	Pt-193	Se-79	U-233
Am-242m	Cm-245	I-129	Pu-238	Sm-151	U-234
Am-243	Cm-247	K-40	Pu-239	Sn-126	U-235
Ba-137m	Cm-248	Nb-94	Pu-240	Sr-90	U-236
C-14	Co-60	Ni-59	Pu-241	Tc-99	U-238
Cf-249	Cs-135	Ni-63	Pu-242	Th-229	Y-90
Cf-251	Cs-137	Np-237	Pu-244	Th-230	Zr-93
Adapted from Table 3.3-1 in SRR-CWDA-2010-00128, Rev. 1.					

3.1.1.2 Inventory Starting Point

DOE used the WCS (see Section 2.2 of SRR-CWDA-2010-00023, Rev. 3) to develop the inventories for those radionuclides which it tracks. The 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, initially developed in 1995, tracks the concentrations of up to 40 radionuclides in each of the SRS waste tanks in each of three waste phases: sludge, salt, and liquid. The radionuclides tracked in the WCS were selected to assist the tank farm criticality analysis for operational safety rather than for developing inventories for tank farm closure. To develop an HTF inventory starting point for radionuclides tracked in the WCS, the concentration (in Ci/gal) tracked in the WCS is calculated by dividing the dry sludge activity (curie) by the corresponding settled sludge volume (gallon) both of which are extracted from the WCS.

In addition, DOE needed to estimate the initial inventories for additional radionuclides that were not historically tracked within the WCS in order to develop inventory starting points for tank closure. The starting inventories for some of the radionuclides that were not tracked historically in WCS were determined by special analyses described in SRR-LWE-2009-00014, Rev. 0. For the remaining radionuclides that were not historically tracked in the WCS (i.e., Cl-36, K-40, Pd-107, and Pt-193), DOE used the detection limits based on sampling from FTF Tanks 5, 18, and 19 to estimate their HTF inventory starting point (SRR-CWDA-2010-00023, Rev. 3). DOE also originally estimated Zr-93 this way, but then instead used a ratio to Sr-90 that was determined in process samples as described in Section 3.2.1.4 of this TER.

Furthermore, WCS does not report data in sludge for some radionuclides in specific HTF tanks. Therefore, DOE needed to develop an inventory starting point for radionuclides in these HTF tanks. Radionuclides for which inventory starting points were not available in the WCS for every HTF tank are: Ba-137m, H-3, Ra-226, Ra-228, Th-229, Th-230, Th-232, U-232, U-234, U-236, and Y-90. DOE developed initial inventory estimates for these radionuclides in HTF tanks for which data from WCS was lacking by using the average concentration from other tanks of similar tank type that contained similar waste. For example, WCS tracks some radionuclide inventory for Tanks 23, and 48, but not for all the radionuclides. Tanks 23 and 48 are somewhat unique due to the nature of the waste in the tanks and the tank strategic use in liquid waste operations. For instance, Tank 23 contains mostly salt waste, which differs from sludge or liquid supernate (WCS tracks sludge, salt, and liquid separately). WCS reports zero inventories for many sludge radionuclides in Tank 23. Therefore, DOE estimated initial inventories for these radionuclides using the average concentration from other tanks of similar tank type. In the case

of Tank 48, the waste it contains differs from the typical waste contained in other tanks because it contains the product of the tetraphenyl borate in-tank precipitation process operated in the 1990s. Furthermore, samples of this slurry were not analyzed for Tc-99, Pu-240, and Pu-241. Therefore, DOE uses the known concentrations of radionuclides that were analyzed in samples (i.e., Cs-137, U-233, U-234, and U-235) to compute concentrations of radionuclides that were not analyzed for in the samples (SRR-LWE-2011-00201, Rev. 0).

3.1.1.3 Tank Projected Volume

To develop inventories for the HTF PA, DOE assumes that approximately 15 m^3 (4,000 gal) of residuals would remain in each tank that has not yet been cleaned. This amount is based on DOE's prior experience with cleaning Tanks 5, 6, 16, 18, and 19 in which the residual material ranges from 0.8 m^3–14.7 m^3 (220–3,900 gal). Tank 16 is the only tank that has completed cleaning. Although DOE estimates that 0.8 m^3 (220 gallons) of residual material remains after tank cleaning, for the purposes of the HTF PA inventory DOE assumes that 4 m^3 (1,000 gal) remain in Tank 16. The volume of residue is multiplied by the concentrations in the WCS to obtain an inventory. DOE will estimate the actual inventory in tanks after they are cleaned, based on sampling and characterization data (SRR-CWDA-2010-00128, Rev. 1).

3.1.1.4 Inventory Adjustments

DOE grouped waste tanks according to tank type (Group 1: Type I & II, Group 2: Type III/IIIA, and Group 3: Type IV). Tank 16 is a special case with its own grouping. For Type III/IIIA tanks, the Pu-238 inventory grouping was split based on two different waste types (salt and sludge), as shown in Table 3-2.

Within each tank group, DOE adjusted the inventories as follows:

- For those radionuclides that have been observed to have greater impact on the overall dose, if the projected inventory was less than the detection limit, then DOE adjusted the inventory up to the analytical detection limit. For non-risk-significant radionuclides, any inventory less than 3.7×10^{10} Bq (1 Ci) was adjusted up to 3.7×10^{10} Bq (1 Ci).

- DOE applied the maximum *concentration* associated with an individual waste tank to the other tanks within the grouping to account for uncertainty surrounding future operations and movement of material within the HTF.

Table 3-2 Waste Tank Groupings (Salt and Sludge Grouping for Pu-238 Only)

Types I and II	Type III/IIIA		Type IV
	Salt	Sludge	
Tanks 9 and 10, 11, 12, 13, 14, and 15	Tanks 29, 30, 31, 36, 37, 38, 41, 48, 49, and 50	Tanks 32, 35, 39, 40, 42, 43, and 51	Tanks 21, 22, 23, and 24

Adapted from Table 3.4-8 in SRR-CWDA-2010-00128, Rev. 1.
Note: Tank 16 is a special case with its own grouping.

- DOE decreased the cesium, strontium, and zirconium inventories by one order of magnitude, based on expectations of the impact of chemical cleaning.

- Also based on the experience from Tank 5, DOE decreased the Tc-99 inventory by one order of magnitude and estimated the Zr-93 inventory by using a measured ratio of Zr-93 to Sr-90.

The applicable detection limits for certain radionuclides are listed in Table 3-3. DOE reviewed recent sample analyses from Tanks 5, 18, and 19 to apply the appropriate detection limits.

The report SRR-CWDA-2010-00023, Rev. 3 provides additional detail on the approach DOE uses to develop the inventory. Table 3.4-9 of the HTF PA lists the inventory projections based on expected radionuclide decay and in-growth to September 30, 2032 for each HTF tank (SRR-CWDA-2010-00128, Rev. 1). Table 3–4 of this TER summarizes the initial estimated inventory of the tanks. Inventories for Tanks 9, 29, 32, and 21 are provided for Type I & II, Type III/IIIA salt, Type III/IIIA sludge, and Type IV 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 because DOE uses two separate tank groupings (sludge and salt) for the Type III/IIIA tanks. Also note that Tank 16 is a Type II tank but is in its own grouping due to additional information available for Tank 16. The NRC staff notes that DOE used the incorrect inventory for Pu-238 for Tank 35. According to DOE's tank groups, Tank 35 should have a Pu-238 inventory equal to that of Tank 32 but instead, Tank 35 has an inventory the same as the salt Type III\IIIA tanks in the HTF PA (SRR-CWDA-2010-00128, Rev. 1).

Table 3-3 Radionuclide Detection Limits

Radionuclide	Detection Limit (µCi/g)*
Cl-36	1×10^{-04}
I-129	1×10^{-05}
K-40	5×10^{-05}
Nb-94	5×10^{-03}
Pa-231	1×10^{-04}
Pd-107	1×10^{-02}
Pt-193	1×10^{-02}
Ra-226	1×10^{-03}
Ra-228	1×10^{-01}
Th-229	1×10^{-04}
Th-230	1×10^{-03}
Th-232	1×10^{-03}
U-232	1×10^{-04}
U-235	1×10^{-03}
U-236	1×10^{-03}
U-238	1×10^{-04}

Adapted from SRR-CWDA-2010-00023, Rev. 3
* To convert µCi/g to Bq/g multiply by 3.7×10^{4}
Note: A density of 1.4 g/ml was used to estimate inventories.

Table 3-4 HTF Performance Assessment Estimated Radionuclide Inventory (Ci)* Decayed to 9/30/2032

Radionuclide	Type I/II (Tank 9)	Tank 16	Type III/IIIA salt (Tank 29)	Type III/IIIA sludge (Tank 32)	Type IV (Tank 21)
Ac-227	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0x10^{+00}$	$1.0x10^{+00}$
Al-26	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Am-241	$7.0\times10^{+02}$	$8.1\times10^{+01}$	$1.1\times10^{+03}$	$1.1\times10^{+03}$	$5.0\times10^{+00}$
Am-242m	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Am-243	$3.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Ba-137m	$7.4\times10^{+02}$	$1.2\times10^{+02}$	$5.2\times10^{+03}$	$5.2\times10^{+03}$	$2.3\times10^{+03}$
C-14	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cf-249	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cf-251	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cl-36	2.1×10^{-03}	5.3×10^{-04}	2.1×10^{-03}	2.1×10^{-03}	2.1×10^{-03}
Cm-243	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cm-244	$2.0\times10^{+01}$	$2.4\times10^{+00}$	$2.2\times10^{+03}$	$2.2\times10^{+03}$	$4.6\times10^{+00}$
Cm-245	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cm-247	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cm-248	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Co-60	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cs-135	5.4×10^{-03}	9.9×10^{-04}	7.1×10^{-03}	7.1×10^{-03}	2.3×10^{-02}
Cs-137	$7.9\times10^{+02}$	$1.3\times10^{+02}$	$5.5\times10^{+03}$	$5.5\times10^{+03}$	$2.4\times10^{+03}$
Eu-152	$2.1\times10^{+01}$	$1.0\times10^{+00}$	$3.8\times10^{+01}$	$3.8\times10^{+01}$	$1.0\times10^{+00}$
Eu-154	$2.9\times10^{+02}$	$3.3\times10^{+01}$	$9.2\times10^{+02}$	$9.2\times10^{+02}$	$8.3\times10^{+00}$
H-3	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
I-129	2.8×10^{-04}	5.3×10^{-05}	6.7×10^{-03}	6.7×10^{-03}	2.1×10^{-04}
K-40	1.1×10^{-03}	2.6×10^{-04}	1.1×10^{-03}	1.1×10^{-03}	1.1×10^{-03}
Nb-94	1.1×10^{-01}	2.6×10^{-02}	1.1×10^{-01}	1.1×10^{-01}	1.1×10^{-01}
Ni-59	$8.6\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Ni-63	$6.3\times10^{+02}$	$1.1\times10^{+02}$	$7.9\times10^{+02}$	$7.9\times10^{+02}$	$9.1\times10^{+00}$
Np-237	2.1×10^{-01}	2.2×10^{-02}	4.0×10^{-01}	4.0×10^{-01}	1.3×10^{-02}
Pa-231	2.1×10^{-03}	5.3×10^{-04}	2.1×10^{-03}	2.1×10^{-03}	2.1×10^{-03}
Pd-107	2.1×10^{-01}	5.3×10^{-02}	2.1×10^{-01}	2.1×10^{-01}	2.1×10^{-01}
Pt-193	2.1×10^{-01}	5.3×10^{-02}	2.1×10^{-01}	2.1×10^{-01}	2.1×10^{-01}
Pu-238	$6.5\times10^{+03}$	$2.9\times10^{+02}$	$2.8\times10^{+03}$	$1.5\times10^{+04}$	$7.2\times10^{+01}$
Pu-239	$8.0\times10^{+01}$	$7.7\times10^{+00}$	$2.4\times10^{+02}$	$2.4\times10^{+02}$	$1.0\times10^{+00}$
Pu-240	$5.0\times10^{+01}$	$3.7\times10^{+00}$	$1.5\times10^{+02}$	$1.5\times10^{+02}$	3.6×10^{-01}
Pu-241	$7.6\times10^{+02}$	$2.0\times10^{+01}$	$4.6\times10^{+03}$	$4.6\times10^{+03}$	$2.1\times10^{+00}$
Pu-242	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Pu-244	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Ra-226	2.1×10^{-02}	5.3×10^{-03}	2.1×10^{-02}	2.1×10^{-02}	2.1×10^{02}
Ra-228	$2.1\times10^{+00}$	5.3×10^{-01}	$2.1\times10^{+00}$	$2.1\times10^{+00}$	$2.1\times10^{+00}$
Se-79	$4.8\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Sm-151	$1.1\times10^{+04}$	$1.8\times10^{+03}$	$7.7\times10^{+04}$	$7.7\times10^{+04}$	$2.4\times10^{+02}$
Sn-126	$4.6\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$

Table 3-4 (continued) HTF Performance Assessment Estimated Radionuclide Inventory (Ci)* Decayed to 9/30/2032

Radionuclide	Type I/II (Tank 9)	Tank 16	Type III/IIIA salt (Tank 29)	Type III/IIIA sludge (Tank 32)	Type IV (Tank 21)
Sr-90	$1.4\times10^{+04}$	$2.2\times10^{+03}$	$2.0\times10^{+04}$	$2.0\times10^{+04}$	$3.1\times10^{+02}$
Tc-99	$8.1\times10^{+00}$	$1.5\times10^{+00}$	$9.7\times10^{+00}$	$9.7\times10^{+00}$	1.6×10^{-01}
Th-229	2.1×10^{-03}	5.3×10^{-04}	2.1×10^{-03}	2.1×10^{-03}	2.1×10^{-03}
Th-230	2.1×10^{-02}	5.3×10^{-03}	2.1×10^{-02}	2.1×10^{-02}	2.1×10^{-02}
Th-232	2.9×10^{-02}	5.3×10^{-03}	2.7×10^{-02}	2.7×10^{-02}	2.1×10^{-02}
U-232	2.1×10^{-03}	5.3×10^{-04}	2.1×10^{-03}	2.1×10^{-03}	2.1×10^{-03}
U-233	5.9×10^{-01}	8.7×10^{-02}	$1.3\times10^{+00}$	$1.3\times10^{+00}$	6.0×10^{-02}
U-234	9.6×10^{-02}	2.4×10^{-02}	6.6×10^{-01}	6.6×10^{-01}	2.2×10^{-02}
U-235	2.1×10^{-02}	5.3×10^{-03}	2.1×10^{-02}	2.1×10^{-02}	2.1×10^{-02}
U-236	2.1×10^{-02}	5.3×10^{-03}	1.1×10^{-01}	1.1×10^{-01}	2.1×10^{-02}
U-238	2.9×10^{-02}	5.3×10^{-04}	8.4×10^{-02}	8.4×10^{-02}	7.4×10^{-03}
Y-90	$1.4\times10^{+04}$	$2.2\times10^{+03}$	$2.0\times10^{+04}$	$2.0\times10^{+04}$	$3.1\times10^{+02}$
Zr-93	4.0×10^{-01}	6.3×10^{-02}	5.7×10^{-01}	5.7×10^{-01}	8.8×10^{-03}

Adapted from Table 3.4-9 in SRR-CWDA-2010-00128, Rev. 1.
* To convert Ci to Bq multiply by 3.7×10^{10}

3.1.1.5 Zeolite Assumptions

Given that spent zeolite was transferred to certain tanks, DOE also accounted for this zeolite in addition to the sludge material. DOE adjusted the projected radiological and chemical concentrations for Tanks 24, 32, 38, 40, 42, and 51 to account for the zeolite and corresponding cesium in the spent zeolite. The solids (sludge and zeolite) concentrations assume that zeolite ratio will remain unchanged during the waste removal processes (i.e., DOE assumes that the zeolite weight and volume fractions are the same in residual material after cleaning).

3.1.1.6 Cooling Coil and Tank Walls

Type I, Type II, and Type IIIA tanks have permanently installed cooling coils which form a dense array of piping at the bottom of the respective waste tank. There are approximately 7,000 linear m (22,800 linear ft) of 2-in (5-cm) carbon steel pipe cooling coils in a Type I tank and approximately 9,000 linear m (29,400 linear ft) of 2-in (5-cm) carbon steel pipe cooling coils in a Type II tank. The Type III tanks utilize deployable cooling coils that were inserted through the waste tank risers after final construction of the waste tanks, as opposed to permanently installed cooling coils. All Type IIIA tanks, except Tank 35, have permanently installed cooling coils similar to those in the Type I and II tanks. Like the Type III tanks, Tank 35 had deployable cooling coils installed after construction. Type IV tanks do not contain cooling coils. The cooling coils comprise a large surface area within the tank. For example, the surface area for Type I tank cooling coils is 20 percent greater than the surface area of the other internal tank surfaces (SRS-REG-2007-00002, Rev. 1).

For the HTF PA, DOE assumes the inventory of material that potentially remains inside failed cooling coils and on the surface of the waste tank walls, cooling coils, and columns is negligible compared to the estimated total tank inventories. DOE states that the cooling coils represent a small percentage of the in-tank volume and that cooling coils with potential for waste build up

will be flushed to minimize residual waste holdup. Flushing will include both inlet and outlets to ensure that both sides are flushed. 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, are expected to be rinsed with oxalic acid, with the exception of Type IV Tanks.

3.1.1.7 Tank Inventory Uncertainty

The uncertainty analysis in the HTF PA (SRR-CWDA-2010-00128, Rev. 1) includes estimates of inventory uncertainty based on tank type and radionuclide (see Table 9.1-1 in SRR-CWDA-2010-00023, Rev. 3 for more information on how these distributions were developed). DOE assumes a log uniform distribution that ranges from a factor of 100 times less to a factor of 10 times greater than the inventory, depending on tank type and radionuclide. The distribution for the non-detected radionuclides whose inventories were adjusted to the detection limit ranged from 100 times less up to the assumed detection limit, but inventories greater than the detection limit for these radionuclides were not considered in the uncertainty analysis.

3.1.2 Annular and Sand Pad Projected Inventory

The estimated annulus inventories are presented in Table 3-5. DOE assumes one of three values for the volume of material in the annulus of tanks depending on tank type and history of leaks. DOE assumes zero annulus inventories for Type III/IIIA and for Type IV Tanks. For Type I and Type II tanks, DOE assumes the volume of residual material in the annulus is either 5 m^3 (3,300 gal) for tanks that have had significant leaks (i.e., equivalent to the amount of material currently estimated to be in Tank 16 annulus) or 0.378 m^3 (100 gal) for tanks that had less significant leaks. The volume assumptions are shown in Table 3-5.

The annulus concentrations are based primarily on four samples that have been taken from the Tank 16 annulus. Since the previous samples were not analyzed for all the HRRs, the remaining radionuclides were estimated using one of three approaches. The approach taken for each constituent in the primary residual inventory estimate determined the method to estimate each constituent's concentration. First, if the Tank 16 primary tank inventory was estimated using the detection limit, annulus concentrations were estimated by taking a ratio of the primary residual volume estimate to the annulus residual volume estimate. For example, Cl-36 was estimated in the Tank 16 primary residual material via a detection limit. Therefore, a ratio of the Tank 16 primary residual volume estimate (1,000 gal) to the annulus residual volume estimate (3,300 gal) was multiplied by the Cl-36 primary inventory to determine the annulus inventory. Second, if the Tank 16 primary inventory was assumed to be 3.7×10^{10} Bq (1 Ci), the inventory in the annulus was also assumed to be 3.7×10^{10} Bq (1 Ci). A third approach was taken for risk significant radionuclides that were not analyzed in the annulus samples and were not estimated via the detection limit used to estimate the Tank 16 primary inventory. Concentrations of these radionuclides were based on a ratio of a chemically similar element and the tank primary floor residual. A ratio to the Pu-238 analysis was used to estimate insoluble radionuclides, while a ratio for the Tc-99 analysis was used to estimate for the soluble radionuclides. For example, Am-241 was not analyzed in the Tank 16 annulus samples. The Am-241 annulus concentration was estimated by multiplying the Pu-238 Tank 16 annulus sample concentration by a ratio of the Am-241 to Pu-238 primary inventories.

Table 3-5 Estimated Annulus Radiological Inventories (2032) (Ci)*

Radionuclide	Tank 9	Tank 10	Tank 11	Tank 12	Tank 13	Tank 14	Tank 15	Tank 16
Ac-227	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Al-26	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Am-241	$7.0\times10^{+00}$	$7.0\times10^{+00}$	2.1×10^{-01}	2.1×10^{-01}	2.1×10^{-01}	$7.0\times10^{+00}$	2.1×10^{-01}	$7.0\times10^{+00}$
Am-242m	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Am-243	$3.0\times10^{+00}$	$3.0\times10^{+00}$	$3.0\times10^{+00}$	$3.0\times10^{+00}$	$3.0\times10^{+00}$	$3.0\times10^{+00}$	$3.0\times10^{+00}$	$1.0\times10^{+00}$
Ba-137m	$1.1\times10^{+04}$	$1.1\times10^{+04}$	$3.5\times10^{+02}$	$3.5\times10^{+02}$	$3.5\times10^{+02}$	$1.1\times10^{+04}$	$3.5\times10^{+02}$	$1.1\times10^{+04}$
C-14	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cf-249	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cf-251	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cl-36	1.7×10^{-03}	1.7×10^{-03}	5.3×10^{-05}	5.3×10^{-05}	5.3×10^{-05}	1.7×10^{-03}	5.3×10^{-05}	1.7×10^{-03}
Cm-243	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cm-244	2.1×10^{-01}	2.1×10^{-01}	6.4×10^{-03}	6.4×10^{-03}	6.4×10^{-03}	2.1×10^{-01}	6.4×10^{-03}	2.1×10^{-01}
Cm-245	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cm-247	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cm-248	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Co-60	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Cs-135	3.2×10^{-03}	3.2×10^{-03}	9.8×10^{-05}	9.8×10^{-05}	9.8×10^{-05}	3.2×10^{-03}	9.8×10^{-05}	3.2×10^{-03}
Cs-137	$1.2\times10^{+04}$	$1.2\times10^{+04}$	$3.7\times10^{+02}$	$3.7\times10^{+02}$	$3.7\times10^{+02}$	$1.2\times10^{+04}$	$3.7\times10^{+02}$	$1.2\times10^{+04}$
Eu-152	$2.1\times10^{+01}$	$2.1\times10^{+01}$	$2.1\times10^{+01}$	$2.1\times10^{+01}$	$2.1\times10^{+01}$	$2.1\times10^{+01}$	$2.1\times10^{+01}$	$1.0\times10^{+00}$
Eu-154	$2.9\times10^{+00}$	$2.9\times10^{+00}$	8.8×10^{-02}	8.8×10^{-02}	8.8×10^{-02}	$2.9\times10^{+00}$	8.8×10^{-02}	$2.9\times10^{+00}$
H-3	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0x10^{+00}$	$1.0x10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
I-129	1.7×10^{-04}	1.7×10^{-04}	5.3×10^{-06}	5.3×10^{-06}	5.3×10^{-06}	1.7×10^{-04}	5.3×10^{-06}	1.7×10^{-04}
K-40	8.7×10^{-04}	8.7×10^{-04}	2.6×10^{-05}	2.6×10^{-05}	2.6×10^{-05}	8.7×10^{-04}	2.6×10^{-05}	8.7×10^{-04}
Nb-94	8.7×10^{-02}	8.7×10^{-02}	2.6×10^{-03}	2.6×10^{-03}	2.6×10^{-03}	8.7×10^{-02}	2.6×10^{-03}	8.7×10^{-02}
Ni-59	$8.6\times10^{+00}$	$8.6\times10^{+00}$	$8.6\times10^{+00}$	$8.6\times10^{+00}$	$8.6\times10^{+00}$	$8.6\times10^{+00}$	$8.6\times10^{+00}$	$1.0\times10^{+00}$
Ni-63	$9.6\times10^{+00}$	$9.6\times10^{+00}$	2.9×10^{-01}	2.9×10^{-01}	2.9×10^{-01}	$9.6\times10^{+00}$	2.9×10^{-01}	$9.6\times10^{+00}$
Np-237	2.6×10^{-02}	2.6×10^{-02}	7.9×10^{-04}	7.9×10^{-04}	7.9×10^{-04}	2.6×10^{-02}	7.9×10^{-04}	2.6×10^{-02}
Pa-231	1.7×10^{-03}	1.7×10^{-03}	5.3×10^{-05}	5.3×10^{-05}	5.3×10^{-05}	1.7×10^{-03}	5.3×10^{-05}	1.7×10^{-03}
Pd-107	1.7×10^{-01}	1.7×10^{-01}	5.3×10^{-03}	5.3×10^{-03}	5.3×10^{-03}	1.7×10^{-01}	5.3×10^{-03}	1.7×10^{-01}
Pt-193	1.7×10^{-01}	1.7×10^{-01}	5.3×10^{-03}	5.3×10^{-03}	5.3×10^{-03}	1.7×10^{-01}	5.3×10^{-03}	1.7×10^{-01}
Pu-238	$2.5\times10^{+01}$	$2.5\times10^{+01}$	7.6×10^{-01}	7.6×10^{-01}	7.6×10^{-01}	$2.5\times10^{+01}$	7.6×10^{-01}	$2.5\times10^{+01}$
Pu-239	$3.6\times10^{+00}$	$3.6\times10^{+00}$	1.1×10^{-01}	1.1×10^{-01}	1.1×10^{-01}	$3.6\times10^{+00}$	1.1×10^{-01}	$3.6\times10^{+00}$
Pu-240	$4.2\times10^{+00}$	$4.2\times10^{+00}$	1.3×10^{-01}	1.3×10^{-01}	1.3×10^{-01}	$4.2\times10^{+00}$	1.3×10^{-01}	$4.2\times10^{+00}$
Pu-241	$1.3\times10^{+01}$	$1.3\times10^{+01}$	3.9×10^{-01}	3.9×10^{-01}	3.9×10^{-01}	$1.3\times10^{+01}$	3.9×10^{-01}	$1.3\times10^{+01}$
Pu-242	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Pu-244	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$	$1.0\times10^{+00}$
Ra-226	1.7×10^{-02}	1.7×10^{-02}	5.3×10^{-04}	5.3×10^{-04}	5.3×10^{-04}	1.7×10^{-02}	5.3×10^{-04}	1.7×10^{-02}
Ra-228	$1.7\times10^{+00}$	$1.7\times10^{+00}$	5.3×10^{-02}	5.3×10^{-02}	5.3×10^{-02}	$1.7\times10^{+00}$	5.3×10^{-02}	$1.7\times10^{+00}$
Se-79	$4.8\times10^{+00}$	$4.8\times10^{+00}$	$4.8\times10^{+00}$	$4.8\times10^{+00}$	$4.8\times10^{+00}$	$4.8\times10^{+00}$	$4.8\times10^{+00}$	$1.0\times10^{+00}$
Sm-151	$1.5\times10^{+02}$	$1.5\times10^{+02}$	$4.7\times10^{+00}$	$4.7\times10^{+00}$	$4.7\times10^{+00}$	$1.5\times10^{+02}$	$4.7\times10^{+00}$	$1.5\times10^{+02}$

Table 3-5 (continued) Estimated Annulus Radiological Inventories (2032) (Ci)*

Radionuclide	Tank 9	Tank 10	Tank 11	Tank 12	Tank 13	Tank 14	Tank 15	Tank 16
Sn-126	$4.6\times10^{+00}$	$4.6\times10^{+00}$	$4.6\times10^{+00}$	$4.6\times10^{+00}$	$4.6\times10^{+00}$	$4.6\times10^{+00}$	$4.6\times10^{+00}$	$1.0\times10^{+00}$
Sr-90	$7.8\times10^{+03}$	$7.8\times10^{+03}$	$2.4\times10^{+02}$	$2.4\times10^{+02}$	$2.4\times10^{+02}$	$7.8\times10^{+03}$	$2.4\times10^{+02}$	$7.8\times10^{+03}$
Tc-99	$4.9\times10^{+00}$	$4.9\times10^{+00}$	1.5×10^{-01}	1.5×10^{-01}	1.5×10^{-01}	$4.9\times10^{+00}$	1.5×10^{-01}	$4.9\times10^{+00}$
Th-229	1.7×10^{-03}	1.7×10^{-03}	5.3×10^{-05}	5.3×10^{-05}	5.3×10^{-05}	1.7×10^{-03}	5.3×10^{-05}	1.7×10^{-03}
Th-230	1.7×10^{-02}	1.7×10^{-02}	5.3×10^{-04}	5.3×10^{-04}	5.3×10^{-04}	1.7×10^{-02}	5.3×10^{-04}	1.7×10^{-02}
Th-232	2.4×10^{-02}	2.4×10^{-02}	7.1×10^{-04}	7.1×10^{-04}	7.1×10^{-04}	2.4×10^{-02}	7.1×10^{-04}	1.7×10^{-02}
U-232	1.7×10^{-03}	1.7×10^{-03}	5.3×10^{-05}	5.3×10^{-05}	5.3×10^{-05}	1.7×10^{-03}	5.3×10^{-05}	1.7×10^{-03}
U-233	1.4×10^{-01}	1.4×10^{-01}	4.3×10^{-03}	4.3×10^{-03}	4.3×10^{-03}	1.4×10^{-01}	4.3×10^{-03}	1.4×10^{-01}
U-234	9.1×10^{-02}	9.1×10^{-02}	2.8×10^{-03}	2.8×10^{-03}	2.8×10^{-03}	9.1×10^{-02}	2.8×10^{-03}	9.1×10^{-02}
U-235	2.6×10^{-04}	2.6×10^{-04}	7.9×10^{-06}	7.9×10^{-06}	7.9×10^{-06}	2.6×10^{-04}	7.9×10^{-06}	2.6×10^{-04}
U-236	1.2×10^{-03}	1.2×10^{-03}	3.6×10^{-05}	3.6×10^{-05}	3.6×10^{-05}	1.2×10^{-03}	3.6×10^{-05}	1.2×10^{-03}
U-238	1.0×10^{-03}	1.0×10^{-03}	3.2×10^{-05}	3.2×10^{-05}	3.2×10^{-05}	1.0×10^{-03}	3.2×10^{-05}	1.0×10^{-03}
Y-90	$7.8\times10^{+03}$	$7.8\times10^{+03}$	$2.4\times10^{+02}$	$2.4\times10^{+02}$	$2.4\times10^{+02}$	$7.8\times10^{+03}$	$2.4\times10^{+02}$	$7.8\times10^{+03}$
Zr-93	5.5×10^{-03}	5.5×10^{-03}	1.7×10^{-04}	1.7×10^{-04}	1.7×10^{-04}	5.5×10^{-03}	1.7×10^{-04}	5.5×10^{-03}

Adapted from Table 3.4-3 in SRR-CWDA-2010-00128, Rev. 1.
* To convert Ci to Bq multiply by 3.7×10^{10}

Type II tanks have a primary sandpad layer in the annulus between the primary and secondary liner and a secondary sandpad layer between the secondary liner and the basemat. Each sandpad layer is 1-in thick. The concentration of the material in each sandpad layer is assumed to have the same concentration as the annulus material. DOE assumes that 0.378 m^3 (100 gal) is in the primary sandpad layer for Tank 13 and Tank 15 because it believes the amount of material that leaked from the primary tank for these tanks is limited, and therefore, not much material reached the sandpad. For Tanks 14 and 16, which have a greater depth of material in the annulus, DOE assumes that the primary sandpad layer is saturated with residual material. This equates to about 5 m^3 (1,350 gal), assuming the area of the tank floor and porosity provided in the HTF PA of 38 percent (SRR-CWDA-2010-00128, Rev. 1). For Tank 16 only, DOE assumes that 100 L (26 gal) of material is in the secondary sandpad layer. No inventory is assumed for the secondary sandpad layer for Tanks 13, 14, and 15. Sandpad inventories are presented in Table 3.4-5 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1). Annulus and sandpad inventories are reproduced in Table 3-6.

Table 3-6 Residual Annulus Volume Estimates for Type I and Type II Tanks (gal)*

Tank	Annulus (gal)	Primary Sandpad (gal)	Secondary Sandpad (gal)
Tank 9	3,300	N/A	N/A
Tank 10	3,300	N/A	N/A
Tank 11	100	N/A	N/A
Tank 12	100	N/A	N/A
Tank 13	100	100	0
Tank 14	3,300	1,350	0
Tank 15	100	100	0
Tank 16	3,300	1,350	26

* To convert gal to m^3, multiply by 3.8×10^{-3}

3.1.3 Ancillary Equipment Projected 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: transfer lines, transfer line secondary containment, diversion boxes, valve boxes, pump tanks, pump pits, the catch tank, concentrate transfer system (CTS) tanks, and the evaporator systems. The following provides a brief description of the equipment:

- The HTF has more than 22,800 m (74,800 ft) of transfer line, typically made of stainless steel and located below ground.

- 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 nine HTF pump tanks received waste transfers from the H-Canyon Facility.

- Ten pump pits, which each house a pump tank, are reinforced concrete structures located below grade at low points of the transfer lines. (Note that the pump pit identified as HPP-1 does not contain a pump tank).

- The single 44,300-L (11,700-gal) HTF catch tank is designed to collect drainage from the diversion box and Type I tank transfer line encasements. No significant contamination has been collected in this catch tank. Therefore, DOE did not assign any inventory to this catch tank.

- The two CTS tanks (old and new) that were used to facilitate transfers of the concentrate from the evaporator to selected waste tanks. A second CTS tank was needed to replace the original one to accommodate additional waste tanks.

- The three evaporator systems were used to reduce the amount of liquid volume of radioactive waste resulting from nuclear processes.

DOE developed projected inventories expected to remain in ancillary equipment specifically the transfer lines, pump tanks, CTS tanks, and evaporators vessels. Secondary containments for transfer lines and pumps, as well as diversion and valve boxes are assumed to have insignificant inventories such that the inventory of the other ancillary equipment bounds the impacts of these components. No leakage of waste from primary core pipe into secondary containment has been identified.

Dry sludge concentrations are 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 reduces 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, DOE considers the use of a slightly higher percentage of the dry sludge concentrations to be reasonable. DOE calculates a weighted average of tank concentrations that contributed to the inventory for any particular transfer line segment.

DOE assumes 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 applies an exponential decay curve with respect to time to estimate the residue remaining after the flushes. Table 3.4-11 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1) lists the estimated inventories of the residual contaminants in the transfer lines after flushing three times at the time of closure.

All of the pump tanks are accessible for waste removal, therefore,DOE assumes only residue is left behind after rinsing and flushing for the inventory of pump tanks. Table 3.4-13 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1) lists the inventory for the pump tanks and the CTS tanks. DOE estimates the inventory of the CTS tanks to be the same as the pump tanks since the CTS tanks are comparable in capacity. DOE estimated the evaporator system inventory based on characterization data for the FTF 242-F evaporator since field characterization data were not available for the HTF evaporator.

3.1.4 Inventory Final Characterization

After each tank is cleaned, DOE will determine the amount of residual material remaining in the tanks and annuli using visual inspection techniques to estimate volumes and sampling to estimate concentrations. DOE has not completed final characterization of any HTF tanks. DOE has completed the final characterization of FTF Tanks 18 and 19 (SRR-CWDA-2011-00091, Rev. 0) and Tanks 5 and 6 (SRR-CWDA-2012-00071, Rev. 0). The process for final characterization for the HTF tanks will be similar to that used for Tanks 5 and 6. DOE will provide justification for those radionuclides not included in the sampling analyses (e.g., lack of risk significance, not detectable, below detection limits). DOE will determine inventories for those radionuclides via special methods (e.g., ratios to other radionuclides or fission yields) in order to conduct an appropriate comparison to the PA modeled residual inventory.

3.1.4.1 Final Volume Determination

DOE has developed an approach, called “tank mapping”, for determining the final volume of material in a tank after heel removal. DOE provided formal documentation of this approach to the NRC staff in support of the FTF monitoring activities, and plans to use this approach for HTF tanks (SRR-LWE-2010-00240, Rev. 1). DOE relies on photographs and video footage along with landmarks of known height to estimate residual volume. A team of individuals visually compares the level of the waste with known landmarks and assigns a height level for areas of similar height. Landmarks depend on the tank type. Examples of known landmarks for Type I tanks include: 1.27 cm (0.5 in) thick lifting plates, the vertical cooling coils that are about 20.32 cm (8 in) from the tank floor, and lower column details that are 11.43 cm (4.5 in) high.

If landmarks are not available, DOE can insert other landmarks with known dimensions (e.g., a measuring pole, or a robotic crawler device) to collect additional height measurements. For

some landmarks, specific criteria are available to guide assignment of heights. For example, using the lifting plate as a landmark, DOE assigns the following heights:

- 0.125 in (0.3175 cm) if a dusting of solids is evident with some clean steel floor visible;
- 0.25 in (0.635 cm) if the sides of the lifting plate are visible;
- 0.5 in (1.24 cm) if the shape of the lifting plate is clearly visible but the material appears to be the same height as the top of the lifting plate; or
- 0.75 in (1.905 cm) if the shape of lifting plate can be discerned through solids.

After DOE maps areas of similar height across the tank bottom, the data are transferred to an Excel spreadsheet using a structured grid with a discretization of 1 ft (0.3 m) × 1 ft (0.3 m). The mapping team assesses uncertainty by assigning a low end value, a high end value, as well as the best estimate of height for each area to calculate the residual solid volume.

Presently, DOE has not yet completed a final volume determination for any of the HTF tanks, including Tank 16. DOE currently estimates that 0.8 m^3 (220 gal) of residual solids remain in the primary liner of Tank 16. However, DOE conservatively assumes 4 m^3 (1,000 gal) of residual solids remain for the inventory in the HTF PA (SRR-CWDA-2010-00023, Rev. 3). DOE estimated that 12.5 m^3 (3,300 gallons) remain in the Tank 16 annulus (SRR-LWE-2012-00039, Rev. 0). These are the most current estimates using visual observation and volume mapping techniques, but the final volume determination will be provided as part of the Tank 16 Closure Module.

3.1.4.2 Final Concentration Sampling Approach

DOE developed a Liquid Waste Sampling and Analysis Plan which will be used for HTF sampling (SRR-CWDA-2011-00050, Rev. 2). Based on the approaches identified in SRR-CWDA-2011-00050, Rev. 2, DOE developed a specific sampling and analysis plan for Tank 16, which was presented to the State of South Carolina (SRR-LWE-2013-00057, Rev. 0).

For the Tank 16 primary liner, DOE plans to analyze three discrete samples. DOE indicates that three discrete samples are appropriate for cases where only minimal material is left for characterization and the volume-proportional compositing sampling approach cannot be implemented. DOE also states that the three sample locations chosen provide the greatest opportunity for material collection and satisfy the requirement for representativeness. While the plan is to collect three samples, the actual amount of material recovered will determine whether discrete, composite, or some combination of discrete and composite sample analyses will be performed.

For the Tank 16 annulus, DOE proposes to collect a total of 15 samples. In 2011, samples of the Tank 16 annulus material were collected from the Tank 16 N, S, E, and W annulus risers. Since these four samples were determined to be acceptable for the current characterization effort, DOE proposes to collect 11 additional samples. Six of the 11 samples will be collected from the material inside the dehumidification duct. Five of the 11 samples will be collected outside the duct. The eleven new samples will be collected from nine accessible annulus riser and inspection port locations. The eleven new samples and four prior samples will be used to create the three 70-gram composites, each consisting of five or six samples. The amount of material from each sample will be volume proportional and informed by the final volume determination and uncertainty estimate. Each of the three composites is planned to be analyzed in triplicate.

3.2 NRC Evaluation of Waste Inventory

This section discusses inventory with respect to the NRC staff's evaluation of compliance with Criterion 2 (see Chapter 3 of this TER for a description of Criterion 2). Section 4.2.7 discusses the NRC staff's evaluation of inventory as it pertains specifically to the Criterion 3 evaluation. It is important to note subtle differences in the NRC staff's approach to evaluating DOE's inventory when considering Criterion 2 versus Criterion 3 (see Chapter 4 of this TER for a description of Criterion 3). For example, because the inventory for HTF tanks and auxiliary components that have not been cleaned and characterized cannot be known at this time, the cleaning effectiveness of the final inventory must be estimated for the purposes of demonstrating removal to the MEP (Criterion 2). It would follow that the same estimation 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 that 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 the results of the PA are relatively insensitive to the inventory (e.g., because the costs of additional removal are relatively high or because most important 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 overestimating the inventory for the Criterion 3 evaluation is more conservative (i.e., 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 underestimate 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 a justification to terminate cleaning operations earlier than necessary. 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 overestimate the potential benefits associated with additional radionuclide removal, resulting in the consideration of additional cleaning technologies and waste retrieval that may not be cost beneficial.

DOE attempts to overestimate 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 cleanup will proceed until it is no longer practical to do so. For the purpose of the Criterion 2 inventory evaluation, the NRC will focus on DOE's approach for final inventory estimates, as well as the experience gained thus far for those FTF tanks that have been characterized (Tanks 18-19 and Tanks 5-6).

3.2.1 NRC Evaluation of Tank Projected Inventory

3.2.1.1 NRC Evaluation of Radionuclide Screening Methodology

The screening methodology applied for HTF is largely similar to that used for FTF with the following key differences:

- Nb-93m, Sb-126, and Sb-126m were part of the FTF list, but were eliminated from the HTF list.

- Cf-251, Ra-228, and Th-232 were eliminated from the FTF list, but remained for HTF.

In a follow-up to the May 16, 2013, teleconference with the the NRC staff, DOE explained the secondary screening steps that reduced the 159 radionuclides to the final 54 radionuclides (SRR-CWDA-2013-00086, Rev. 0). The 159 radionuclides were screened per the process described in Appendix B of SRR-CWDA-2010-00023, Rev. 3, which eliminated radionuclides based on short half-life, the presence or absence of parent radionuclides, and the expected amount present in the waste inventory. The rationale described in Appendix B eliminated 89 isotopes from the list of 159 (in the May 16, 2013, teleconference DOE corrected that the text in Appendix B inaccurately states that 90 instead of 89 radionuclides were eliminated). Table 3-7 lists the 89 radionuclides eliminated based on the rationale in Appendix B of SRR-CWDA-2010-00023, Rev. 3. In addition to the 89 radionuclides eliminated, the following three daughters were also eliminated because they are only present as part of a decay chain: Nb-93m, Sb-126, and Sb-126m.

Table 3-7 Radionuclides Eliminated Per Appendix B of SRR-CWDA-2010-00023, Rev. 3

Ac-225	Bi-210m	Cm-250	Ho-166m	Np-240	Pm-145	Pu-243	Si-32	Th-228
Ac-228	Bi-211	Co-60m	Ir-192	Pa-233	Po-210	Pu-246	Sm-146	Th-231
Ag-108m	Bi-212	Eu-150	Ir-192m	Pa-234	Po-211	Ra-223	Sm-147	Th-234
Am-242	Bi-213	Fe-60	La-137	Pb-202	Po-212	Ra-224	Sn-121m	Ti-44
Am-246	Bi-214	Fr-221	La-138	Pb-205	Po-213	Ra-225	Ta-182	Tl-207
At-217	Bk-247	Fr-223	Lu-176	Pb-209	Po-214	Rb-87	Tb-157	Tl-208
At-218	Bk-250	Gd-148	Mn-53	Pb-210	Po-215	Re-186m	Tb-158	Tl-209
Be-10	Ca-41	Hf-178m	Mo-93	Pb-211	Po-216	Rn-219	Tc-97	Tl-210
Bi-207	Cf-250	Hf-182	Np-236	Pb-212	Po-218	Rn-220	Tc-98	U-240
Bi-210	Cm-246	Hg-194	Np-239	Pb-214	Pu-236	Rn-222	Th-227	
Adapted from Table A3-1 of SRR-CWDA-2013-00086, Rev. 0.								

DOE also explained in a follow-up to the May 16, 2013, teleconference that a description of the final elimination step was inadvertently left out of SRR-CWDA-2010-00023, Rev. 3 (SRR-CWDA-2013-00086, Rev. 0). As the final step, DOE removed radionuclides with a half-life of less than five years from inventory estimates. This was based on the consideration that active institutional control over the closed waste tanks will be maintained for 100 years following closure of the tank farm. Therefore, any radionuclides with less than a 5-year half-life would decrease to insignificant levels due to radioactive decay during the institutional control period. This elimination did not apply to radionuclides that are part of a decay chain with a parent having a greater than 5-year half-life. Although these radionuclide inventories were not estimated, they will continue to be included in dose modeling provided one of their ancestors is included. The 13 radionuclides removed from further inventory

estimates based on the 5-year half-life criteria are listed in Table 3-8 (SRR-CWDA-2013-00086, Rev. 0).

DOE did not eliminate Ra-228 and Th-232 for HTF because thorium fuel was exclusively processed at the H-Canyon. The waste removed from individual waste tanks travels different paths to the SDF and the Defense Waste Processing Facility (DWPF) depending on the originating waste tank farm. Material originating in HTF will be sent to SDF and DWPF from HTF; whereas the waste removed from waste tanks in FTF will pass through a waste tank(s) within HTF. Since the two tank farms were fed from different processing facilities and the SDF receives material originating from both the FTF and HTF, differences between the FTF and HTF inventories are to be expected (SRNL-STI-2012-00479, Rev. 0). The NRC staff has reviewed the screening approach and finds the list of radionuclides for which DOE has developed inventories to be reasonable.

Table 3-8 Radionuclides Eliminated Based on Half-life Less Than 5 Years

Bk-249	Cm-242	Na-22	Rh-106	Te-125m
Ce-144	Cs-134	Pm-147	Ru-106	
Cf-252	Eu-155	Pr-144	Sb-125	

3.2.1.2 NRC Evaluation of Inventory Starting Point

Since the WCS did not track all constituents for all tanks, the following radionuclides were estimated using other methods for certain tanks: Ba-137m, Cl-36, H-3, K-40, Pd-107, Pt-193, Ra-226, Ra-228, Th-229, Th-230, Th-232, U-232, U-234, U-236, Y-90, and Zr-93. The methods used to estimate some of the radionuclides not tracked by the WCS are described in Table 3-9. For those radionuclides not listed in Table 3-9, DOE applies an average concentration of similar tank types and materials. The NRC staff notes that by averaging across similar tank types, DOE may have averaged FTF and HTF tank inventories in estimating some of these constituents.

Table 3-9 Initial Concentration Estimate Method

Constituent	Estimate Method
Ba-137m	Secular equilibrium with Cs-137 based on its decay chain
C-14	An average C-14 HTF concentration
H-3	Interstitial liquid concentration
Y-90	Secular equilibrium with Sr-90
Adapted from Table 2.2-1 in SRR-CWDA-2010-00023, Rev. 3.	

The NRC staff noted during its review that there are chemical differences in processes between HTF and FTF that are not reflected in their projected inventories (RAI-INV-3; Mohseni, 2013a [ML13196A135]). The HTF and FTF processed different waste streams, which would lead to expected differences in the inventories. The waste processed at the FTF resulted from the plutonium-uranium redox extraction (PUREX) processing performed to recover uranium and plutonium from irradiated depleted uranium targets. The waste solids received into the HTF were from PUREX processing and the H-modified (HM) processing performed to recover uranium and neptunium from burned enriched uranium fuel.

The HTF PA estimates for some radionuclides (e.g., U-234, U-236) do not appear to agree with expectations provided in SRNL-STI-2012-00479, Rev. 0, which documents chemical differences between sludge solids at FTF and HTF to inform HTF inventory assignments. The NRC staff's concern is that if some HTF inventories were estimated based on data from FTF, it is possible that certain radionuclides could have been underestimated because there are expected differences between the waste streams that HTF and FTF processed. For example, SRNL-STI-2012-00479, Rev. 0 states that there is expected to be approximately 2,000 times more U-234 at HTF than at FTF, which results from zero inventory being reported in SRNL-STI-2012-00479, Rev. 0 for some of the FTF tanks (e.g., Tank 5). Likewise, SRNL-STI-2012-00479, Rev. 0 reports expectations of about 70 times more U-236 at HTF than at FTF. DOE states that the ratio of HTF to FTF inventories for certain radionuclides is distorted since SRNL-STI-2012-00479, Rev. 0 lists zero inventories for several waste tanks in FTF even though it is known that those tanks contain those radionuclides.

The potential impact of averaging across tank types is lessened by the conservatism built into the WCS values and the inventory adjustments made by DOE (see Section 3.2.1.4). The WCS generated concentration values are generally conservative because each reactor spent fuel assembly that was reprocessed is assumed to have received the maximum burn-up possible, and therefore, the amounts of actual fission products contained in an assembly were actually less than those entered into WCS. Also the use of oxalic acid cleaning in the tanks is expected to reduce the concentrations for some constituents in the residual sludge on the waste tank bottoms. The actual concentrations are expected to be less than the concentrations for dried sludge given in the WCS (SRR-CWDA-2010-00023, Rev. 3).

In response to RAIs (Response to RAI-INV-3; SRR-CWDA-2013-00106, Rev. 1), DOE explained why inventory projection ratios of HTF to FTF may not follow the expectations in the report SRNL-STI-2012-00479, Rev. 0. Specifically, one of the inventory adjustments involves assuming the maximum inventory of radionuclides within a tank group (see Section 3.2.1.4). While this adjustment biases the inventory high, it could also lead to the ratio of HTF to FTF inventories being different than expected. Also, DOE pointed out that the U-234 ratio presented in SRNL-STI-2012-00479, Rev. 0 is distorted by the fact that zero is reported for the inventories for several tanks in FTF (Response to RAI-INV-3; SRR-CWDA-2013-00106, Rev. 1). Given the information provided by the DOE as follow-up to the May 16, 2013, teleconference and in response to the RAIs, the NRC staff finds the inventory starting point for HTF reasonable.

3.2.1.3 NRC Evaluation of Projected Tank Volume

For the purposes of the FTF PA, DOE assumed that 0.2 cm (0.06 in) of residual material would remain in the tanks. During the FTF review, the NRC staff expressed concern that this value for residual volume was overly optimistic. In response, DOE increased the inventory for all tanks except Type III tanks in the FTF by a factor of 10. This equates to approximately 6.2 m^3 (1,650 gal) for a 75-ft (23-m) diameter tank. For the HTF PA, DOE applies a more conservative estimated residual volume of 15 m^3 (4,000 gal) for each tank. The value of 15 m^3 (4,000 gal) is based on the experience with cleaning Tank 5 (7.2 m^3 [1,900 gal]), Tank 6 (11.3 m^3 [3,000 gal], Tank 16 (0.83 m^3 [220 gal]), Tank 18 (14.7 m^3 [3,900 gal]), and Tank 19 (7.6 m^3 [2,000 gal]). Since DOE's volume estimate is based on actual experience with tanks cleaned to date, including Type I tanks, which are likely to be the most difficult to clean given the internal obstructions, the NRC staff finds the volume estimate to be reasonable and appropriate.

3.2.1.4 NRC Evaluation of Inventory Adjustments

The following are key differences between FTF and HTF inventory adjustments:

- For HTF, for the purposes of estimating Pu-238 inventories in the Type III/IIIA tanks, the grouping is split based on the two different waste types (salt and sludge). The NRC staff notes that DOE uses the incorrect inventory for Pu-238 for Tank 35. According to DOE's tank groups, Tank 35 should have a Pu-238 inventory equal to that of the *sludge* Type III/IIIA but instead, Tank 35 has an inventory equal to that of the *salt* Type III\IIIA tanks in the HTF PA (SRR-CWDA-2010-00128, Rev. 1).

- For FTF Type I and IIIA tanks, DOE increased the individual waste tank inventories by one order of magnitude to account for uncertainties in cleaning effectiveness. This inventory adjustment was not made for HTF, because the assumed volume for HTF was considered large enough to account for uncertainty in cleaning effectiveness. Still, because the 15 m^3 (4,000 gal) is a larger inventory than 10 times the initial volume estimate of 0.06 inches assumed for FTF, this particular adjustment of inventory is more conservative for HTF.

- For HTF, the cesium, strontium, and zirconium inventories were reduced by one order of magnitude.

During the May 16, 2013, clarification teleconference, as well as during the RAI process, the NRC staff asked DOE to clarify the assumptions regarding Cs-137, Sr-90, and Zr-93 inventories (Shaffner, 2013d [ML13193A072]; RAI-INV-5, Mohseni, 2013a [ML13196A135]). In response, DOE provided SRR-CWDA-2013-00086, Rev. 0 which describes that the Cs-137 and Sr-90 concentrations are calculated using total inventory and volume data from the WCS. The Zr-93 concentrations are estimated by using a ratio of Sr-90 to Zr-93 of 58,000:1 developed from sludge batch samples. The response, SRR-CWDA-2013-00086, Rev. 0, reports initial concentrations in Table A3-5 that are adapted in Table 3-10. DOE also provided this information in response to the staff's RAI (Response to RAI-INV-5; SRR-CWDA-2013-00106, Rev. 1).

Table 3-10 Initial Concentrations of Cs-137, Sr-90, and Zr-93 for Tanks 9H–15H

	Units*	Tank 9	Tank 10	Tank 11	Tank 12	Tank 13	Tank 14	Tank 15
Cs-137	Ci/gal	$1.2\times10^{+00}$	1.2×10^{-01}	$3.2\times10^{+00}$	8.4×10^{-01}	$1.3\times10^{+00}$	3.7×10^{-01}	$2.1\times10^{+00}$
Sr-90	Ci/gal	$1.6\times10^{+01}$	$1.7\times10^{+00}$	$5.8\times10^{+01}$	$1.5\times10^{+01}$	$2.1\times10^{+01}$	$5.4\times10^{+00}$	$3.7\times10^{+01}$
Zr-93	Ci/gal	2.8×10^{-04}	3.0×10^{-05}	1.0×10^{-03}	2.5×10^{-04}	3.7×10^{-04}	9.3×10^{-05}	6.3×10^{-04}
Sr-90:Zr-93		$5.7\times10^{+04}$	$5.7\times10^{+04}$	$5.8\times10^{+04}$	$6.0\times10^{+04}$	$5.7\times10^{+04}$	$5.8\times10^{+04}$	$5.9\times10^{+04}$

Adapted from Table A3-5 of SRR-CWDA-2013-00086, Rev. 0.
* To convert Ci/gal to Bq/L multiply by 9.8×10^{9}

DOE calculates the total activity by assuming 15 m^3 (4,000 gal) per tank and decays it to the time of site closure in year 2032. The maximum value within each tank type grouping is chosen to represent that tank type. DOE then reduces the magnitude of the cesium, strontium, and zirconium inventories for all tank types by one order of magnitude based on process samples taken before and after chemical cleaning of Tank 5. Table A3-9 of SRR-CWDA-2013-00086, Rev. 0, reports Tank 5 process sample concentrations before and after chemical cleaning, which is reproduced in Table 3-11.

Table 3-11 Tank 5 Concentrations for Cs-137, Sr-90, Elemental Sr and Zr Before and After Chemical Cleaning

		Before (WSRC-STI-2007-00192, Rev. 1)		After (SRNL-STI-2009-00492, Rev. 0)	
	Units*	**Aqua Regia**	**Peroxide Fusion**	**Aqua Regia**	**Peroxide Fusion**
Cs-137	mCi/kg	$1.09\times10^{+03}$	—	$4.86\times10^{+01}$	$3.51\times10^{+01}$
Sr-90	mCi/kg	$3.70\times10^{+04}$	—	$5.82\times10^{+03}$	$5.46\times10^{+03}$
Sr	mg/kg	$1.29\times10^{+03}$	$1.71\times10^{+03}$	$1.08\times10^{+02}$	$<3.97\times10^{+02}$
Zr	mg/kg	$3.91\times10^{+03}$	—	$1.11\times10^{+03}$	

Adapted from Table A3-9 of SRR-CWDA-2013-00086, Rev. 0.
* To convert mCi/kg to Bq/kg multiply by 3.7×10^{7}

The results in Table 3-11 were measured from a process sample that was taken during mechanical sludge removal campaigns and a process sample taken after chemical cleaning, and therefore, may not adequately represent the heterogeneity of the tank. Furthermore, since the "before" sample was taken in the middle of the mechanical sludge removal campaigns, the values may not accurately represent the actual effectiveness of chemical cleaning.

Given that FTF Tanks 5 and 6, which were cleaned with oxalic acid, have been fully characterized, it is useful to compare the projected inventories to the final characterization results for these tanks to better understand whether HTF inventory projections might be reasonable. Staff notes that FTF Tanks 18 and 19 (Type IV) have also been fully characterized, but they were not chemically cleaned due to the zeolite resin in these tanks. As can be seen in Table 3-12, the majority of the projected inventories for the tanks that have already been cleaned were not overestimated by an order of magnitude (i.e., 10 times). All inventories were estimated within one order of magnitude and some of the projected inventories were underestimated, especially with respect to Zr-93. It should be noted that the projected inventory for Zr-93 for FTF was based on the analytical detection limit. Also, the final ratio of Sr-90 to Zr-93 is substantially less than the 58,000:1 for Tanks 5-6 and Tank 19.

In response to the NRC staff's RAI, DOE provided additional justification for estimating the Zr-93 inventory based on an assumed ratio to the Sr-90 concentration given that the measured results for the tanks that have been cleaned do not reflect a 58,000:1 Sr-90 to Zr-93 ratio (Response to RAI-INV-5; SRR-CWDA-2013-00106, Rev. 1). DOE stated that the sludge batch analyses incorporate material from many different waste tanks, particularly HTF material. Therefore, DOE believed this ratio was a better indication of the overall ratio for HTF, as opposed to the samples from a small number of FTF tanks.

Table 3-12 Projected and Measured Inventories of Cs-137, Sr-90, and Zr-93 for Selected FTF Type I and IV Tanks and HTF Projections for Type I and IV Tanks

	Tank 5 and Tank 6 (Type I)		Tank 18 and Tank 19 (Type IV)		HTF Projections	
	Projected† (Ci)	Measured‡ (Ci)	Projected† (Ci)	Measured‡ (Ci)	Type I§ (Ci)	Type IV§ (Ci)
Cs-137	**5:** $9.2\times10^{+03}$ **6:** $9.2\times10^{+03}$	**5:** $3.5\times10^{+03}$ **6:** $6.7\times10^{+03}$	**18:** $9.7\times10^{+03}$ **19:** $6.5\times10^{+03}$	**18:** $9.2\times10^{+03}$ **19:** $4.2\times10^{+03}$	7.9×10^{-02}	$2.4\times10^{+03}$
Sr-90	**5:** $1.3\times10^{+05}$ **6:** $1.3\times10^{+05}$	**5:** $9.7\times10^{+04}$ **6:** $2.0\times10^{+05}$	**18:** $1.1\times10^{+03}$ **19:** $5.2\times10^{+00}$	**18:** $2.5\times10^{+03}$ **19:** $6.9\times10^{+00}$	$1.4\times10^{+04}$	$3.1\times10^{+02}$
Zr-93	**5:** 1.0×10^{-03} **6:** 1.0×10^{-03}	**5:** $3.0\times10^{+01}$ **6:** $2.2\times10^{+01}$	**18:** 1.0×10^{-03} **19:** 1.0×10^{-03}	**18:** 8.6×10^{-02} **19:** 1.8×10^{-02}	4.0×10^{-01}	8.8×10^{-03}
Sr-90: Zr-93		**5:** $3\times10^{+03}$ **6:** $9\times10^{+03}$		**18:** $3\times10^{+04}$ **19:** $4\times10^{+02}$	$3.5\times10^{+04}$	$3.5\times10^{+04}$

* To convert Ci to Bq multiply by $3.7\text{x}10^{+10}$
† SRS-REG-2007-00002, Rev. 1
‡ SRR-CWDA-2012-00071, Rev. 0
§ SRR-CWDA-2010-00128, Rev. 1

The NRC staff has reviewed the inventory adjustments. The NRC staff recommends that DOE analyze the impact of correcting the Pu-238 Tank 35 inventory. The NRC staff finds the further adjustment for Zr-93 in the Type I tanks reasonable, but notes that Zr-93 may also be present in unexpected amounts in other types of tanks in addition to Type I tanks. Also, if it is not practical to clean the tanks with zeolite, adjusting the inventory of these tanks may not be appropriate depending on the alternative cleaning strategy selected. DOE should continue to sample for Zr-93 (even though Zr-93 is not an HRR), given the unexpected results in FTF Tanks 5 and 6, to reduce uncertainty in the Zr-93 projections.

3.2.1.5 NRC Evaluation of Zeolite Assumptions

Regarding tanks with zeolites (Tanks 24, 32, 38, 40, 42, and 51), DOE's approach for accounting for zeolite is reasonable. DOE assumes that zeolite weight and volume fractions are the same in residual material after cleaning as they are prior to cleaning. In other words, the HTF PA does not assume any preferential removal of zeolite. During the teleconference on May 16, 2013, DOE cited CBU-PIT-2005-00099, Rev. 0 as a source for the weight and volume fractions that are applied (Shaffner, 2013d [ML13193A072]). The NRC staff finds the zeolite assumptions reasonable, given that the cleaning technologies DOE will use are not expected to remove zeolite in a different proportion to the residual solids.

3.2.1.6 NRC Evaluation of Cooling Coils and Tank Walls

For the HTF PA, DOE assumes that the inventory of material that potentially remains inside failed cooling coils and on the surface of the waste tank walls, cooling coils and columns is negligible compared to the estimated total tank inventories. For FTF Tank 18, due to poor visibility during cleaning, DOE overlooked scale build-up on the Tank 18 walls that contained a significant inventory of Pu-238. This oversight led DOE to conclude that a second washing of the tank wall was not necessary (Camper, 2011 [ML112371751]). During FTF monitoring, the NRC staff expressed concerns with DOE's ability to inspect tank walls and the miles of cooling coils, where present, during tank cleaning and sampling and analysis. DOE indicated that

significant technological improvements have been made with respect to the tools used to inspect internal tank surfaces and that oxalic acid treatment is effective at removing any build-up on internal tank surfaces. The NRC staff indicated that it would continue to evaluate this technical concern during future monitoring activities for FTF and provided suggestions to DOE on how it could address the NRC staff's technical concerns in the future (Barr, 2013a [ML13085A291]).

The NRC staff finds the assumptions for inventory for cooling coils and tank walls reasonable given the experience to date with cleaning of Tanks 5 and 6. However, the NRC staff notes that if oxalic acid is not available to be used for cleaning future tanks and a technology with similar proven effectiveness is not used as an alternative, DOE may need to reconsider the validity of assuming that the cooling coil and tank wall surface inventory is negligible.

3.2.1.7 NRC Evaluation of Tank Inventory Uncertainty

DOE's approach to developing parameter distributions for the HTF inventory for tanks that have not yet been cleaned is biased towards lower values. DOE uses a log-uniform distribution for the inventory for HTF that ranges from a factor of 100 times less to a factor of 10 times higher inventory, depending on tank type and radionuclide. In SRR-CWDA-2010-00023, Rev. 3, DOE compares the inventory projections to the sample results for FTF Tanks 5, 18, and 19. For the majority of radionuclides, the uncertainty assumptions bound the actual results. However, in a separate FTF monitoring activity related to the development of Tanks 5 and 6 inventory, the NRC staff concluded that a significant percentage of radionuclides were underestimated, and in some cases grossly underestimated, by more than an order of magnitude (Barr, 2013b ML13273A299]). One radionuclide that was significantly underestimated is Zr-93, which had a projected inventory equal to the detection limit of 0.001 Ci, but was measured as 30 Ci in Tank 5 (and 22 Ci in Tank 6). From this experience, DOE increased the projected inventory of Zr-93 for HTF, as compared to FTF, by two orders of magnitude from the detection limit to be the following: 0.4 Ci for Type I tanks, 0.0088 Ci for Type IV tanks, and 0.57 Ci for Type III/IIIA tanks (Response to RAI-INV-5; SRR-CWDA-2013-00106, Rev. 1).

In an FTF technical review, the NRC staff indicated that DOE should provide a stronger technical basis for the projected inventory multipliers used in the probabilistic analysis (Barr, 2013b [ML13273A299]). However, because the probabilistic analysis is not strictly relied on, but rather informs the demonstration of compliance with the performance objectives in 10 CFR Part 61, Subpart C, this technical concern can be addressed as a longer-term activity under Monitoring Factor 6.2 "Model and Parameter Support" in the NRC staff's FTF Monitoring Plan (Camper, 2013a [ML12345A322]). Given the significant fraction of radionuclide inventories that were underestimated, it was not clear to the NRC staff that the inventory multipliers should be biased at 100 times less and only 10 times higher. The NRC staff also encouraged DOE to continue to investigate the reasons behind the unexpected radionuclides with significant inventories, and to revise its inventory and uncertainty assumptions accordingly. As more sample data are available from tanks that have been cleaned, DOE should analyze trends in projections versus actual inventories by radionuclide to update the multiplier assumptions for the probabilistic analysis. This is especially important if DOE desires to rely partially on the probabilistic model results for the compliance conclusions with the 10 CFR Part 61 performance objectives per NDAA Criterion 3.

3.2.2 NRC Evaluation of Annular and Sandpad Inventory

Waste in the annulus and/or sandpad can be more risk significant than the inventory in the primary tank, because the waste is expected to be more soluble and is located outside of the primary containment. During its review, the NRC staff had several questions relating to the annulus inventory, in particular the Tank 16 annulus projected inventory, because other tank annulus inventories are based on the Tank 16 annulus inventory. Tank 16, a Type II Tank, was put into service in May 1959. Shortly thereafter, DOE discovered leakage into the annulus from the primary tank. On September 8, 1960, leakage from the primary tank overfilled the pan in the annular space and some material leaked through the concrete encasement (DP-1358).

DOE estimates the remaining volume in the Tank 16 annulus to be 12.5 m^3 (3,300 gal) through the use of camera views and interior landmarks (i.e., duct diameter, annulus wall radius) (SRR-LWE-2012-00039, Rev. 0). There is residual material in the bottom of the annulus (8 m^3 [2,100 gal]) as well as inside the duct (4.5 m^3 [1,200 gal]). In addition, DOE estimates a separate inventory for the primary sandpad (5 m^3 [1,350 gal]) and the secondary sandpad (0.1 m^3 [26 gal]). There are many areas of the annulus (and duct) where visual determination of the waste level was not possible. In those areas, DOE extrapolates the waste level using the data from surrounding areas. Also, the sandpads of the Type II tanks are not visible. DOE assumes the primary sandpad of Tank 16 (as well as Tank 14) is saturated at the same concentration as the annulus, which is equivalent to 5 m^3 (1350 gal). However, DOE assumes only 100 gallons of residual material is in the primary sandpads for Tanks 13 and 15. The NRC staff notes that it is difficult for DOE to predict how much residual waste may have migrated into the primary sandpad in each of these tanks over time. The only evidence DOE presented is photographs of the annuli interior showing salt deposits of past leak sites. Since the sandpads are not visible, DOE could have underestimated the volume of material in these areas for Tanks 13 and 15.

A DOE report from 1974 estimates that a maximum of 2.6 m^3 (700 gal) of waste rose above the top of the steel secondary containment pan, and that about 60 L (16 gal) of waste leaked from the annulus into the surrounding soil, although the amount of waste that leaked into the soil and groundwater was not directly measured (DP-1358). In the HTF PA, DOE assumes that 100 L (26 gal) is present in the secondary sandpad layer of Tank 16. The inventory in the secondary sandpad is meant to represent a conservative estimate of what leaked into the surrounding environment. While the NRC staff finds the projected inventory in the secondary sandpad not as risk significant as the inventory in the primary sandpad, the staff is concerned that the cases[5] modeled in the HTF PA may underestimate annular contamination risk as discussed in Section 4.2.9 (cf. RAI-MEP-7, RAI-NF-12 and RAI-NF-13; Mohseni, 2013a [ML13196A135]).

In Section 9.0 of SRR-CWDA-2010-00023, Rev. 3, DOE describes how in the HTF PA multipliers are applied to the tank inventories to account for uncertainty. However, similar inventory multipliers are not applied for the annuli of the tanks. The HTF PA states that

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[5] DOE evaluates several cases, designated as Case A through Case E among others, in the HTF PA. These cases describe potential degradation of the HTF and release of radionuclides over time. Case A represents DOE's expectation of the future degradation of the tanks and release of radionuclides and is used to demonstrate compliance with the 10 CFR Part 61 performance objectives. The other cases are intended to represent uncertainty in the degradation of tanks and release of radionuclides. Chapter 4 of this TER describes the cases in more detail.

estimates for the annulus material are extremely conservative compared to the amount anticipated to remain, so uncertainty multipliers are not necessary. DOE estimates that Tanks 9, 10, and 14 annuli contain about 12.5 m^3, 4.2 m^3, and 21.2 m^3 (3,300, 1,100, and 5,600 gal) of material respectively, although in the HTF PA, DOE assumes 12.5 m^3 (3,300 gal) for all tanks with residual material in the annulus (Shaffner, 2013e [ML13183A410]).

The NRC staff requested that DOE provide estimates of the uncertainty of the remaining volume in the Tank 16 annulus and describe how this uncertainty is related to the uncertainty of the material in the annuli of other tanks (RAI-INV-1; Mohseni, 2013a [ML13196A135]). In response, DOE stated that the preliminary results of volume levels obtained during Tank 16 annulus sampling, which was ongoing during the preparation of this TER, confirms that the 3,300 gallon estimate is conservative (RAI-INV-1; Mohseni, 2013a [ML13196A135]).

The NRC staff requested that DOE clarify the technical basis for using Tank 16 annulus concentrations in order to estimate the concentrations to be found in the annuli of Tanks 9, 10, and 14, given the expected chemical differences between Tank 16 annulus material and that of the other tanks. The NRC staff also requested that DOE describe the impact of the anticipated differences on the inventory estimates in terms of radionuclides that could be under- or overestimated as a result of assuming the Tank 16 annulus concentrations are representative of other tanks (RAI-INV-2; Mohseni, 2013a [ML13196A135]). DOE responded by stating that it predicts that a significant fraction of the salt waste in the annulus of Tanks 9, 10, and 14 will be removed based on the physical and chemical differences between the waste in those tanks and Tank 16. DOE expects the waste in Tanks 9, 10, and 14 to be more soluble and does not contain the sand material that Tank 16 annulus contains. Also, because the leaks into the annuli of Tanks 9, 10, and 14 occurred slowly, the waste in the tank is expected to have precipitated less soluble radionuclides (into the sludge) and the waste reaching the annulus is expected to be more chemically similar to the supernatant (i.e., largely containing more soluble radionuclides). In contrast, the Tank 16 leak occurred more rapidly, which is expected to have limited the amount of time for less soluble radionuclides to precipitate and form sludge, thereby, allowing both soluble and less-soluble radionuclides to reach the annulus. DOE uses this justification as the rationale for why the concentration assignments were considered conservative (Response to RAI-INV-2; SRR-CWDA-2013-00106, Rev. 1).

With regard to the potential use of annulus inventory multipliers, DOE states that by using a conservative deterministic annulus inventory, as opposed to a distribution that would include the possibility of much smaller inventory realizations, the analysis is more conservative. Also, DOE will account for volumetric uncertainty in the tank-specific special analysis (SA) (Response to RAI-INV-1; SRR-CWDA-2013-00106, Rev. 1).

DOE uses the concentration of the four samples analyzed in 2011 (all from outside of the ventilation duct) to represent all of the Tank 16 annulus material (SRNL-STI-2012-00178, Rev. 0). The material inside the ventilation duct, which DOE estimates to be roughly one-third of the total material (see SRR-CWDA-2010-00128, Rev. 1), has higher concentrations of several HRRs based on comparison of the 2006 samples to the 2011 samples (WSRC-STI-2008-00203, Rev. 0).

The NRC staff requested that DOE clarify why the 2006 annulus samples were not used (in addition to the 2011 annulus samples) to help inform the inventory projection for the Tank 16 annulus. The NRC staff also requested that DOE provide justification for why the concentration of the samples taken from outside the ventilation duct in 2011 is presumed to adequately

represent the concentration of material inside the duct for the purposes of the HTF PA (RAI-INV-4; Mohseni, 2013a [ML13196A135]). In response, DOE explained that at the time the inventory was developed, DOE expected the material outside the duct to represent the bulk of the residual material. DOE had assumed that the portion inside the duct (which is more soluble) would be removed due to cleaning. Since DOE no longer plans to remove additional material from the Tank 16 annulus, DOE analyzed the impact of incorporating the 2006 samples into the inventory projections. If the 2006 samples are used along with the 2011 samples, the Sr-90 inventory would be about 5.8 times higher and the Tc-99 about 1.8 times higher as examples. Tc-99 is the primary contributor to the peak dose of 0.13 mSv/yr (13 mrem/yr) (at around 400 years) under the "Flow Run 65, No Holdup" assumptions, as discussed in RAI NF-12. DOE stated that the current sampling of Tank 16 in support of final characterization shows that the residual volume in the Tank 16 annulus is likely to be significantly less than the assumed value of 12.5 m^3 (3,300 gal) (Response to RAI-INV-4; SRR-CWDA-2013-00106, Rev. 1).

Furthermore, for those risk significant radionuclides that were not analyzed in the four samples from 2011 and were not estimated via the detection limit, DOE bases the annulus inventory on a ratio of a chemically similar element and the tank primary floor residual. For insoluble radionuclides, DOE uses a ratio to the Pu-238 analysis, while for soluble radionuclides, DOE uses a ratio to the Tc-99 analysis. For example, Am-241 was not analyzed in the Tank 16 annulus samples. The Am-241 annulus concentration was estimated by multiplying the Pu-238 Tank 16 annulus sample concentration by a ratio of the Am-241 to Pu-238 primary inventories (Response to RAI-INV-4; SRR-CWDA-2013-00106, Rev. 1). The NRC staff notes that if the waste in the annulus is expected to contain greater amounts of the soluble radionuclides than the waste in the primary tank, using a ratio to the primary inventory for these radionuclides may not be appropriate.

The NRC staff finds the concentration assignments for the annuli of Tanks 9, 10, and 14 appropriate if the material associated with the annuli of those tanks is cleaned. However, staff notes that the concentration of soluble radionuclides is likely higher in the material in the annuli of these tanks than in Tank 16, and therefore, the annulus inventory assumed in the HTF PA (especially for Tank 14 which has an estimated volume of 21 m^3 [5,600 gal]) could be less than what currently remains in the annulus. Furthermore, since the sandpad is not visible and most likely impractical to sample, there is uncertainty surrounding the sandpad inventories. If plans change and DOE decides not to perform additional cleaning of these annuli, the NRC staff recommends that DOE revise the annulus inventory assumption in the HTF PA.

3.2.3 NRC Evaluation of Ancillary Equipment Inventory

DOE's approach for estimating ancillary equipment inventory for HTF is similar to the approach that the NRC staff reviewed for the FTF in its FTF TER (Camper, 2011 [ML112371751]). The FTF TER discusses the NRC staff's previous analysis of DOE's approach for ancillary equipment.

The HTF PA Section 8.2, "Further Work," discusses future characterization of the transfer lines (SRR-CWDA-2010-00128, Rev. 1). 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. The NRC staff will follow-up on the need for auxiliary equipment characterization during the HTF monitoring period.

3.2.4 NRC Evaluation of Inventory Final Characterization

Although DOE had not completed final characterization for any of the HTF Tanks as of the development of this TER, DOE has characterized, and the NRC staff has reviewed, the final inventory of residual material for FTF Tanks 18 and 19 (Camper, 2011 [ML112371751]) as well as FTF Tanks 5 and 6 (Barr, 2013a [ML13085A291]). The information from the previous FTF reviews remains pertinent to HTF because DOE largely plans to use the same process for final characterization at HTF. The reader may refer to those separate documents for details on the NRC staff's prior evaluations. For convenience, the major conclusions of these evaluations are summarized below.

During its review for FTF Tanks 18 and 19, the NRC staff had concerns with the quantification of volume uncertainty and the assumptions surrounding development of inventory multipliers for the probabilistic analysis for FTF Tanks 18 and 19 (Camper, 2011 [ML112371751]). The NRC staff suggested that DOE 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.

Concurrent with sampling and analysis of FTF Tanks 5 and 6, DOE made several improvements to its tank sampling and volume estimation programs. For example, DOE developed (1) a Liquid Waste Tank Residuals Sampling and Analysis Program Plan (SRR-CWDA-2011-00050, Rev. 1) and (2) a Quality Assurance Program Plan (SRR-CWDA-2011-00117, Rev. 0). DOE also developed training and procedures (SRR-LWE-2010-00240, Rev. 1) to formalize the tank residual volume mapping program. The NRC staff reviewed (1) and (2) above as an FTF monitoring activity and concluded that these improvements have led to a more technically defensible program (Barr, 2013a [ML13085A291]). The NRC staff identified several follow-up actions for the sampling plan related to the compositing approach, representativeness of the sampling, heterogeneity, and uncertainty. The NRC staff also identified follow-up actions related to volume mapping uncertainty, including the suggestion of field validation activities for a range of solid material heights. . These follow-up actions that were identified during FTF reviews are also applicable at HTF in so far as final HTF characterization activities follow a similar process to what has been done at FTF.

3.2.4.1 NRC Evaluation of Final Volume Determination Approach

DOE's approach to developing the volume estimates for Tanks 5 and 6 appears to be technically sound and adequately executed. However, it is less clear that volume uncertainty is adequately managed in this area. The NRC staff provided the following recommendations as part of FTF Monitoring, which are also relevant to HTF because the process will be largely similar to what was used at FTF (Barr, 2013a [ML13085A291]):

- DOE should better understand the accuracy of mapping team height estimates through additional field validation activities for a range of solid material heights.

- DOE should clearly communicate how it delineates the size of areas of similar height that are mapped, and how it manages uncertainty related to height estimates for discretized areas in its deterministic analysis. Likewise, DOE should clarify how it represents uncertainty in the assignment of high and low end heights to these areas (e.g., does it use a height that is clearly below/above the non-uniform surface of the delineated areas).

- DOE should consider uncertainty in the volume estimates resulting from the transfer of data from photographic and video evidence to hand contoured maps (and then to Excel spreadsheets with a finer discretization).

- DOE should be more transparent with respect to its approach to (1) mapping annular volumes, including the use of a crawler to inspect internal surfaces, and (2) estimating residual waste volumes in ventilation ducts. DOE should consider uncertainty in annulus volume estimates.

- Alternatively, volume mapping uncertainty could be managed through the use of estimates that are biased towards higher volumes.

3.2.4.2 NRC Evaluation of Final Concentration Sampling Plan

Although it had not been completed at the time the draft basis for the WD for HTF was written, DOE provided the sampling and analysis plan for Tank 16 to the NRC (SRR-LWE-2013-00057, Rev. 0). The NRC staff provides some initial comments in this section on the information that was available on Tank 16 final characterization plans, but the detailed review of the Tank 16 final inventory will be conducted during monitoring.

As summarized in Section 3.1.4.2, DOE plans to use 11 new samples and four prior samples to create the three composites for the Tank 16 annulus. Each composite sample will consist of 5 or 6 sample locations.and will be analyzed in triplicate. In the future, DOE should consider FTF technical review (Barr, 2013a [ML13085A291]) comments that were discussed during an August 27-28, 2013, onsite observation visit (Mohseni, 2013b [ML13267A452]). The NRC staff provided the following comments (Barr, 2013a [ML13085A291]):

- DOE should consider, in its tank sampling design, historical information on tank waste receipts and information related to the alteration and redistribution of waste due to cleaning operations that may impact horizontal and vertical waste heterogeneity.

- DOE should evaluate the option to composite samples within segments (or strata) to preserve information about segment (or strata) variance.

- DOE should evaluate and present information on the relative contributions of various forms of uncertainty in its estimation of mean tank concentrations.

- DOE should clarify the statistical approach used to estimate the 95^{th} percentile upper confidence limit (e.g., treatment of all nine measurements as independent when computing the upper confidence limit).

- DOE should also consider how it can better ensure sample representativeness by improving tank sampling designs, collection tools, and instructions.

- Alternatively, DOE could manage sampling and analysis uncertainty through the use of estimates that are biased towards higher inventories.

3.2.5 NRC Evaluation of Inventory Conclusion

The 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 these projected inventories is to predict doses for Criterion 3. Additionally, because DOE has not provided final inventories, the NRC staff cannot make the Criterion 2 determination at this time. The NRC staff's review of the inventory focuses on the DOE's process for characterizing tanks, including its experience to date in applying that process to the tanks that have been cleaned for FTF. Staff notes that since some of the inventory assumptions are based on the cleaning experience obtained with baseline technologies, DOE may need to revisit those assumptions should the baseline technologies change.

3.3 Identification of Highly Radioactive Radionuclides

HRRs are those radionuclides that contribute most significantly to risk to the public, workers, and the environment. In the context of the NRC staff's reviews of DOE's basis documents for WDs conducted under the NDAA, the term is not limited to radionuclides with high-specific activity. The NRC staff considers the term "highly radioactive radionuclides," as used in the context of the NDAA, to be equivalent to the term "key radionuclides" used in the manual for DOE Order 435.1 (DOE M 435.1-1), the West Valley Policy Statement, and in some of the NRC staff's reviews of DOE basis documents for WDs. For radionuclides with initial insignificant inventories, the parents are included for consideration in the list of HRRs, as opposed to the specific progeny. Even though the specific progeny may not be listed as an HRR, it is still considered to be a key radionuclide.

DOE performed an evaluation considering the risk to workers, the public, and the environment to identify HRRs at HTF that, according to NDAA Section 3116(a)(2), must be removed to the MEP. DOE summarizes the methodology for the evaluation in Section 5.1.1 of DOE/SRS-WD-2013-001, Rev. 0. 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, 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 HTF PA (SRR-CWDA-2010-00128, Rev. 1). The remainder of this section will summarize the assessment of compliance with the performance objectives.

To determine which of the 54 radionuclides are HRRs DOE considers radionuclides that are important to demonstrating compliance with the 10 CFR Part 61 performance objectives, because they contribute to the dose to members of the public, inadvertent intruders, and workers. Specifically, the assessment considers analyses performed for the HTF PA including (1) doses estimated for the groundwater pathway at 100 m; (2) doses estimated for the airborne pathway at 100 m; (3) doses estimated for the inadvertent intruder; and (4) results of uncertainty and sensitivity analyses. The evaluation resulted in the 13 HRRs that are listed inTable 3-13. The methodology for selecting these 13 HRRs is summarized next.

Table 3-13 H-Area Tank Farm HRRs

Radio-nuclide	Half-Life (yrs)	Table 1 of 10 CFR 61.55	Table 2 of 10 CFR 61.55	Groundwater Dose	Intruder Dose	Uncertainty in Dose	Worker Dose
Sr-90	2.9×10^{1}		x		x	x	x
Tc-99	2.1×10^{5}	x		x		x	
I-129	1.6×10^{7}	x		x			
Cs-137	3.0×10^{1}		x		x	x	x
U-233	1.6×10^{5}				x		
U-234	2.5×10^{5}			x		x	
U-235	7.04×10^{8}			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	
Pu-240	6.6×10^{3}	x				x	
Am-241	4.3×10^{2}	x		x	x	x	
Am-243	7.37×10^{3}	x			x		

Adapted from Table 5.1-1 in DOE/SRS-WD-2013-001, Rev. 0.
Note: Table 5.1-1 of DOE/SRS-WD-2013-001, Rev. 0 does not identify I-129 as being listed in Table 1 of 10 CFR 61.55. The NRC staff notes that this is incorrect. Table 1 of 10 CFR 61.55 does list I-129.

To identify potential HRRs, DOE examines resulting doses from the groundwater analysis in the HTF PA at 100 m over a 20,000-year period for all HTF sources. Those radionuclides, which in aggregate were not estimated to contribute more than 0.0125 mSv/yr (1.25 mrem/year) are not considered HRRs. The groundwater evaluation identifies the following radionuclides for consideration as HRRs: Ni-59, Tc-99, I-129, Ra-226, Pa-231, and Np-237. Further, the analysis considers the projected inventories of these radionuclides at the time of site closure, as described in the following bullets:

- Radionuclides with initial insignificant inventories were removed from consideration and their parents were included for consideration as HRRs.
    - Two radionuclides (Pa-231 [progeny of both U-235 and Pu-239] and Ra-226 [progeny of both U-234 and Pu-238]) were removed for this reason.
    - U-235, U-234, and Pu-238 were added to the HRR list. While Pa-231 is a progeny of both U-235 and Pu-239, about 85% of the ingrowth of Pa-231 is from U-235. Therefore, Pu-239 was not added to the list of HRRs.
- One parent radionuclide (Am-241 [parent of Np-237]) was added to the HRR list based on progeny ingrowth.
- Ni-59, although important for the groundwater pathway, was not added to the HRR list because DOE believes that the HTF PA overestimates its importance. In the HTF PA, the dose contribution from Ni-59 is primarily associated with the inventory in the annuli of Type I tanks. DOE estimates that the actual Ni-59 inventory in the annuli of Type I tanks

will be much less than the inventory projected in the HTF PA. The inventory in the HTF PA is projected by setting the annulus inventory equal to the primary tank residual inventory for Type I tanks, or 8.6 Ci. Furthermore, the HTF PA PORFLOW™ model does not account for the stable nickel that will be present at two orders of magnitude greater than the Ni-59.

Following the steps described above, the groundwater analysis resulted in the inclusion of the following radionuclides as HRRs: Tc-99, I-129, U-234, U-235, Np-237, Pu-238, and Am-241.

DOE also examines resulting doses from the airborne pathway analysis in the HTF 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. It contributes, in aggregate, approximately 1.0×10^{-3} mSv/yr (0.1 mrem/yr) to the total dose. Therefore, no radionuclides were included as HRRs based on the airborne pathway.

Using the results from the HTF PA inadvertent intruder dose analysis, DOE identifies potential HRRs. Those radionuclides that, in aggregate, would not contribute more than 0.25 mSv/yr (25 mrem/yr) were not considered HRRs. The intruder evaluation identified the following radionuclides for consideration as HRRs: Sr-90, Y-90, Tc-99, I-129, Ra-226, Th- 229, U-233, U-234, Pu-239, Pu-240, Pu-244, Am-241, Am-243, Cm-245, Cm-247, Cm-248 and Pb-210.

Further, DOE's analysis considers the projected inventories at the time of closure for the radionuclides described in the following bullets.

- Radionuclides with insignificant inventories were removed from consideration if their parent was included for consideration as an HRR. The following radionuclides—Ra-226 (progeny of U-234 and Pu-238), Th-229 (progeny of U-233), and Pb-210 (progeny of Ra-226)—were removed for this reason.

- Pu-244, Cm-245, Cm-247 and Cm-248 were not included in the HRR list because DOE expects the actual inventory to be significantly less than the projected inventory. DOE projected the inventory of these radionuclides as 3.7×10^{10} Bq (1 Ci), which is at least one order of magnitude higher than the estimates utilizing Tanks 5, 6, 18 and 19 sample analyses and WCS estimates (SRR-CWDA-2010-00023, Rev. 3).

- Y-90 was not included as an HRR since it is only present when Sr-90 is present, due to its very short half-life.

The inadvertent intruder pathway analysis resulted in the inclusion of the following radionuclides as HRRs: Sr-90, Tc-99, I-129, U-233, U-234, Pu-239, Pu-240, Am-241, and Am-243.

DOE also examines results of uncertainty and sensitivity analyses for protection of the public 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 HTF PA, DOE identifies two significant radionuclides: Sr-90 and Tc-99. Using the results of the sensitivity analysis, DOE identifies Tc-99 and Ra-226 to be significant for protection of the public. Using the results of the deterministic sensitivity and barrier analyses for protection of the public, DOE identifies Tc-99, I-129, Ra-226, and Np-237. As a result of sensitivity analyses for the calculation of public dose, Sr-90, Tc-99, I-129, U-234, Np-237, Pu-238, and Am-241 were included as HRRs.

DOE also examines results of uncertainty and sensitivity analyses for the intruder scenario. Deterministic sensitivity analysis identified Ra-226 and U-234 as potentially significant. Ra-226 is primarily controlled through its parents U-234 and Pu-238 so they are included as HRRs. Results of the probabilistic sensitivity analyses for the intruder scenario led DOE to further consideration of the following radionuclides as potential HRRs: Sr-90, Y-90, Tc-99, Cs-137, and Ba-137m. DOE eliminates Y-90 and Ba-137m from further consideration as HRRs because they are typically associated with their parent radionuclides (i.e., Sr-90 and Cs-137, respectively), due to their short half-lives. In conclusion, the results of the sensitivity analyses for the inadvertent intruder analysis resulted in the inclusion of the following radionuclides as HRRs: Sr-90, Tc-99, Cs-137, U-234, and Pu-238.

3.4 NRC Evaluation of Highly Radioactive Radionuclides

The definition of "HRRs" used by DOE appears to be consistent with the NRC staff's understanding of the term. Specifically, the 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 3.3 summarizes DOE's approach to developing the list of HRRs. Based on the approach, DOE identifies 13 radionuclides as HRRs. The HRRs are listed in Table 3-13.

The HRR list for HTF includes the 10 HRRs that were identified for FTF, plus three additional radionuclides (U-233, U-235, and Am-243). The differences in the HRR lists between FTF and HTF result from the differences in the waste streams processed at each tank farm as explained in the following bullets:

- U-233 is produced from neutron irradiation of Th-232 and is present in HTF waste due to the thorium waste which was processed at H-Area.

- The inventory of U-235 is greater in HTF than in FTF because the HM process used at H-Area recovered material from burned enriched uranium fuel, while the PUREX reprocessing conducted at F-Area recovered material from irradiated depleted uranium targets.

- Am-243 is produced when a Pu-239 atom captures four neutrons. Weapons grade plutonium, which contains more Pu-239, was processed at FTF, whereas highly-burned plutonium was processed at HTF which contains more Am-243 than weapons grade plutonium.

As explained in Section 3.3, Ra-226 is an important radionuclide for both the groundwater and the intruder analyses. DOE eliminates Ra-226 as an HRR because DOE assumes Ra-226 will be present primarily due to its parent Pu-238. The NRC staff notes that the report SRNL-STI-2012-00479, Rev. 0 expected the average Pu-238 concentration in HTF sludge to be about twenty times that of FTF sludge, which would result in more Ra-226 over the long-term. The report also expected more Ra-226 initially present due to impurities in the thorium fuel that was processed at HTF (SRNL-STI-2012-00479, Rev. 0). NRC staff notes that the amount of thorium fuel processed at HTF was relatively small. However, if the initial sampling of HTF tanks shows Ra-226 to be present in greater quantities than projected, DOE should reconsider listing it as an HRR.

As explained in Section 3.3, several curium isotopes are important for the intruder analyses. DOE eliminates Cm-245, Cm-247, and Cm-258 based on the initial results from sampling FTF Tanks 5, 6, 18, and 19. However, HTF tanks are expected to have more Cm-245, Cm-247, and Cm-248 than the FTF tanks due to the differences in processes carried out at HTF. According to SRNL-STI-2012-00479, Rev. 0, HTF is expected to have 20 times more Cm-245 than FTF and 4 times the amount of Cm-244. Curium-247 and Cm-248 are formed when Cm-244 absorbs neutrons in a reactor. Therefore, one would also expect HTF to have more Cm-247 and Cm-248. DOE assumes an inventory of 20 Ci of Cm-245 in total at FTF and 29 Ci at HTF in the PAs. The NRC staff asked DOE in an RAI to clarify why the projected inventories do not reflect the ratios predicted in SRNL-STI-2012-00479, Rev. 0 for some radionuclides. DOE responded by stating that the ratios in SRNL-STI-2012-00479, Rev. 0 were distorted because several of the FTF tanks had zero inventories reported for specific radionuclides (SRR-CWDA-2013-00106, Rev. 1). The NRC staff recommends that if the initial sampling for HTF shows that concentrations of these curium isotopes are higher than expected, DOE should reconsider adding these curium isotopes to the list of HRRs.

The NRC staff evaluated DOE's selection of HRRs and the staff concludes that the combination of approaches used by DOE to identify HRRs in the context of the draft basis for the WD for HTF is reasonable. However, as discussed in Section 3.1 and Chapter 4, as DOE continues to evaluate assumptions for the HTF 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.

DOE's analytical list is broader than the HRR list, so DOE will have assurance that it will develop inventories for other radionuclides which could potentially be risk-significant but that are not on the HRR list because they are expected to be present in such low concentrations. An example of a radionuclide which was found to be present at levels much higher than expected is the activation product Zr-93 in FTF Tank 5. DOE indicated that it thoroughly reviewed waste transfer records in attempts to explain the Tank 5 Zr-93 inventory but did not find any additional information as to why the levels were higher than expected (Felsher, 2013b [ML13150A219]; Barr, 2013b [ML13273A299]). The NRC staff acknowledges DOE's efforts in this area and recommends that DOE continue to examine the reasons for unexpected results, should they occur, and attempt to trace them back to known waste streams or processes that might reveal other radionuclides that could have been underestimated by the projections based on WCS data. DOE should assess, through future tank residual characterization, the validity of prior assumptions and the resulting impacts to the list of HRRs.

3.5 Alternative Treatment Technologies

DOE has defined (1) baseline technologies and (2) alternative technologies for cleaning of the tanks. The baseline technologies are proven to have met a set of program requirements and constraints. Because historical activities at SRS incorporated several different radioactive separations processes, each producing waste streams with different physical properties, the waste in the HTF 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 for tank cleaning, DOE applies 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 the 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 (e.g., 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 (SLPs and SMPs): Three or four 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 (i.e., 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 depends upon the physical properties of the waste. Results have demonstrated that the sticky, mud-like sludge found in Tank 18 is more difficult to remove than the grainy, sand-like sludge found in Tank 19).

Chemical Heel Removal:

- Oxalic Acid: is added to the tank through downcomers or spray wash nozzles while SMPs or SLPs provide mixing. Oxalic acid was chosen for its strength, effectiveness, reducing tendency, and because it was determined to be less corrosive than other options, and readily available. Oxalic acid has been used for chemical cleaning on Tanks 16 and 24 in HTF and Tanks 5 and 6 in FTF. It is planned to be used on Tank 12 in HTF.

- Low-Temperature Aluminum Dissolution: DOE has deployed this technology at HTF to help remove solids from waste tanks that have high percentages of aluminum in the solids. Under this process, aluminum is dissolved from sludge waste into the supernate by treatment with caustic, followed by decantation, and subsequently water washing to remove aluminum. Low-Temperature Aluminum Dissolution was not developed for the purpose of targeting HRR removal from waste tanks, but instead was developed to reduce the amount of sludge solids being sent to DWPF, resulting in fewer vitrification canisters being produced.

In 1996, SRS initiated an effort to investigate alternative technologies. 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. DOE started using a specific free jet flow agitator, called a Flygt mixer, 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 the 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 rather than 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 one to two minutes. As with the SLP, the assembly is rotated or oscillated using a turntable above the waste tank opening. Due to the large size of the tanks and the smaller radius of the pulse jet mixer, a greater number of mixers is required and there is not a sufficient number of openings in the tank roofs for this technology to be highly effective (SRR-LWE-2013-00077).

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

- 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 small crawler vacuum cleaning system development (e.g., improved Mantis) based on lessons learned from Tanks 18 and 19

3.6 NRC Evaluation of Alternative Treatment Technologies

DOE evaluated alternative technologies in a Systems Engineering Evaluation (SEE) to select the current baseline technologies. Formal SEEs were conducted for the following waste removals:

- 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, listed the following constraints and weighting factors: 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). 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 timeframe (PIT-MISC-0040).

The SEE completed in 2001 for the Tank 18 waste 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, Rev. 0).

The Waste Removal Balance of Program SEE completed in 2003, which established much of the current baseline technologies, 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, Rev. 0).

The SEE completed in 2009 for Tank 5 and 6 Heel Removal (SRR-CES-2009-00022, Rev. 0) was not provided as a reference, but the references describing the history of waste removal in Tank 5 and 6 includes a discussion of this SEE. Mechanical feed-and-bleed 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 down-comer for Tank 5 (SRR-CWDA-2011-00033, Rev. 1). For Tank 6, a recirculation loop was used instead to maximize mixing and transfer turnover rate and reduce the amount of wastewater added to the system (SRR-CWDA-2011-00005, Rev. 1).

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 important 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, Rev. 0), 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. 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 the removal of HRRs because the studies do not discuss removal of key or important radionuclides.

The NRC staff requested that DOE provide an updated comprehensive description of DOE's current process for selection and evaluation of waste retrieval technologies to show that NDAA Criterion 2 will be met for tanks yet to be cleaned. The NRC staff requested that DOE describe whether or not the technology is intended to remove HRRs and how it accomplishes their removal if intended to do so (RAI-MEP-2; Mohseni, 2013a [ML13196A135]).

DOE summarized the current status of the technology baseline in its response (Response to RAI-MEP-2; SRR-CWDA-2013-00106, Rev. 1):

- For tanks with cooling coils, the baseline technology remains SLPs or SMPs for mechanical cleaning, followed by oxalic acid technology for chemical cleaning.

- For tanks without cooling coils, the baseline remains the Mantis technology.

Since issuance of *Waste Removal Technology Baseline: Technology Development Description*, V-ESR-G-00003, Rev. 1, DOE has provided two annual updates (SRR-LWE-2012-00082, SRR-LWE-2013-00077) to SCDHEC: one in April 2012, and another in April 2013. The most recent updates are summarized below:

- SRS continues to interface with Hanford regarding its Mobile Arm Retrieval System technology which SRS might utilize for the heavier particles (e.g., those that were present in FTF Tanks 5 and 6) that could not be suspended with pumps. The Mobile Arm Retrieval System will be considered for Tanks 21-24 Type IV tanks at SRS since Type IV tanks do not have coils.

- SRS is also monitoring the following new technologies:

    - Laser technology for final mapping

    - Miniaturized, high-definition sonar to help look beneath the waste level for interferences

    - Remote drilling methods to provide more access points into tanks

- DOE is cooperating with Hanford on the development of mixing models that can predict different slurry behavior. This is still in the beginning stages, but is a growing area with potential to enhance cleaning efforts at SRS. DOE clarified in its RAI response that the mixing model activities do not appear to have specific application to the SRS waste tanks but that DOE will continue to interface with their counterparts at Hanford, and that DOE will provide updates during the annual technology briefing to SCDHEC (Response to RAI-MEP-3; SRR-CWDA-2013-00106, Rev. 1).

- SRS continues to develop alternative mixers, with the goal of having a single mixer for all phases of waste removal. The alternative mixer would also ideally be capable of mixing at lower liquid levels at which the SMPs cannot operate due to aerosolization concerns (e.g., see NRC staff's concerns with respect to use of SMPs, particularly during oxalic acid treatment, in Tanks 5 and 6 closure module comments [Felsher, 2013a {ML13081A050}]).

- SRS continues to develop sampling crawlers and techniques for sampling minimal residuals.

- SRS is optimizing the chemical cleaning flow sheet, tailoring the process for each tank, and applying caustic and acid cleaning as the primary tools.

- DOE indicated that Low-Temperature Aluminum Dissolution was never intended to target HRRs directly, but instead was meant to change the rheology of the waste in order to facilitate future removal of HRRs (Shaffner, 2013e [ML13183A410]). In Tank 12, DOE has used Low-Temperature Aluminum Dissolution after bulk mechanical removal and prior to oxalic acid cleaning. Only aluminum is assumed to dissolve and all other solids are assumed to remain constant (X-CLC-H-00921).

Since challenges were encountered related to the use of oxalic acid in FTF Tanks 5 and 6, the NRC staff requested that DOE clarify how limitations associated with the use of oxalic acid might impact the cumulative number of HTF tanks that can undergo chemical cleaning with oxalic acid, and the likelihood that oxalic acid will remain part of the technology baseline for cleaning of future tanks. The NRC staff requested that DOE clarify other technologies that DOE may be pursuing as an alternative, given that enhanced chemical cleaning, which was explored

by DOE to reduce the impacts from the use of oxalic acid (e.g., operational capacity, formation of oxalates), is no longer being pursued (RAI-MEP-1; Mohseni, 2013a [ML13196A135]).

DOE stated that it is still evaluating the effect of oxalic acid on the rheology of the material and potential improvement in the mechanical removal of the insoluble solids. DOE has commissioned a study that will evaluate the downstream impacts of oxalic acid cleaning on the liquid waste system compared to the benefits gained by utilizing oxalic acid in the cleaning process. The study will include data from chemical cleaning campaigns in Tanks 12, 16, and 24 in HTF and Tanks 5 and 6 in FTF, as well as existing research information to evaluate the overall effectiveness of oxalic acid cleaning in order to determine the role of oxalic acid, as well as of enhanced chemical cleaning, in future heel removal efforts for both HTF and FTF (Response to RAI-MEP-1; SRR-CWDA-2013-00106, Rev. 1). Based on the results of the evaluation, DOE will update, as needed, the current baseline technologies and documentation regarding waste removal technologies for use at SRS. The NRC staff will review the oxalic acid study as part of its HTF monitoring activities.

Most of the baseline technologies, with the exception of the chemical cleaning with oxalic acid, involve removing waste volume, as opposed to preferably removing certain 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 estimates that the current waste cleaning process removes more than 99 percent of the waste.

In addition to the mechanical technologies used for cleaning Tanks 5, 6, and 17-20 in FTF (Camper, 2011 [ML112371751]), the following is a summary of the mechanical technologies that DOE has either used or plans to use in HTF (DOE-SRS-WD-2013-001):

- Tank 10—technical design currently underway for mechanical bulk waste removal; after completion of the technical design, bulk waste removal and heel removal will be completed.

- Tank 11—bulk waste removal completed using several slurry pumps; awaiting heel removal.

- Tank 12—bulk waste removal completed using several slurry pumps; slurry pumps continue to be used for heel removal.

- Tank 13—bulk waste removal ongoing using several SMPs; heel removal will be completed after bulk waste removal.

- Tank 16—bulk waste removal completed using several slurry pumps; heel removal successfully completed using a combination of slurry pumps and chemical (oxalic acid) cleaning.

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. 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, Rev. 1). The NRC staff

recommends DOE to continue evaluating new technologies for future use as tank closure progresses, especially if previously used technologies are no longer practical to use. Furthermore, for those tanks in which conditions are dissimilar (e.g., Tank 48) the NRC staff would expect DOE to re-evaluate technologies as opposed to relying on previously performed technology evaluations.

The focus of the NRC staff'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. The NRC staff thinks that DOE can place more emphasis on the development of technologies, such as an alternative to oxalic acid that would have less downstream impacts. New technologies such as the mixing models, robotic arm, and jet spray technologies may help facilitate removal of low volume but high specific activity material. The NRC staff acknowledges the inherent challenges in removing the last increments of material that add significantly to the residual source term. The 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 the NRC staff that technology selection should consider the long-term risk from HRRs left in HTF tanks at closure.

3.7 Removal to the Maximum Extent Practical

Removal of the HRRs to the MEP occurs through a sequence of the following cleaning activities (specific activities will vary depending on tank type):

- Bulk waste removal
- Mechanical heel removal
- Chemical cleaning
- Cooling coil flushing
- Annulus cleaning
- Final sampling

In its draft basis document for the WD for HTF (DOE-SRS-WD-2013-001, Rev. 0), DOE presents historical information on its waste retrieval activities. Historically, DOE has performed bulk sludge removal on Type I Tanks 9, 10 and 11; Type II Tank 15; Type IV Tanks 21 and 22; and Type IIIA Tank 42. DOE has also historically performed bulk saltcake removal on Type IV Tanks 22 and 24 and Type IIA Tanks 37 and 42. These tanks were subsequently re-used to support ongoing operations, but the prior experience with cleaning is still relevant to future cleaning efforts.[6] Table 3-14 summarizes the current cleaning efforts (as of publication of this TER).

[6] Tank 16 also underwent an extensive waste removal campaign and was discontinued from service in 1972.

Table 3-14 Current Tank Cleaning Efforts

	No Final Cleaning Efforts	Bulk Waste Removal	Mechanical Heel Removal	Chemical Cleaning	Cooling Coil Flushing	Annulus Cleaning	Final Sampling
Type I							
Tank 9	x						
Tank 10		x					
Tank 11		x					
Tank 12		x	x				
Type II							
Tank 13		x					
Tank 14	x						
Tank 15	x						
Tank 16		x	x	x		x	x
Type IV							
Tank 21	x						
Tank 22	x						
Tank 23	x						
Tank 24	x						

While Section 3116 of the NDAA 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 97 percent of the HRR inventory in HTF. 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 to determining when to terminate cleaning operations consists of the following phases: initial technology selection, technology implementation, technology execution, technology effectiveness evaluation, and additional technology evaluation. In Appendix B of DOE/SRS-WD-2013-001, Rev 0, DOE provides details on the cleaning approach used for each of the tanks and the ancillary structures.

Each stage is briefly summarized below, and a figure from DOE/SRS-WD-2013-001, Rev. 0 that illustrates DOE's approach is replicated in Figure 3-1:

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, as well as identifies specific metrics and data collection requirements for indicating effectiveness (e.g., radiation levels on transfer lines, solids concentration removed).

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

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

5. DOE 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.

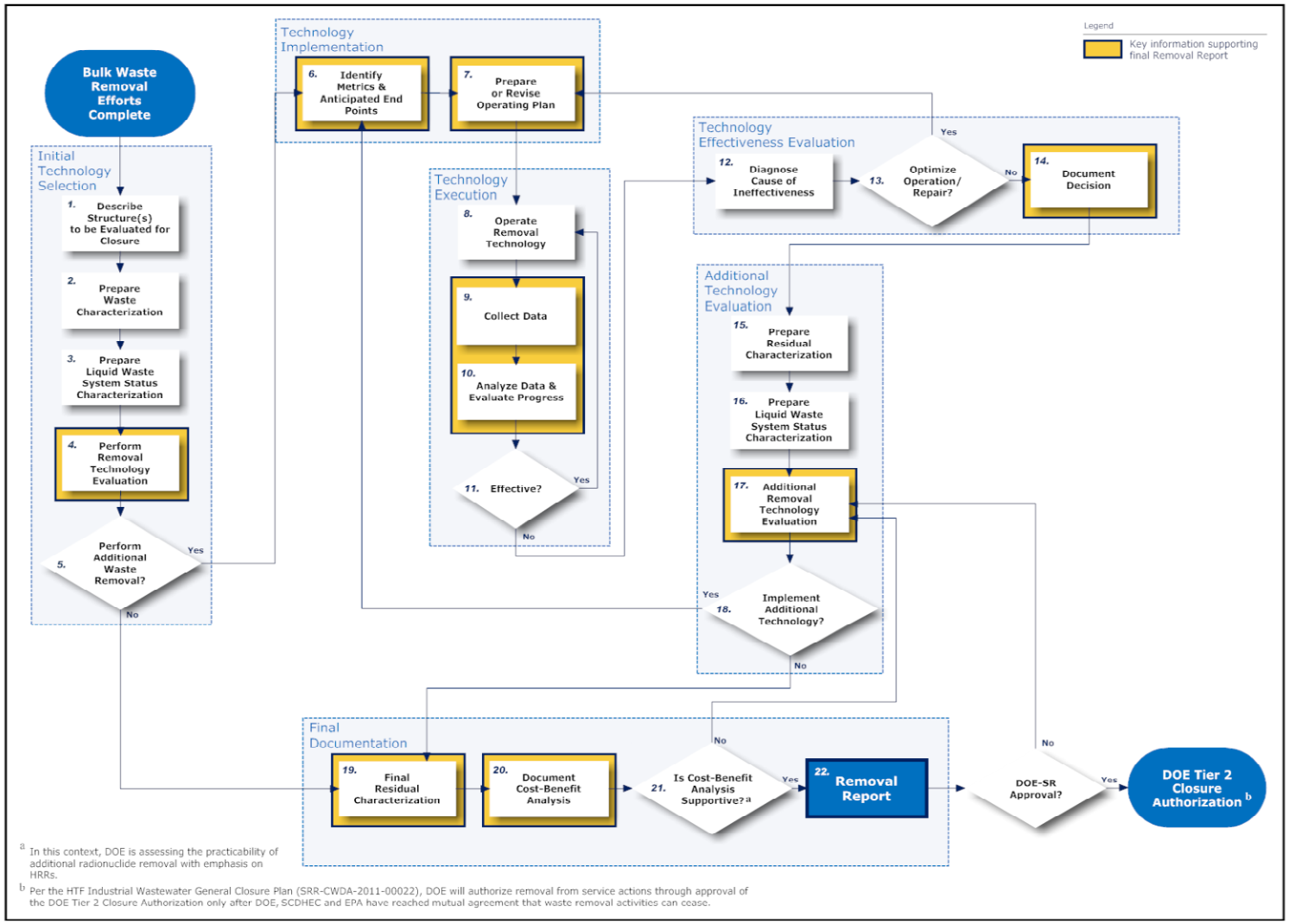
2. Identification of removal technologies (HRR-specific removal technologies, as well as overall volume reduction technologies).

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 SCDHEC and EPA. If the three agencies (DOE, SCDHEC, and EPA) concur, DOE will suspend waste removal activities. Then DOE will perform final characterization of the tanks or ancillary structures and evaluate the costs and benefits of additional cleanup of utilizing the final inventory. Considering the final radionuclide inventory, DOE will conduct an SA to ensure that the HTF PA conclusions are still valid. The same approach used to perform the SA for FTF Tanks 5, 6, 18, and 19 will be used for other cleaned tanks in the future, including those in HTF. 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 rationale for evaluating removal to the MEP will be informed by the SA, as well as the costs and risks associated with additional cleaning.

Figure 3-1 Approach for Documenting Radionuclide Removal to the Maximum Extent Practical



Adapted from Figure B.3-1 in DOE/SRS-WD-2013-001, Rev. 0.

DOE's approach for documenting removal to the MEP is shown in Figure 3-1. A final report documenting removal of the HRRs to the MEP may be written that includes summaries of more than one tank or ancillary structure if several are removed from service at the same time. The report will document the revised dose estimates using actual inventories from all waste tank systems that have been removed from service. DOE will use the knowledge gained from these actual inventories to adjust the estimated inventory of those waste tank systems that have not yet been removed from service. This approach has been completed for Type IV Tanks 18 and 19 and for Type I Tanks 5 and 6, as described in the basis document for the FTF WD and the Tanks 5 and 6 Closure Module (DOE/SRS-WD-2012-001, Rev. 0; SRR-CWDA-2012-00071, Rev. 0). The NRC staff reviewed DOE's approach for the FTF and provided comments and recommendations with respect to Tanks 18 and 19 in the FTF TER (Camper, 2011 [ML112371751]) and performed a technical review of the updated Tank 18 cost benefit analysis during monitoring (Parks, 2013 [ML13080A401]). The NRC staff also provided comments on the Tanks 5 and 6 Closure Module to SCDHEC in fulfillment of its monitoring responsibilities under the NDAA (Felsher, 2013a [ML13081A050]). At the time this HTF TER was developed, heel removal had been completed for Tank 16 and had been initiated for Tank 12 at HTF.

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. Ancillary equipment may be closed in conjunction with an individual tank, with a group of tanks, or may be 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 for Type I and Type IV waste tanks. Type III and Type IIIA waste tanks will remain in service until they are no longer needed to support waste treatment. The dense array of permanently installed piping in Type I, II, and IIIA tanks is expected to have more of an impact on waste removal efforts than the deployable cooling coils installed in the Type III tanks. The deployable cooling coils are fairly localized within the waste tanks and do not provide significant impediments at the bottom of the tank.

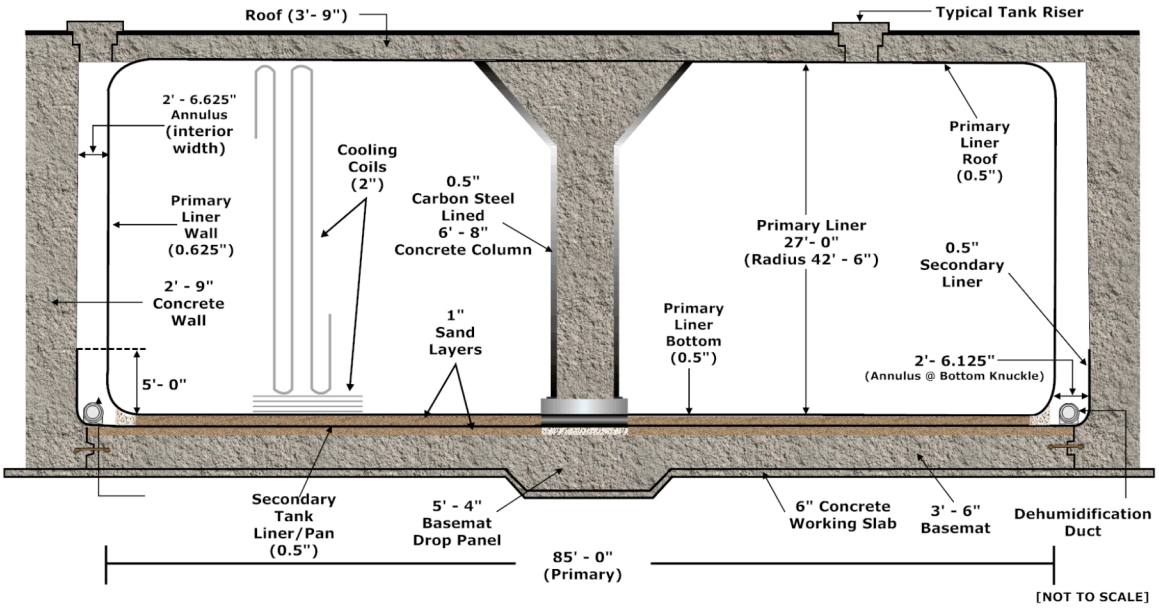
3.7.2 Cleaning Progress with Tank 16 and Tank 12

To date, none of the HTF Tanks have undergone closure. Tank 16 (Type II) and Tank 12 (Type I) at HTF have undergone heel removal activities, which are summarized in this section. Several FTF tanks, however, have undergone closure. While the FTF tanks do not fall under the scope of this draft basis for the WD for HTF, DOE's experience cleaning the FTF tanks is relevant to HTF since DOE is relying on the FTF experience as a basis for demonstrating the waste in HTF tanks will be removed to the MEP. Tanks 5, 6, 18, and 19 at FTF have been grouted and are currently undergoing closure. DOE also cleaned, removed from service, and grouted FTF Tanks 17 and 20 prior to the passage of the NDAA, therefore, different criteria were used and the experience is not relevant to this review (DOE/SRS-WD-2010-001, Rev. 0).

3.7.2.1 Tank 16

The construction of Tank 16 is described briefly in this section in order to better understand the location of residual material in Tank 16 and the practicality of removing additional waste. Tank 16 is a Type II Tank, with an annulus containing a secondary liner (annulus pan) and dehumidification/ventilation duct as depicted in Figure 3-2. The four Type II tanks are co-located in a four-tank area at the same elevation and a 15-cm (6-in) concrete construction working pad was installed over the entire four-tank area. The pad extended some 6 m (20 ft) beyond the edge of the tanks and provided a working surface for construction equipment. The basemat, which sits on top of the concrete working slab, is reinforced concrete and is 1.07-m

Figure 3-2 Type II Tank Cross Section

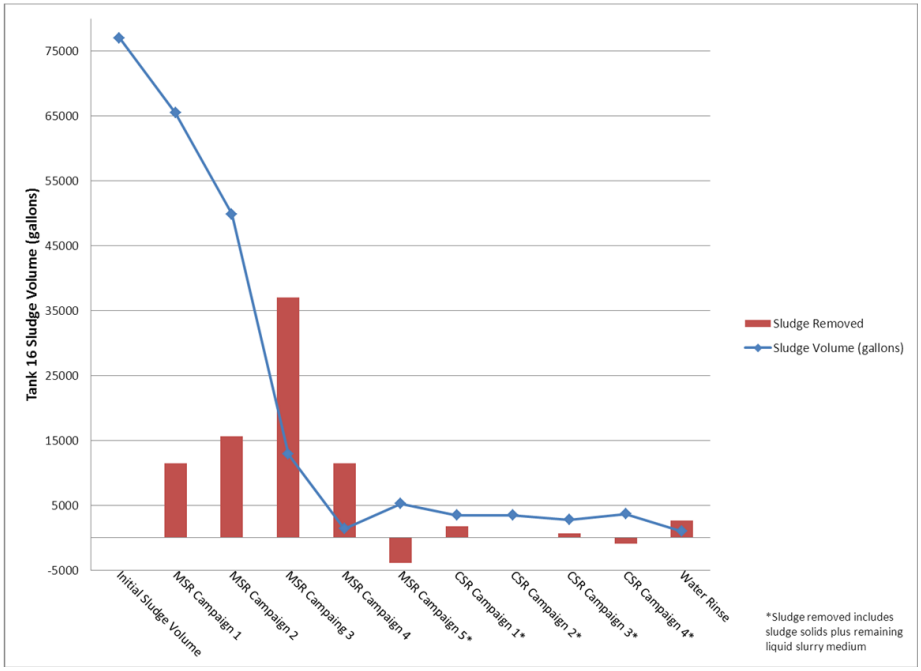


Adapted from Figure 2.1-35 in DOE/SRS-WD-2013-001, Rev. 0.
To convert feet to meters, multiply by 0.3048.
To convert inches to centimeters, multiply by 2.54.

(42-in) thick near the edges and thicker under the center column. A layer of sand was spread to a uniform depth of 2.54 cm (1 in) on top of the basemat. The annulus pan, which sits on top of the sandpad layer, is approximately 27.5 m (90 ft) in diameter and 1.5-m (5-ft) high. The annulus space is approximately 90-cm (36-in) wide with the ventilation duct at the bottom. The ventilation duct diameter ranges from 50 cm (20 in) to 30 cm (12 in). A 15-cm (6-in) by 10-cm (4-in) stiffener angle with lead flashing wrapped around it was welded to the top of the steel annulus pan. The purpose of the flashing is to divert any water away from the concrete vault and into the steel annulus pan (DP-1358).

In 1978, Tank 16 was the first tank to use slurry pumps for mechanical solids removal. Over the course of five mechanical sludge removal campaigns, DOE reduced the volume of material from 290 m^3 (77,000 gal) to approximately 19 m^3 (5,000 gal). The mechanical sludge removal phase was followed by four chemical sludge removal (CSR) campaigns, which further reduced the volume to 14 m^3 (3,680 gal). Finally, DOE applied a water rinse, which reduced the final volume to less than 3.78 m^3 (1,000 gal). DOE assumes that 3.78 m^3 (1,000 gal) is remaining in the tank for the purposes of the HTF PA, but believes that the actual amount is less. The waste removal activities are summarized in Figure 3-3, which is taken from SRR-CWDA-2011-00126, Rev. 0. Tank 16 is unique in that a significant leak occurred from the primary tank into the annulus. Historical records about the nature of the leak are relevant because they provide information regarding the potential flow paths through the annulus into the environment. In 1960, shortly after Tank 16 was put into operation, approximately 700 m^3 (185,000 gal) of waste leaked from the Tank 16 primary into the annulus. Most of the waste that leaked into the annulus was pumped out via the transfer pump in the north riser and transferred to Tank 14 (SRR-CWDA-2011-00126). Of the 700 m^3 (185,000 gal), DOE calculates that approximately 2.6 m^3 (700 gal) of waste rose above the top of the annulus pan for about six hours during the leak.

Figure 3-3 Tank 16 Removal Summary



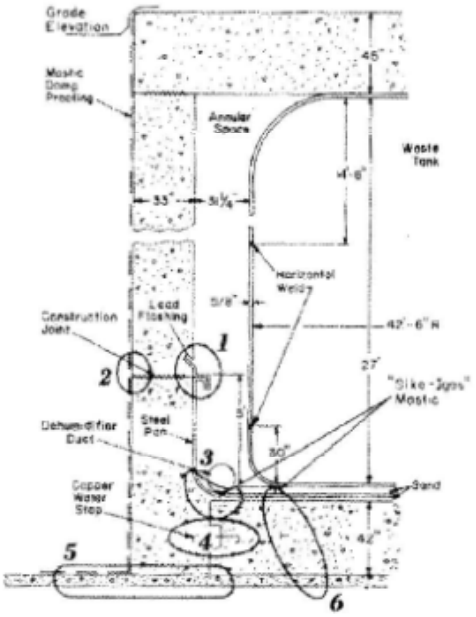
Adapted from Figure 3.5-2 in SRR-CWDA-2011-00126, Rev. 0.
To convert gallons to cubic meters, multiply by 3.785×10^{-03}.

Investigations and monitoring in 1974 estimated that 0.06 m^3 (16 gal) of waste escaped the concrete encasement into the surrounding soil. DOE predicts that the mostly likely path through which the waste escaped the concrete encasement was the horizontal construction joint around the circumference about 5 feet from the bottom of the tank. DOE calculates the maximum capacity of the construction joint to be about 0.06 m^3 (16 gal). See Figure 3-4 for a profile of the annulus and the location of the construction joint (DP-1358).

DOE estimates that flow velocities on the order of 3 to 6 m/hr (10 to 20 ft/hr) would have been required for the full 2.6 m^3 (700 gal) of waste that rose above the annulus pan to have escaped (1) horizontally through the space between the upper and lower concrete encasements (Path #4 of Figure 3-5), or (2) down through the path between the concrete encasement and the annulus pan (Path #5 of Figure 3-5). The 1974 report (DP-1358) states that flow velocities of this magnitude could not have been developed with the estimated head available, and therefore, only a small fraction of the 2.6 m^3 (700 gal) most likely ever escaped the annulus. Instead, DOE assumes that the majority of the waste which leaked from the primary into the annulus was pumped out of the annulus when material was transferred to Tank 14.

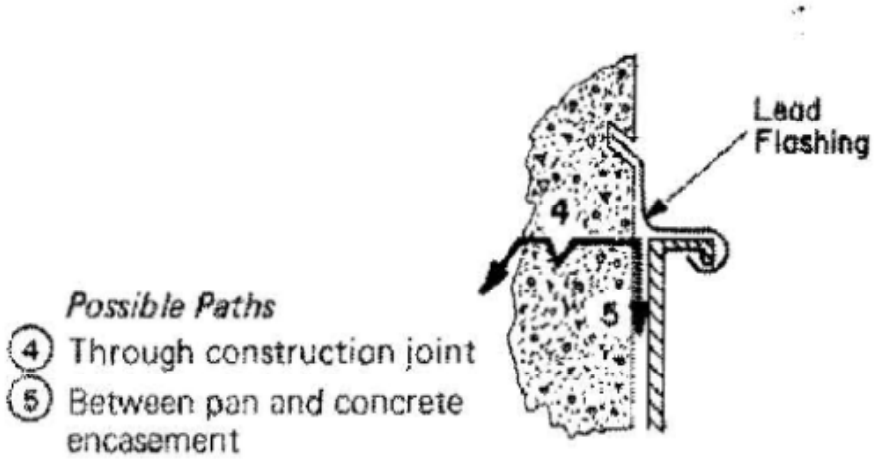
In 1974, DOE introduced approximately 0.5 m^3 (20 ft^3) of sand into the annulus during sandblasting of the tank walls to identify leak sites. In 1976, about 75 percent of the sand was vacuumed out of the annulus; in 1977, process water was added to the annulus to dissolve the saltcake into solution and transfer it out of the annulus through the transfer jet in the north riser. Steam jets were used in the east, west, and south risers to aid in mixing (see Figure 3-6. This effort removed approximately 5 m^3 (1,400 gal) of saltcake. After annulus cleaning was suspended in 1978, DOE collected samples of waste under inspection ports (IP) 118, 151, 207, and 272, which indicated that the annulus contained mainly water insoluble sand (DPSP-80-17-21).

Figure 3-4 Section of Tank 16 Annulus and Concrete Encasement



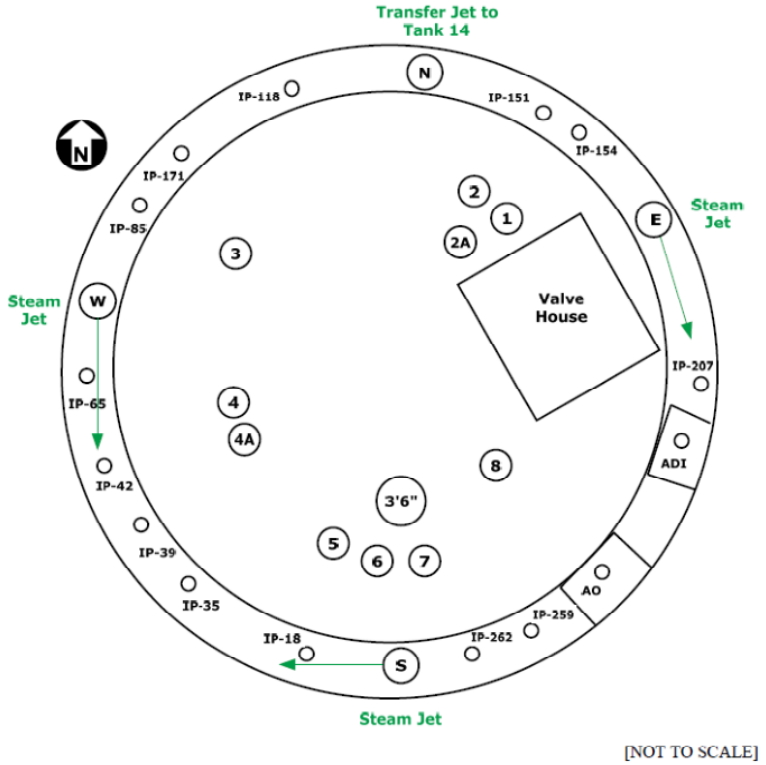
Adapted from Figure 25 in DP-1358.
To convert feet to meters, multiply by 0.3048.
To convert inches to centimeters, multiply by 2.54.

Figure 3-5 Area 1 Close-Up of Possible Flow Paths out of Tank 16 Annulus



Adapted from p. 62 in DP-1358.

Figure 3-6 Tank 16 Jet Configuration for 1977 Annulus Cleaning



Adapted from Figure 3.6-2 in SRR-CWDA-2011-00126, Rev. 0.

In 2007, DOE estimated that approximately 18 m^3 (4,700 gal) of residual material remained in the annulus and it initiated a project to identify and demonstrate a technology capable of dislodging and removing waste from the Tank 16 annulus. As part of the effort, DOE collected three additional samples from the annulus. One sample was collected from IP-118, and two samples were collected from IP-35 (one from inside the dehumidification duct and one from outside the duct) (see Table 3-15). The results from outside the dehumidification duct at IP−35 show a large fraction of the sample to be soluble in water (45-65 wt%), however, the sample from inside the duct was even more soluble in water (60-70 wt%). The IP-35 sample outside the duct also had a composition similar to that of the sample from inside the duct but with considerably more silicon present. The amount of soluble sodium salts in sample IP-118 (25-25 wt%) appears to be much lower than the two IP-35 samples. This may indicate that washing was more effective in the north area of the annulus near IP-118. It also indicates that aluminosilicates in the waste may have formed as a result of the sand left behind from sandblasting operations conducted in the Tank 16 annulus. Overall, the sampling results indicate that the waste material in Tank 16 annulus may have a wide range of compositions at different locations (WSRC-STI-2008-00203, Rev. 0). In November 2011, DOE revised the estimate of the remaining material in the Tank 16 annulus to be 12.5 m^3 (3,300 gal) based on camera views and samples taken. See Section 3.1 of this TER for details on the projected inventory for the Tank 16 annulus.

Table 3-15 Estimate of the Solubility of the Material in the Tank 16 Annulus Samples

Sample ID	Estimate of the Total Dried Solids Soluble in Water (wt%)	Sample Location in Annulus
HTF-16-06-104	60-70	From IP-35 inside of duct
HTF-16-06-105	45-65	From IP-35 outside of duct
HTF-16-06-106	25-35	From IP-118 outside of duct
Adapted from WSRC-STI-2008-00203, Rev. 0.		

DOE historical records indicate the agency's intent to remove additional waste from the Tank 16 annulus. DOE's intent is shown by (1) its investigation of annulus cleaning throughout the 1980s, 1990s, and 2000s as indicated by the sampling efforts in 1995 and 1998, as well as (2) demonstration activities in 2007. These prior cleaning efforts are discussed in Section 3.8.1 of this TER. Although DOE previously intended to remove additional waste from the annulus, in 2013, DOE completed an evaluation to determine whether to cease annulus waste removal activities for Tank 16. In the 2013 evaluation, DOE concluded that it would cease waste removal activities based on the high worker dose and costs associated with removal compared to DOE's estimated potential reduction in long-term risk to a member of the public that is calculated in the HTF PA (U-ESR-H-00107, Rev. 0).

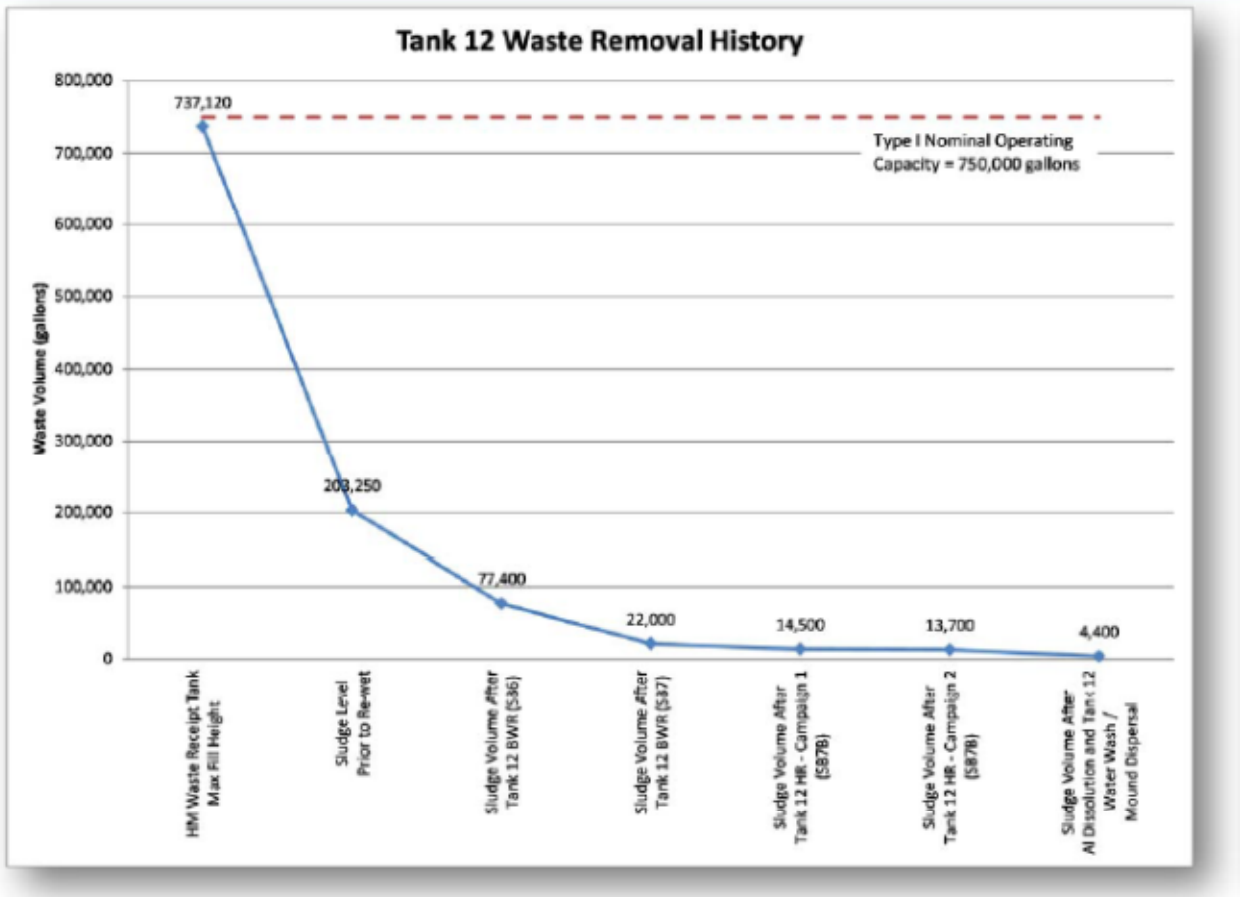
3.7.2.2 Tank 12

At the time DOE drafted the basis document for the WD for HTF, Tank 12 had already undergone bulk and mechanical heel removal and was in the chemical cleaning phase. As shown in Figure 3-7, upon completion of the bulk waste removal efforts, DOE estimated the total volume of sludge remaining to be approximately 83 m^3 (22,000 gal). Mechanical heel removal utilizing four slurry pumps, initiated in 2010, resulted in removal of approximately 30 m^3 (8,000 gal) of sludge. In 2011, DOE deployed the Low-Temperature Aluminum Dissolution technology, followed by a water washing campaign, because the solids content in Tank 12 had a high concentration of aluminum. The volume of sludge remaining at the conclusion of the Low-Temperature Aluminum Dissolution and subsequent water washes was estimated to be approximately 16 m^3 (4,400 gal) (DOE/SRS-WD-2013-001, Rev. 0).

DOE incorporated several lessons learned from experience with FTF Tanks 5 and 6 into the plan for chemical cleaning of HTF Tank 12 that addressed many of the NRC staff's comments on the FTF Tanks 5 and 6 Closure Module (Felsher, 2013a [ML13081A050]), such as the following:

- 4 weight percent (wt%) is the most advantageous concentration of acid for performing cleaning in the tank. Acid will be added at 8 wt% and diluted down to 4 wt%.
- Acid will be introduced into the tank at a temperature of approximately 55 °C (± 5 °C).
- The pH of the material will be maintained to be less than 2 to prevent the material from precipitating out, thereby, reducing the effectiveness of the oxalic acid.
- DOE is maximizing contact of residuals with oxalic acid by providing adequate mixing with four slurry pumps and three strikes.

Figure 3-7 Tank 12 Waste Removal History



Adapted from Slide 4 in SRR-STI-2013-00198.
To convert gallons to cubic meters, multiply by 3.785×10^{-03}.

3.8 NRC Evaluation of Removal to Maximum Extent Practical

3.8.1 NRC Evaluation of Cleaning Process and Criteria for Termination of Cleaning Operations

The NRC staff reviewed DOE's technology implementation and optimization, as well as DOE's approach for evaluating technology effectiveness. Based on its review of DOE's FTF Tank 5 and 6 Closure Module (Felsher, 2013a [ML13081A050]), the NRC staff requested that DOE describe its approach to developing an implementation strategy for a cleaning technology, including the steps and criteria used in decisionmaking (RAI-MEP-3; Mohseni, 2013a [ML13196A135]). The NRC staff requested that DOE include a specific example of how this generic approach is implemented with the development of the pumping/mixing strategy for a specific tank. In response, DOE explained that waste removal strategies must consider each tank's unique existing conditions at the start of heel removal, as well as the liquid waste system conditions and area-specific logistical capabilities such as electrical feed constraints. DOE described the specific factors that are considered to determine the type, quantity, and location of mixer pumps. DOE also described the specific factors that are considered when determining the initial plan for the pump run programs (i.e., indexing angle, fixed versus oscillating). In establishing the pump run program, DOE considers the location of cooling coils, the location of

waste mounds, and whether or not the mounds are known to have been previously disturbed. DOE adjusts the pump run at the end of each campaign after evaluating the changes during the previous campaign (Response to RAI-MEP-3; SRR-CWDA-2013-00106, Rev. 1).

DOE specified that for future waste tank cleanings, the operating plans for heel removal activities will describe the specific technology to be implemented (e.g., mechanical mixing with SMPs, oxalic acid cleaning, or a combination of technologies) and the methods of implementation. Also, operating plans will include the planned flowsheet, including mixing strategies, chemical cleaning strategies, and lessons learned from earlier waste removal efforts, as applicable (Response to RAI-MEP-3; SRR-CWDA-2013-00106, Rev. 1).

During its review, the NRC staff noted in an RAI that DOE's approach to optimization of technology through sampling and monitoring during cleaning should be documented (RAI-MEP-4; Mohseni, 2013a [ML13196A135]). In response to the RAI, DOE indicated that the operating plans will also document the metrics to be used to track waste removal progress in order to determine continued effectiveness of the technology. Examples of the specific metrics include: radiation levels on transfer lines, current drawn by a mixer pump, density readings for a solution, solids concentrations being removed, and video/photographic evidence. DOE reiterated that a predetermined set of metrics and criteria has not been established, but stated that metrics will be standardized to the extent possible as more experience is gained (Response to RAI-MEP-4; SRR-CWDA-2013-00106, Rev. 1).

DOE provided the NRC staff with its procedures for the development of operating plans and implementation of the documentation approach provided in Appendix B of the draft basis document for the WD for HTF (ENG.50, Rev. 2; ADM.53, Rev. 0). The NRC staff reviewed the procedures and agrees that they will improve the consistency with which DOE documents its decisions regarding technology implementation and optimization.

The NRC staff requested that DOE clarify the approach to evaluating the effectiveness of cleaning technologies because DOE was more successful in cleaning Tank 16 than with FTF Tanks 5 and 6, but it had not critically evaluated the factors that led to a more successful cleaning (Felsher, 2013a [ML13081A050]; RAI-MEP-5, Mohseni, 2013a [ML13196A135]). DOE evaluates effectiveness through both changes in waste volume and changes in radionuclide concentrations. DOE evaluates changes in volume using the tank mapping methodology improvements discussed in Section 3.2.4 throughout the various cleaning activities (Response to RAI-MEP-5; SRR-CWDA-2013-00106, Rev. 1). The NRC staff notes that, although the results from mapping contain uncertainties, performing the tank mapping methodology during multiple cleaning phases will provide additional information on the effectiveness of specific technologies. As such, the NRC staff recommends that DOE perform the tank mapping consistently and as frequently as practical throughout the cleaning process.

Evaluation of a change in radionuclide concentration is more difficult because it requires sampling and analysis of the material at various stages of cleaning. DOE indicated that it has not yet identified a specific set of criteria for use in determining the optimal intermediate sampling program which would be applicable to all waste tanks. DOE will analyze the benefits of intermediate sampling as it gains experience, and will establish sampling criteria, as appropriate. The NRC staff also noted in its comments on the FTF Tank 5 and 6 Closure Module that DOE should obtain better baseline information from which it could better assess oxalic acid effectiveness (Felsher, 2013a [ML13081A050]). Prior to the oxalic acid cleaning on Tank 12, a sludge sample was taken from the waste tank because the location of the waste

within the tank allowed for the sample to be relatively easily collected. DOE will use this sample along with the final sludge samples to help evaluate the effectiveness of oxalic acid cleaning in Tank 12.

In addition to reviewing generic information about DOE's approach to technology implementation, optimization, and evaluation, the NRC staff also reviewed the specific lessons learned from the removal activities at FTF Tanks 5 and 6, as recommended in the NRC staff's Tanks 5 and 6 Closure Module comments (Felsher, 2013a [ML13081A050]). One specific issue DOE encountered was limitations with SMPs due to low liquid levels. In its review of the FTF Tanks 5 and 6 Closure Module, the NRC staff pointed out that DOE documentation indicates that a minimum liquid level of around 0.8 to 1.1 m (30 to 45 in) is needed to prevent cavitation and aerosolization of the waste during SMP operation. The report SRR-CWDA-2012-00071, Rev. 0 (p. 57) indicates that waste aerosolization may occur above a liquid level of 30 inches for SMPs. Because the SMPs could not be operated at lower liquid levels, ineffective mixing during acid strike two in Tanks 5 and 6 appears to have contributed to the formation of solids during chemical cleaning.

Given the limitations with the SMPs at low liquid levels, the NRC staff inquired as to whether DOE was pursuing a low-volume mixing pump technology (RAI-MEP-6; Mohseni, 2013a [ML13196A135]). In response, DOE clarified that it is in communication with personnel at the Sellafield Site in the United Kingdom regarding the pulse-jet mixers used at Sellafield that allow for mixing at low tank levels. However, the pulse-jet mixers are designed for small vessels and would not be effective on SRS tanks. At this time DOE is not pursuing development of a low-volume pump for use in SRS tanks (Response to RAI-MEP-6; SRR-CWDA-2013-00106, Rev. 1). To help overcome the limitations encountered with cleaning Tanks 5 and 6 for the cleaning of future tanks, the NRC staff recommends that DOE evaluate the effectiveness of the SMPs with respect to bulk sludge removal versus residual heel removal. The NRC staff also recommends that DOE compare the efficiency and effectiveness of the SMP to previously used technologies or readily available technologies.

The NRC staff also requested that DOE clarify the reasons for the long residence time of oxalic acid during CSR strike two for FTF Tanks 5 and 6 while there was no mixing (RAI-MEP-6; Mohseni, 2013a [ML13196A135]). The long residence time of oxalic acid was a contributing factor to the formation of oxalates in these tanks, due to the precipitation of metal oxalates as the pH increased over time. DOE clarified that oxalic acid's extended residence time was the result of an upper mechanical seal failure in one of the standard, long-shafted 150-hp slurry pumps in FTF Tank 7 that delayed the Tank 5 to Tank 7 transfer and the Tank 6 to Tank 7 transfer. To address this specific concern with Tank 12, DOE added a requirement to the operating plan for Tank 12 that required confirmation that the transfer receipt tank was available prior to acid addition to Tank 12, in order to facilitate timely transfers (Response to RAI-MEP-6; SRR-CWDA-2013-00106, Rev. 1).

As discussed above, DOE incorporated several lessons learned from the cleaning of FTF Tanks 5 and 6 into the cleaning strategy for Tank 12, to minimize the formation of oxalates. For example, the sludge heel in Tank 12 was pre-washed prior to adding oxalic acid to reduce the sodium molarity; the liquid level in Tank 12 was increased during each of the three oxalic acid strikes to allow the pumps to run at maximum speed; four SMPs were run at maximum speed for longer periods; and the pH of the material in Tank 12 was kept below 2 prior to being transferred from the waste tank for each of the three strikes. DOE states that preliminary review

of video and photographs within Tank 12 following the oxalic acid campaign indicate that significant precipitation of the oxalates did not occur (Response to RAI-MEP-6; SRR-CWDA-2013-00106, Rev. 1).

In Appendix B of the draft basis for the WD for HTF (DOE/SRS-WD-2013-001, Rev. 0), DOE outlines a general approach to demonstrate that the HRRs will be removed to the MEP. Over the course of FTF monitoring activities and the HTF consultation, the NRC staff believes that DOE has improved the level of detail regarding the specific implementation of the process. For example, DOE has formalized the tank mapping methodology and has developed procedures for tank cleaning operating plans. However, DOE could still improve the standardization of metrics for determining that the anticipated end states have been reached. Without a standardized set of metrics, the distinction between anticipated end states and removal goals is unclear. Also, it is the view of the NRC staff that the practicality of additional waste removal depends upon a continued evaluation of the availability and maturity of alternative cleaning technologies, as opposed to a previously performed technology evaluation. For this reason, the NRC staff supports DOE's efforts to re-evaluate oxalic acid cleaning against downstream impacts to determine the future role of oxalic acid cleaning, as opposed to relying on previous evaluations of oxalic acid technology.

If DOE elects to reduce the number of analytes (i.e., radionuclides and metals) in the final characterization of a given tank based on prior analyses that indicated that certain non-risk-significant, hard-to-detect radionuclides were not present in a another tank, DOE would need to provide evidence to support 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 a Quality Assurance and Quality Control Plan.

During the NRC staff's review, DOE noted that when conducting the SA there may not be a need to re-run models for every case if the final inventories are not significantly greater than the predicted inventories applied in the HTF PA. If the final inventory is less than that which was projected, DOE will rely on the PA results to determine the risk. The reasonableness of this approach relies on the adequate consideration of potentially risk-significant scenarios in the HTF PA. As discussed below in Section 3.8.2, the NRC staff does not have confidence that DOE has adequately evaluated the risk associated with the projected inventory of the Tank 16 annulus (see Section 4.2.9.3). The NRC staff acknowledges that the final characterization may show an inventory much less than what was projected. However, the NRC staff recommends that DOE appropriately evaluate the risk of that final inventory even if it is less than what was projected.

The 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 oxalic acid) to provide effectiveness measurements for chemical cleaning and mechanical feed-and-bleed (Mohseni, 2013a [ML13196A135]). It is important to note that volume estimates alone are not always the best measure of effectiveness and that not considering 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 Section B.3.2.6 of Appendix B of DOE/SRS-WD-2013-001, Rev. 0 regarding 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 remains unclear to the NRC staff if DOE will follow a consistent format with the appropriate content for the future cost-benefit analyses.

In many cases, cleaning processes were ceased, disrupted, or negatively impacted by the unavailability of waste receipt tanks or to avoid disrupting other ongoing demands for the liquid waste system. DOE noted that a SEE precedes all tank cleaning activites to attempt to avoid excessive demands on the liquid waste system, but unanticipated circumstances sometimes affect results. The NRC staff notes that DOE improved the operating plan for Tank 12 by requiring the availability of the transfer receipt tank to be confirmed prior to acid addition. The NRC staff 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.

The 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. 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 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. The NRC staff notes that DOE has a process in place to meet Criterion 2 of the NDAA as it proceeds with closure of tanks in the HTF. The NRC staff recommends that DOE continue to collaborate with the NRC as tank farm closure progresses. The NRC staff plans to monitor DOE's efforts in this area under the as low as reasonably achievable (ALARA) provisions of 10 CFR Part 61, Subpart C.

3.8.2 NRC Evaluation of Cleaning Progress for Tank 16 and Tank 12

3.8.2.1 Tank 16

It is the the NRC staff's position that the amount of material that remains in the Tank 16 annulus is potentially significant. While the following discussion pertains to Sr-90, the NRC staff is still evaluating the potential significance of other radionuclides (e.g., Cs-137 [$8.1\times10^{+14}$ Bq or ~21,900 Ci; Tc-99 [$2.6\times10^{+11}$ Bq or ~ 7 Ci], Pu isotopes [$2.8\times10^{+12}$ or ~75 Ci]) present in the annular contamination. DOE defines three annular contamination source terms for the Tank 16 annulus: (1) annulus contamination ($2.8\text{x}10^{+14}$ Bq or 7,800 Ci of Sr-90); (2) contamination ($1.1\times10^{+14}$ Bq or 3,100 Ci of Sr-90) within the 1-in (2.54-cm) sand pad layer between the primary and secondary liners (i.e., primary sandpad); and (3) contamination ($2.3\times10^{+12}$ Bq or 63 Ci of Sr-90) within 1-in (2.54-cm) sand pad layer between the secondary liner and the concrete basemat (i.e., secondary sandpad). These three sources result in approximately $4.1\times10^{+14}$ Bq (11,000 Ci) of Sr-90 outside of the primary containment for Tank 16. As a comparison, DOE estimates the inventory of Sr-90 in the residual material within the primary tank liner of Tank 15 to be $5.2\times10^{+14}$ Bq (14,000 Ci) or approximately equivalent to the total annular Sr-90 contamination associated with Tank 16 (DOE/SRS-WD-2013-001, Rev. 0).

The NRC and DOE staff met following the June 6, 2013, site visit to continue discussions regarding challenges associated with the quantification and removal of contaminants within the Tank 16 annulus. The discussion focused on Tank 16, but also included Tanks 9, 10, and 14 annulus materials. DOE estimates that the Tank 9, 10, and 14 annuli contain about 12, 4.2 and 21 m3 (3,300, 1,100, and 5,600 gal) of material, respectively, although it assumes 12.5 m^3 (3,300 gal) is in the annuli of all those tanks in the HTF PA (Shaffner, 2013e [ML13183A410]). Unlike Tank 16, DOE expects to be successful at cleaning the annuli of Tanks 9, 10, and 14 with traditional cleaning methods because it expects the material in those tanks to be more soluble.

Tank 16 is unique in that it was subject to significant leakage in its early days of operation and DOE has previously attempted to clean the annulus using sand blasting. DOE estimates that the contamination within the annulus of Tank 16 consists of three distinct regions in terms of characteristics of the waste. The first region (5 m^3 [1,300 gal]) lies on the annulus floor in the north. DOE believes this region has been chemically altered due to the addition of silica from sandblasting that was conducted for leak inspections and due to high heat from steam jets that were used for a previous cleaning campaign. The combination of the addition of silica, a high pH, and high heat is believed to have chemically altered the residual material leading to the formation of alumina-silicates that DOE believes combined with the radionuclides in the residual material to reduce their solubility. The second region (3 m^3 [800 gal]) is on the annulus floor in the south. This second region has also been chemically altered due to the introduction of sand, but less so than the north area. DOE has indicated it believes these two regions would be challenging to remove and potential removal options would involve nuclear safety concerns because they break up the material into particles that could be aerosolized within the annulus. A third region of annular material (4.5 m^3 [1,200 gal]) resides within the annulus ventilation duct. DOE expects the contamination within the ventilation duct to be more soluble because it was shielded from significant amounts of silica from sandblasting due to the ductwork. However, DOE has stated that limited access to the ventilation ducts would make removal of the material within the duct challenging (Shaffner, 2013e [ML13183A410]).

The NRC staff requested that DOE provide additional information on the practicality of removing additional material from the Tank 16 annulus, especially material from the ventilation duct since it is more soluble than the annulus floor material. In response, DOE provided the results of a qualitative analysis completed in 2013 to determine if additional technologies should be deployed to remove additional waste from the Tank 16 annulus (U-ESR-H-00107, Rev. 0). The report summarizes DOE's prior evaluation of alternative technologies and provides a recommendation on whether or not to cease cleaning efforts. The content of the report is briefly summarized in the following paragraphs.

In March 2007, DOE evaluated cleaning alternatives for the Tank 16 annulus. DOE issued an Expression of Interest and a Request for Proposal, to which three vendor teams responded. A proof of concept test was completed in 2007, but the project was suspended due to funding constraints. When funding was reinstated in 2010, the proposal proved to be more difficult and costly to implement, and would have taken longer than originally planned. Therefore, DOE re-examined alternative cleaning strategies. DOE evaluated (1) water dissolution and sluicing, and (2) chemical cleaning with oxalic acid. Water dissolution and sluicing was chosen as the preferred alternative based on operational simplicity, lower cost, and a favorable schedule. However, a mock demonstration in 2012 revealed that removal was likely to be less than originally anticipated and that aerosolization of the waste was a possibility (U-ESR-H-00107, Rev. 0).

In addition to describing the difficulties in using certain cleaning approaches, the qualitative analysis also summarizes DOE's estimated risk of the residual material using data from the HTF PA. DOE estimated that the Tank 16 remaining waste contributes a dose of less than 0.002 mSv/yr (0.2 mrem/yr) to a hypothetical future member of the public (U-ESR-H-00107, Rev. 0). The NRC staff is concerned that DOE has not appropriately evaluated the risk from annular waste in Tank 16. This concern is discussed in detail in Section 4.2.9.3.

DOE will provide more detailed documentation on the practicality of removing additional waste from the Tank 16 annulus in a Tank 16 Removal Report that will be issued after the completion of final residual characterization and subsequent cost-benefit analysis. The NRC staff recommends that DOE evaluate a waste release scenario due to groundwater in-leakage into and out of the annular region and contacting the high-solubility waste in the annuli of those tanks with residual material in the annulus and sandpads, including Tank 16. The NRC staff also acknowledges that it may be impractical to further clean the annulus of Tank 16 due to the recalcitrant nature of some of the annular contamination and nuclear safety concerns. As part of its monitoring responsibility, the NRC staff will review the Tank 16 final characterization data and documentation of removal to the MEP to ensure that DOE has adequately weighed the potential risk of this material against the practicality of additional removal.

3.8.2.2 Tank 12

Because DOE had not yet conducted chemical cleaning on Tank 12 as of the development of this TER, the NRC staff has not evaluated the progress of removing HRRs to the MEP from Tank 12 in great detail. The NRC staff has provided an evaluation of aspects of the Tank 12 cleaning approach as part of its review of the DOE cleaning process and criteria for terminating cleaning in Section 3.8.1.

3.9 NRC Conclusions for Criterion 2

The NRC staff evaluated DOE's demonstration of compliance with NDAA Criterion 2 including DOE's approach to (1) development of inventories for HTF tanks and auxiliary components; (2) identification of HRRs; (3) selection of treatment technologies; and (4) demonstration of removal to the MEP, including consideration of the costs and benefits of additional radionuclide removal. The 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 (Mohseni, 2013a [ML13196A135]), most of which were addressed in DOE's RAI responses (SRR-CWDA-2013-00106, Rev. 1).

The 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). The NRC staff will continue to monitor DOE's efforts towards reducing the HTF source term as it pertains to Criterion 3, including ALARA optimization.

- Although, DOE's approach to developing inventories for tanks is reasonable, although the approach to managing and quantifying uncertainty in sampling and volume estimates, could be improved (Barr, 2013a [ML13085A291]; Barr, 2013b [ML13273A299]).

- 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 tanks that have not yet been cleaned, DOE has a general process in place to demonstrate removal of HRRs to the MEP. DOE has refined the approach since it was first developed, but this process could still benefit from additional detail as DOE gains experience cleaning the tanks.

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

1. The NRC staff recommends that DOE explore methods to improve the process by which residual waste volumes and associated uncertainty are estimated. The NRC staff also recommends DOE evaluate and clearly communicate the relative contributions of various forms of uncertainty related to the sampling process on estimates of confidence bounds on average concentrations for individual radionuclides. DOE should analyze trends in projections versus actual inventories by radionuclide to update the multiplier assumptions for the probabilistic analysis.

2. The NRC staff recommends that DOE continue to evaluate its HRR list for HTF as additional information becomes available. The HRR list should be evaluated especially where it 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.

3. The NRC staff acknowledges DOE's efforts in understanding unexpected results and recommends that DOE continue to examine the reasons for such, should they occur, and attempt to trace them back to known waste streams or processes that might reveal other radionuclides that could have been underestimated by the projections based on WCS data. DOE should assess, through future tank residual characterization, the validity of prior assumptions and the resulting impacts to the list of HRRs.

4. As practical, the 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.

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

6. If oxalic acid is not available to be used for cleaning future tanks and a technology with similar proven effectiveness is not used as an alternative, the NRC staff recommends that DOE reconsider the validity of assuming that the cooling coil and tank wall surface inventory is negligible.

7. The 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 HTF facility.

8. The NRC staff recommends that DOE continue to refine the methods to be used to demonstrate removal to the MEP to ensure consistent (non-arbitrary) application of the criterion.

DOE has suspended waste removal and has entered into the final sampling and analysis phase for Tank 16. Prior to ceasing waste removal, DOE performed a qualitative analysis of the practicality of additional waste removal from Tank 16, which showed that additional removal was impractical (U-ESR-H-00107, Rev. 0). DOE, SCDHEC, and EPA concurred on the information presented in the qualitative analysis. The final volume and characterization information will be utilized by DOE to support a final decision on the practicality of additional waste removal from the Tank 16 waste tank system. DOE will document the final Tank 16 inventory, radionuclide removal effectiveness (with emphasis on HRRs), and final cost-benefit analysis in a final removal report. The final removal report is required before DOE would provide Tier 2 authorization and approval to stabilize (i.e., grout) Tank 16.

The primary reasons cited by DOE for why the additional cleaning of the Tank 16 annulus is impractical are (1) the site worker dose and safety risks far exceed the marginal reduction in potential dose to a future member of the public, and (2) the financial cost greatly exceeds the benefit. However, the NRC staff is not confident that DOE has adequately evaluated the potential dose to a future member of the public from the remaining material should it escape the near-field environment. The NRC staff acknowledges that projected inventory does not reflect the final characterization. However, even if the final inventories are lower than the projected inventory, the final inventories will not be significantly lower as to eliminate the potential risk should the annulus inventory be released into the environment.

Considering information in the uncertainty analysis, Tank 16 is a potential risk driver for the HTF facility. Given its risk significance, the NRC staff recommends that DOE should more fully evaluate the practicality of additional radionuclide removal from the Tank 16 annulus versus the long-term benefit of reduced risk for the reasons listed below. At this stage DOE has provided a rough order of magnitude cost-benefit analysis of additional HRR removal from the Tank 16 annulus to the NRC staff (U-ESR-H-00107, Rev. 0). The NRC staff acknowledges that DOE is still preparing the final removal report and recommends that DOE provide a more detailed cost-benefit analysis to support the Criterion 2 demonstration for Tank 16 in the final removal report. The NRC staff would like to obtain a copy of the final removal report when it is complete.

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

Section 3116(a) of the NDAA 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 this section referred to as the "Secretary"), in consultation, with the Nuclear Regulatory Commission (in this section referred to as the "Commission"), determines—"
>
> "(3) (A) does not exceed concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, and will be disposed of –
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations; and
>
> (ii) pursuant to a state-approved closure plan or state-issued permit, authority for the approval or issuance of which is conferred on the State outside of this Section; or
>
> (B) exceeds concentration limits for Class C low-level waste as set out in section 61.55 of title 10, Code of Federal Regulations, but will be disposed of –
>
> (i) in compliance with the performance objectives set out in subpart C of part 61 of title 10, Code of Federal Regulations;
>
> (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 DOE can determine whether Section 3116 (a)(3)(A) or (a)(3)(B) above applies, it must first determine whether the waste exceeds concentration limits for Class C LLW provided in 10 CFR 61.55. After applying the NRC's guidance on classification of waste that is incidental to reprocessing found in Section 3.5 of NUREG-1854, DOE believes that the stabilized HTF waste at closure will not contain concentrations greater than the limits for Class C waste. Therefore, NDAA Section 3116(a)(3)(A) applies to HTF waste (see Section 4.1). However, if DOE or the NRC concluded that the HTF waste is Greater-than-Class C, the additional requirement in NDAA Section 3116(a)(3)(B)(iii) would apply. As such, DOE is required by the NDAA to consult with the NRC in developing its disposal plans. DOE indicated in its draft basis document for the WD for HTF (DOE/SRS-WD-2013-001, Rev. 0) that while it believes there is a reasonable basis to conclude that none of the stabilized residuals at HTF will exceed the Class C concentration limits in 10 CFR 61.55, it nevertheless elects to take advantage of the consultation process established under NDAA Section 3116(a)(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 in Subpart C of 10 CFR Part 61 and pursuant to a state-approved closure plan or state-issued permit. An SCDHEC industrial wastewater construction permit, issued January 25, 1993, governs tank waste storage and removal operations for HTF closure (Sadler, 1993). Stabilization of the HTF waste tanks and ancillary structures will be carried out pursuant to the HTF General Closure Plan (SRR-CWDA-2011-00022, Rev. 0) approved by the State of South Carolina for all HTF 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 protection of the general population from releases of radioactivity, protection of individuals from inadvertent intrusion into the waste, protection of individuals during operations, and stability of the disposal site after closure. Protection of the general population (including inadvertent intruders) is typically demonstrated through a PA calculation that takes into account the relevant physical processes and the temporal evolution of the system. Section 4.2 of this TER presents the NRC staff's assessment of DOE's HTF PA (SRR-CWDA-2010-00128, Rev. 1) and the demonstration of compliance with the performance objectives specified at 10 CFR 61.41 (protection of the general population from releases of radioactivity) and 10 CFR 61.42 (protection of individuals from inadvertent intrusion). Section 4.3 presents the NRC staff's overall conclusion with respect to demonstrating compliance with the performance objectives specified at 10 CFR 61.43 (protection of individuals during operations) and 10 CFR 61.44 (stability of the disposal site after closure). Section 4.4 presents the NRC staff's overall conclusions related to Criterion 3 and lists key recommendations that are important to the compliance demonstration (and the NRC's ability to assess compliance with the 10 CFR Part 61 performance objectives).

4.1 Assessment of Waste Classification

Section 3116 of the NDAA lists two sets of criteria for non-HLW: Section 3116 (a)(3)(A) and (a)(3)(B). The applicable set of criteria is dependent on the classification of waste. If HTF waste is Class C (or less), then the criteria in (a)(3)(A) apply. If the waste is greater than Class C, then additional criteria provided in Section 3116 (a)(3)(B) apply. Tables 4-1 and 4-2 present the NRC staff's calculations and DOE waste classification results for (1) an example tank (Tank 32) and (2) for HTF transfer lines, respectively. DOE's waste classification results indicate that HTF components will not be greater than Class C at the time of closure (DOE/SRS-WD-2013-001, Rev. 0).

LLW intended for near-surface disposal is classified as Class A, B, or C based on concentration limits provided in 10 CFR 61.55. The 10 CFR Part 61 waste classification system was designed to protect an inadvertent intruder. To this end, additional requirements are specified in 10 CFR Part 61 for Class B and C wastes that pose a greater potential risk to an inadvertent intruder. To determine if HTF waste is Class C or less, DOE must compare concentrations of HTF components to the limits provided in 10 CFR 61.55. The concentration limits in 10 CFR 61.55, Table 1, for long-lived radionuclides include limits for a general class of radionuclides (alpha-emitting transuranic radionuclides with half-lives greater than five years) that DOE must identify.

Table 4-1 Sum of Fractions for Tank 32 (Using Site-Specific Factors Developed Based on Tank 13)

Radio-nuclide	Inventory (Ci)*	Concentration (Ci/m^3)*,†	Site-Specific Factor	Adjusted Concentration (Ci/m^3)*	Class C Limit (Ci/m^3)*	Fraction of Limit
Radionuclides Listed in Table 1 of 10 CFR 61.55						
C-14	$1.0\times10^{+00}$	6.3×10^{-02}	3.0×10^{-02}	1.9×10^{-03}	$8.0\times10^{+00}$	2.5×10^{-04}
Ni-59	$1.0\times10^{+00}$	6.3×10^{-02}	1.0×10^{-01}	6.3×10^{-03}	$2.2\times10^{+02}$	3.0×10^{-05}
Nb-94	1.1×10^{-01}	6.9×10^{-03}	8.4×10^{-03}	5.8×10^{-05}	2.0×10^{-01}	3.1×10^{-04}
Tc-99	$9.7\times10^{+00}$	6.1×10^{-01}	5.8×10^{-01}	3.5×10^{-01}	$3.0\times10^{+00}$	1.2×10^{-01}
I-129	6.7×10^{-03}	4.2×10^{-04}	$7.0\times10^{+00}$	2.9×10^{-03}	8.0×10^{-02}	3.9×10^{-02}
Np-237	4.0×10^{-01}	$1.3\times10^{+01}$	9.3×10^{-03}	‡1.2×10^{-01}	‡$1.0\times10^{+02}$	1.2×10^{-03}
Pu-238	$1.5\times10^{+04}$	$5.0\times10^{+05}$	9.5×10^{-06}	‡$4.8\times10^{+00}$	‡$1.0\times10^{+02}$	4.8×10^{-02}
Pu-239	$2.4\times10^{+02}$	$8.0\times10^{+03}$	2.2×10^{-03}	‡$1.8\times10^{+01}$	‡$1.0\times10^{+02}$	1.8×10^{-01}
Pu-240	$1.5\times10^{+02}$	$5.0\times10^{+03}$	9.2×10^{-04}	‡$4.6\times10^{+00}$	‡$1.0\times10^{+02}$	4.6×10^{-02}
Pu-241	$4.6\times10^{+03}$	$1.5\times10^{+05}$	4.9×10^{-04}	‡$7.5\times10^{+01}$	‡$3.5\times10^{+03}$	2.2×10^{-02}
Pu-242	$1.0\times10^{+00}$	$3.3\times10^{+01}$	1.3×10^{-02}	‡4.3×10^{-01}	‡$1.0\times10^{+02}$	4.4×10^{-03}
Pu-244	$1.0\times10^{+00}$	$3.3\times10^{+01}$	1.3×10^{-02}	‡4.3×10^{-01}	‡$1.0\times10^{+02}$	4.4×10^{-03}
Am-241	$1.1\times10^{+03}$	$3.7\times10^{+04}$	4.5×10^{-04}	‡$1.7\times10^{+01}$	‡$1.0\times10^{+02}$	1.7×10^{-01}
Am-242m	$1.0\times10^{+00}$	$3.3\times10^{+01}$	2.0×10^{-03}	‡6.7×10^{-02}	‡$1.0\times10^{+02}$	6.7×10^{-04}
Am-243	$1.0\times10^{+00}$	$3.3\times10^{+01}$	2.5×10^{-02}	‡8.3×10^{-01}	‡$1.0\times10^{+02}$	8.4×10^{-03}
Cm-243	$1.0\times10^{+00}$	$3.3\times10^{+01}$	1.4×10^{-06}	‡4.7×10^{-05}	‡$1.0\times10^{+02}$	4.7×10^{-07}
Cm-244	$2.2\times10^{+03}$	$7.3\times10^{+04}$	4.8×10^{-06}	‡3.5×10^{-01}	‡$1.0\times10^{+02}$	3.5×10^{-03}
Cm-245	$1.0\times10^{+00}$	$3.3\times10^{+01}$	1.8×10^{-02}	‡6.0×10^{-01}	‡$1.0\times10^{+02}$	6.0×10^{-03}
Cm-247	$1.0\times10^{+00}$	$3.3\times10^{+01}$	3.1×10^{-02}	‡$1.0\times10^{+00}$	‡$1.0\times10^{+02}$	1.0×10^{-02}
Cm-248	$1.0\times10^{+00}$	$3.3\times10^{+01}$	2.5×10^{-01}	‡$8.3\times10^{+00}$	‡$1.0\times10^{+02}$	8.4×10^{-02}
Cf-249	$1.0\times10^{+00}$	$3.3\times10^{+01}$	3.9×10^{-04}	‡1.3×10^{-02}	‡$1.0\times10^{+02}$	1.3×10^{-04}
Cf-251	$1.0\times10^{+00}$	$3.3\times10^{+01}$	6.1×10^{-04}	‡2.0×10^{-02}	‡$1.0\times10^{+02}$	2.0×10^{-04}
					Sum of Fractions:	**7.4×10^{-01}**
Radionuclides Listed in Table 2 of 10 CFR 61.55						
H-3	$1.0\times10^{+00}$	6.3×10^{-02}	NA	NA	NA	NA
Co-60	$1.0\times10^{+00}$	6.3×10^{-02}	NA	NA	NA	NA
Ni-63	$7.9\times10^{+02}$	$5.0\times10^{+01}$	1.2×10^{-03}	6.0×10^{-02}	$7.0\times10^{+02}$	8.9×10^{-05}
Sr-90	$2.0\times10^{+04}$	$1.3\times10^{+03}$	3.1×10^{-03}	$3.9\times10^{+00}$	$7.0\times10^{+03}$	5.9×10^{-04}
Cs-137	$5.5\times10^{+03}$	$3.5\times10^{+02}$	8.4×10^{-04}	2.9×10^{-01}	$4.6\times10^{+03}$	6.6×10^{-05}
					Sum of Fractions:	**7.4×10^{-04}**

* To convert from Ci (or Ci/m^3) to Bq (or Bq/m^3), multiply by $3.7\times10^{+10}$.
† The volume of waste assumed is assumed to be 15 m^3 and the weight of the waste is assumed to be $3\times10^{+04}$ kg.
‡ Concentrations are expressed in nCi/g. To convert from nCi/g to Bq/g, multiply by 37.

Table 4-2 Sum of Fractions Calculation for 3-in (8 cm) Transfer Line

Radio-nuclide	Inventory (Ci)*	Concentration (Ci/m^3)*,†	Site-Specific Factor	Adjusted Concentration (Ci/m^3)*	Class C Limit (Ci/m^3)*	Fraction of Limit
Radionuclides Listed in Table 1 of 10 CFR 61.55						
C-14	1.8×10^{-09}	3.3×10^{-06}	$1.6\times10^{+03}$	5.3×10^{-03}	$8.0\times10^{+00}$	6.6×10^{-04}
Ni-59	1.7×10^{-06}	3.1×10^{-03}	$2.0\times10^{+02}$	6.2×10^{-01}	$2.2\times10^{+02}$	2.8×10^{-03}
Nb-94	3.9×10^{-10}	7.1×10^{-07}	$1.7\times10^{+02}$	1.2×10^{-04}	2.0×10^{-01}	6.1×10^{-04}
Tc-99	1.7×10^{-05}	3.1×10^{-02}	$2.1\times10^{+00}$	6.5×10^{-02}	$3.0\times10^{+00}$	2.2×10^{-02}
I-129	1.8×10^{-10}	3.3×10^{-07}	$8.7\times10^{+03}$	2.9×10^{-03}	8.0×10^{-02}	3.6×10^{-02}
Np-237	7.5×10^{-08}	1.8×10^{-02}	$2.0\times10^{+01}$	‡3.5×10^{-01}	‡$1.0\times10^{+02}$	3.5×10^{-03}
Pu-238	1.1×10^{-03}	$2.6\times10^{+02}$	3.7×10^{-04}	‡9.6×10^{-02}	‡$1.0\times10^{+02}$	9.5×10^{-04}
Pu-239	1.8×10^{-05}	$4.3\times10^{+00}$	$1.1\times10^{+00}$	‡$4.7\times10^{+00}$	‡$1.0\times10^{+02}$	4.6×10^{-02}
Pu-240	1.1×10^{-05}	$2.6\times10^{+00}$	5.3×10^{-01}	‡$1.4\times10^{+00}$	‡$1.0\times10^{+02}$	1.4×10^{-02}
Pu-241	2.2×10^{-04}	$5.2\times10^{+01}$	4.5×10^{-02}	‡$2.3\times10^{+00}$	‡$3.5\times10^{+03}$	6.6×10^{-04}
Pu-242	3.1×10^{-08}	7.3×10^{-03}	$1.6\times10^{+02}$	‡$1.2\times10^{+00}$	‡$1.0\times10^{+02}$	1.2×10^{-02}
Pu-244	1.4×10^{-10}	3.3×10^{-05}	$3.7\times10^{+04}$	‡$1.2\times10^{+00}$	‡$1.0\times10^{+02}$	1.2×10^{-02}
Am-241	1.4×10^{-04}	$3.3\times10^{+01}$	5.3×10^{-02}	‡$1.8\times10^{+00}$	‡$1.0\times10^{+02}$	1.7×10^{-02}
Am-242m	1.0×10^{-07}	2.4×10^{-02}	1.5×10^{-02}	‡3.5×10^{-04}	‡$1.0\times10^{+02}$	3.5×10^{-06}
Am-243	2.2×10^{-06}	5.2×10^{-01}	$1.4\times10^{+01}$	‡$7.3\times10^{+00}$	‡$1.0\times10^{+02}$	7.2×10^{-02}
Cm-243	5.3×10^{-08}	1.3×10^{-02}	6.4×10^{-05}	‡8.0×10^{-07}	‡$1.0\times10^{+02}$	7.9×10^{-09}
Cm-244	1.8×10^{-05}	$4.3\times10^{+00}$	9.4×10^{-06}	‡4.0×10^{-05}	‡$1.0\times10^{+02}$	4.0×10^{-07}
Cm-245	7.3×10^{-09}	1.7×10^{-03}	$1.0\times10^{+03}$	‡$1.7\times10^{+00}$	‡$1.0\times10^{+02}$	1.7×10^{-02}
Cm-247	1.7×10^{-17}	4.0×10^{-12}	$7.3\times10^{+11}$	‡$2.9\times10^{+00}$	‡$1.0\times10^{+02}$	2.9×10^{-02}
Cm-248	1.8×10^{-17}	4.3×10^{-12}	$5.7\times10^{+12}$	‡$2.4\times10^{+01}$	‡$1.0\times10^{+02}$	2.4×10^{-01}
Cf-249	9.5×10^{-17}	2.2×10^{-11}	$3.9\times10^{+01}$	‡8.8×10^{-10}	‡$1.0\times10^{+02}$	8.7×10^{-12}
Cf-251	3.3×10^{-18}	7.8×10^{-13}	$4.7\times10^{+08}$	‡3.7×10^{-04}	‡$1.0\times10^{+02}$	3.6×10^{-06}
					Sum of Fractions:	**5.3×10^{-01}**
Radionuclides Listed in Table 2 of 10 CFR 61.55						
H-3	5.3×10^{-07}		NA	NA	NA	NA
Co-60	3.7×10^{-06}		NA	NA	NA	NA
Ni-63	1.2×10^{-04}	2.7×10^{-01}	1.5×10^{-02}	4.1×10^{-03}	$7.0\times10^{+02}$	4.7×10^{-06}
Sr-90	2.7×10^{-02}	$6.1\times10^{+01}$	$1.5\times10^{+01}$	$9.2\times10^{+02}$	$7.0\times10^{+03}$	1.1×10^{-01}
Cs-137	6.1×10^{-03}	$1.34\times10^{+01}$	$4.3\times10^{+00}$	$6.0\times10^{+01}$	$4.6\times10^{+03}$	1.0×10^{-02}
					Sum of Fractions:	**1.2×10^{-01}**

* To convert from Ci (or Ci/m^3) to Bq (or Bq/m^3), multiply by $3.7\times10^{+10}$.

† The volume of waste is assumed to be 4×10^{-04} m^3 and the weight of the waste is assumed to be $3.4\times10^{+03}$ g for 1 linear foot of piping.

‡ Concentrations are expressed in nCi/g. To convert from nCi/g to Bq/g, multiply by 37.

DOE lists several radionuclides that are members of this class (i.e., Np-237, Pu-238, Pu-239, Pu-240, Pu-242, Pu-244, Am-241, Am-243, Cm-243, Cm-244, Cm-245, Cm-247, Cm-248, Cf-249, and Cf-251). Table 2 of 10 CFR 61.55 lists another class of radionuclides, namely, short-lived radionuclides that have half-lives less than five 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 radionuclides with half-lives less than five years in Class B or C wastes and that the waste shall be Class B (or less) unless the concentrations of other radionuclides dictate that the waste is Class C or greater, independent of these radionuclides. Therefore, no radionuclides are evaluated in the class of short-lived radionuclides with half-lives less than five years. The HTF waste addressed in this TER contains a mixture of long- and short-lived radionuclides; therefore, 10 CFR 61.55 (a)(5) is also applied to determine waste classification. Because a mixture of radionuclides comprise the residual waste at HTF, a sum of fractions approach is used to determine the waste classification as provided in 10 CFR 61.55(a)(7).

The NRC's guidance on waste classification found in Section 3.5 of NUREG-1854 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 the waste classification for HTF components. The 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 protect an 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 analysis for potential LLW disposal facilities, the NRC staff developed Category 3 in NUREG-1854 to provide some flexibility for determining the waste classification for non-HLW WDs. The NRC staff recognized site- and problem-specific characteristics for tanks and auxiliary components at facilities handling waste that is incidental to reprocessing 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 steel tanks and grouted waste forms), deeper depths of disposal for tank waste residuals, and potential differences in residual waste geometries and volumes.

The 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 allow consideration of changes in internal dose methods and uncertainty in the calculations. However, while risk information such as actual depth of disposal and the volume of waste can be incorporated in the waste classification calculations under Category 3, NRC guidance also indicates that approaches should be conservative (i.e., calculations for waste classification should be at least as restrictive as the PA calculations). The NRC's guidance also indicates that application of a factor increase to the Class C limits based on considerations such as disposal depth might not be appropriate. The waste classification limits in 10 CFR Part 61 already consider some degree of mixing of higher-concentration waste with lower-concentration waste. A simple correction to the concentration limits in 10 CFR 61.55 to account for dilution of waste in clean soil would, therefore, be inappropriate because some dilution of waste in cleaner, relatively uncontaminated materials was already considered when the waste classification tables were developed. Although accessibility of the waste (e.g., depth and intruder barriers) is considered for concentration averaging applied to incidental waste, the NRC's guidance also

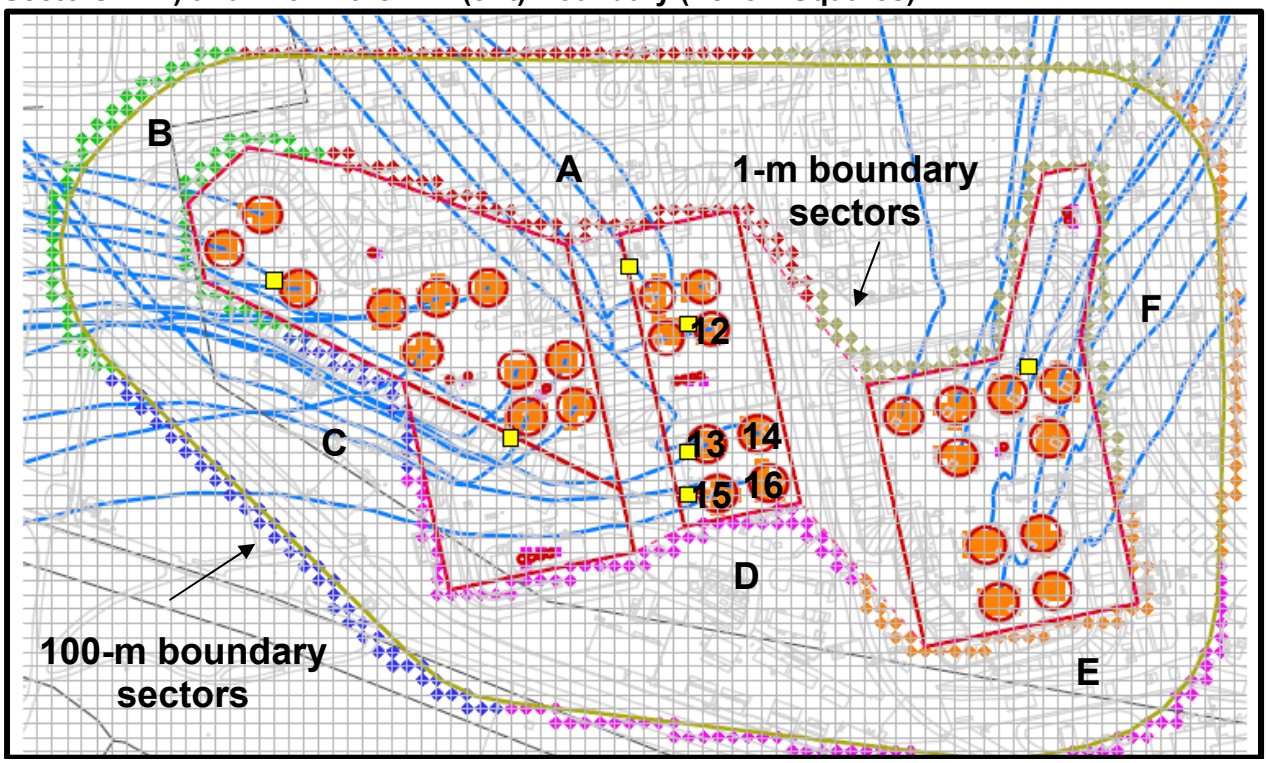
indicates that managing of intruder risk with complex, engineered facilities to prohibit access to the waste for very long periods of time (more than 500 years) is likely impractical. Thus, there are practical limits as to the types of waste that might be deemed suitable for near-surface disposal.

As stated above, DOE uses an approach consistent with Category 3 of the NRC's concentration averaging guidance described in Section 3.5 and Appendix B of NUREG-1854. Using this approach, DOE develops site-specific averaging expressions for HTF, based on the results of the HTF PA inadvertent intruder analyses. These site-specific averaging expressions include a site-specific factor that accounts for the unique conditions at HTF that differ from the assumptions inherent to the generic 10 CFR Part 61 waste classification limits. For example, due to the depth of residual waste assumed to be present in HLW tanks and auxiliary components (i.e., greater than 3 m or 10 ft below the closure cap), DOE determines that an excavation scenario is not plausible for the HTF. Therefore, this scenario is not considered in developing the site-specific averaging expressions. Typically, when waste is located at depths greater than 3 m or 10 ft, a well drilling scenario is evaluated as a more plausible intrusion event. In the HTF PA, DOE assumes 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 and only evaluates intrusion into a waste tank in sensitivity analyses after 500 years. For the purpose of the waste classification calculations, DOE assumes that intrusion could occur into a waste tank at 500 years. Due to the greater vulnerability of the transfer lines as compared to the waste tanks, DOE assumes in the HTF PA that intrusion into the transfer lines could occur immediately following the 100-yr institutional control period. DOE considers this potential intrusion into a transfer line at 100 years a conservative assumption given the presence of robust intruder barriers and considering the small footprint of the transfer lines at HTF [transfer lines represent 4 percent of the footprint of HTF(Page 769; SRR-CWDA-2010-00128, Rev. 1)]. DOE assumes that intrusion could occur into the transfer lines at 100 years in the waste classification calculations, consistent with the HTF PA.

DOE calculates site-specific factors for the drill cutting source based on assumed concentrations at the expected year of closure (2032): (1) a 3-in (8-cm) diameter transfer line, (2) a 4-inch (10-cm) diameter transfer line, or (3) a waste tank. Ancillary equipment concentrations are assumed to be bounded by a waste tank. Site-specific factors are calculated for each radionuclide regardless of the time of the peak dose from that radionuclide.

With respect to the groundwater pathway, DOE develops site-specific factors based on Tank 13, because Tank 13 is found to be "a major contributor to the peak dose from the 1-m well in the HTF" (DOE/SRS-WD-2013-001, Rev. 0). However, it is unclear which 1-m well DOE is referring to. Furthermore, unless the 1-m well downgradient of Tank 13 observes the highest concentrations of key radionuclides contributing to the risk to the inadvertent intruder, it is not clear why DOE selected that 1-m well. In fact, insufficient information was provided to show that a 1-m well downgradient of Tank 13 would be the most limiting well for waste classification purposes. In addition, information provided in the HTF PA suggests that other tank sources may be more limiting when developing site-specific factors. For example, peak intruder doses are attributed to Sector D in the HTF PA (see SRR-CWDA-2010-00128, Rev. 1, page 768). Sector D wells are not located in particle tracks downgradient of Tank 13 (Figure 4-1 shows a particle track from Tank 13 intersecting Sector C at the 1-m boundary).

Figure 4-1 Intruder Well Locations at the 1-m (3-ft) Boundary (Inner Boundary Grouped in Sectors A–F) and Within the 1-m (3-ft) Boundary (Yellow Squares)



Adapted from Figure 6.5-5 in SRR-CWDA-2010-00128, Rev. 1.
Also illustrated are particle tracks (blue lines).

Furthermore, the intruder dose reported on Figure 6.5-6 of the HTF PA (Page 776; SRR-CWDA-2010-00128, Rev. 1) for a well located within the 1-m boundary directly downgradient of Tank 13 is not the highest "1-m Boundary" intruder dose reported on the figure. The peak intruder dose reported in the HTF PA is associated with a well located within the 1-m boundary adjacent to Tank 12 (see yellow square to the left of Tank 12 in Figure 4-1). The Tank 12 intruder well dose is an order of magnitude higher than the dose reported for Tank 13 (Page 776; SRR-CWDA-2010-00128, Rev. 1). The HTF PA intruder simulations show the sensitivity of the results to exposure point locations (see Section 4.2.16). Notwithstanding the intruder doses reported in the HTF PA, because final inventories for HTF tanks and other components have not yet been developed, the intruder well location that represents the most limiting case with respect to protection of the inadvertent intruder and consequently, the waste classification, is not yet known.

DOE averages the total inventory associated with the tanks, including the annulus and sand pads, in calculating the site-specific factors for the tank drill cuttings source. It appears that the same approach is taken when calculating the site-specific factors for the groundwater pathway (e.g., SRR-CWDA-2012-00109, Rev. 0 lists one set of site-specific factors that presumably combines the tank drilling source with the groundwater source). Because radionuclides in annular waste may pose a higher or lower risk compared to radionuclides residing in the tank (e.g., annular waste may be more soluble [Section 3.4.2.2; SRR-CWDA-2010-00128, Rev. 1] and released earlier in time compared to tank waste), DOE should develop separate site-specific factors for risk-significant annular waste versus tank waste sources in the future.

Annular and tank sources would then be separately compared to adjusted waste classification concentration limits to determine the classification of HTF components.

DOE assumes the waste has the same density as the grout it plans to use to stabilize the tanks and vaults. While the density of the tank waste is assumed to be similar to the density of the stabilizing (grout) materials in the concentration calculations, the waste is not expected to be well mixed with the tank grout and the actual density of the waste is expected to be significantly lower than the density of the grout based on the measured waste density following waste retrieval in FTF tanks (1.1-1.4 g/cm^3 for tank waste versus 2 g/cm^3 for grout). However, because the same density is used in the waste classification calculations and in the calculation of site-specific factors, it does not appear that the density of the waste impacts the waste classification calculations.

With the exceptions noted above, the NRC staff has evaluated DOE's methodology for classifying waste and generally finds the approach to be an acceptable application of the Category 3 concentration averaging guidance in Section 3.5.1.1 of NUREG-1854. Information provided for Tank 32 and the transfer lines in the draft basis for the WD for HTF (DOE/SRS-WD-2013-001, Rev. 0) indicates that the residual waste in these components does not exceed Class C concentration limits. Furthermore, DOE calculates site-specific factors for each waste classification radionuclide based on the peak dose for that radionuclide, independent of the timing of the overall peak dose for all radionuclides, thereby adding conservatism to the calculation.

Although DOE documents its expectation that all tanks and auxiliary equipment at HTF will meet Class C limits, no final inventories are available for HTF components at this time. During an FTF RAI resolution meeting on January 20, 2011 (Shaffner, 2011a [ML110250118]), DOE indicated its intent to confirm the waste classification at the time of individual component closure for FTF, at which time the final inventory of the components will be known. The NRC staff expects that DOE will follow the same approach for HTF. Section 3116(a)(3)(B)(iii) requires DOE to consult with the NRC in its development of disposal plans if the waste exceeds Class C concentration limits. In the draft basis document for the WD for HTF (DOE/SRS-WD-2013-001, Rev. 0), DOE stated its intent to "…take full advantage of the consultation process established by NDAA Section 3116," and asked the NRC to "…identify what changes, if any, NRC would recommend to DOE's disposal plans," as described in the draft basis document. The NRC staff has reviewed DOE's preliminary disposal plans for the HTF waste as part of the consultation process that is documented in this TER, thereby satisfying the requirements of Section 3116(a)(3)(B)(iii) for tanks containing residual waste that might exceed Class C concentrations at the time of preparation of this TER. Should DOE find that HTF waste exceeds Class C limits based on final inventories developed in the future, the NRC staff could provide further recommendations related to DOE's final disposal plans, if requested by DOE.

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 compliance with the pertinent requirements, which include the NDAA criteria, for waste tank operational closure and eventual final facility closure of the HTF. Performance assessment components include: i) the evaluation of potential initiating events (both natural and anthropogenic) that can cause releases of radioactive material; ii) estimates of the release rates of radionuclides into the accessible environment; iii) modeling the fate and transport of radionuclides; and iv) evaluation of the

potential exposure pathways and associated risks (i.e., doses) to human health. The PAs submitted by DOE to support non-HLW WDs include a collection of loosely integrated process models to demonstrate compliance with performance objectives in 10 CFR Part 61, Subpart C.

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 PA results. Because of the long time periods analyzed, which can span thousands of years, PAs for the disposal of long-lived radionuclides cannot be validated in a traditional sense. However, the decisionmakers can rely on the results of laboratory and field experiments, monitoring data, natural analogs, peer reviews, expert elicitation, and supporting models to provide multiple lines of evidence, which develop increased confidence in the PA results. Analysts developing a PA should not limit model support to information that affirms the conceptual model implemented in the PA, but should also include information that challenges the conceptual modeling approaches for the consideration of decisionmakers. The amount of model support provided should be commensurate with the importance of the barrier or process being simulated for the compliance demonstration.

4.2.1 Summary of Performance Objectives

The NDAA establishes the applicable criteria for determining that waste is not HLW, and includes a requirement that the waste will be disposed of in compliance with the performance objectives in 10 CFR Part 61, Subpart C. The performance objectives provide criteria to ensure that the public, workers, and the environment will be protected from exposures to radioactivity from the disposed waste, as well as specifying site-stability requirements.

The regulations at 10 CFR 61.41 state 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)

A 0.25 mSv/yr (25 mrem/yr) total effective dose equivalent (TEDE) limit applies for the post-closure period of a disposal facility (NUREG-1854). The other radiological control limits of 10 CFR Part 20 (NRC, 2001b), apply during facility operation.

The regulations at 10 CFR 61.42 state 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." (NRC, 2001a)

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 for 10 CFR Part 61

(NUREG-0782), the NRC used a 5-mSv/yr (500-mrem/yr) dose limit to an 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" (NRC, 2001a).

The regulations at 10 CFR 61.43,"Protection of individuals during operations," which applies to both the public and disposal facility workers, 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." (NRC, 2001a)

The regulations at 10 CFR 61.44, states the following:

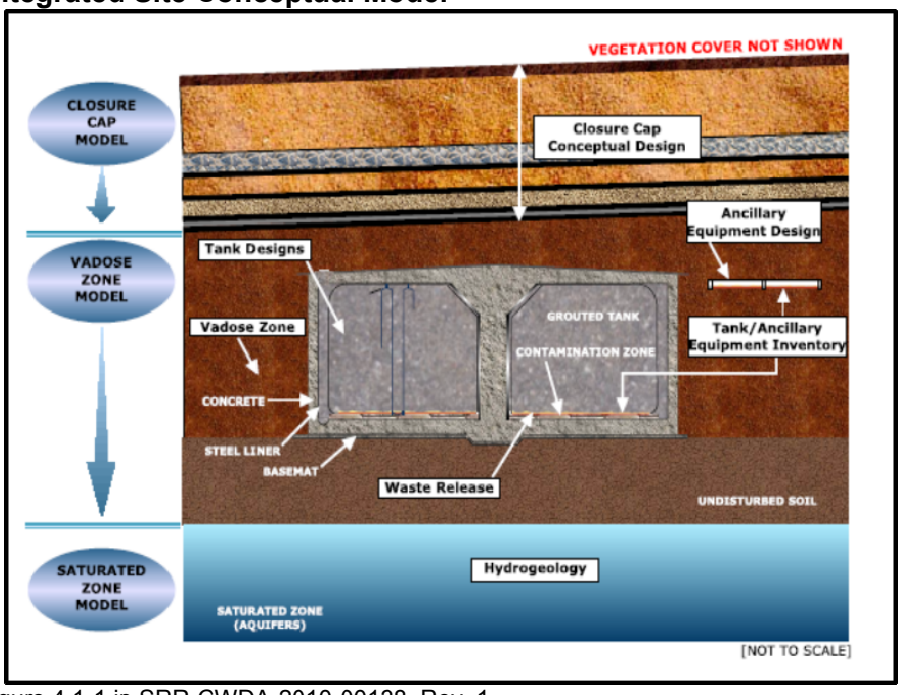
> "The disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for ongoing active maintenance of the disposal site following closure so that only surveillance, monitoring, or minor custodial care are required." (NRC, 2001a)

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" (NRC, 2001a).

4.2.2 Summary of Performance Assessment Approach and Results

DOE developed a PA for the HTF (SRR-CWDA-2010-00128, Rev. 1) to estimate the release of radiological constituents from 29 waste tanks and associated ancillary equipment. DOE evaluates the releases of radiological constituents using both deterministic and probabilistic analyses. The releases and resulting radiological exposures to members of the public from a deterministic analysis are estimated for comparison to the 10 CFR Part 61, Subpart C, performance objectives. DOE uses both deterministic and probabilistic analyses to assess the uncertainty in the demonstration of compliance with the 10 CFR Part 61, Subpart C performance objectives. To evaluate the long-term ability of the HTF waste tanks and ancillary equipment to limit radiological exposures to members of the public, DOE constructed an integrated conceptual model (ICM) with three main components: an engineered closure cap conceptual model, a vadose zone conceptual model, and a saturated zone conceptual model (see Figure 4-2). DOE states that a comprehensive list of features, events, and processes (FEPs) were considered to develop the ICM. DOE prepared two reports that contain an *ex post facto* analysis of FEPs that are considered for the liquid waste PAs at SRS and an evaluation of the adequacy of the disposition of included FEPs in the HTF PA (SRR-CWDA-2012-00011, Rev. 0 and SRR-CWDA-2012-00044, Rev. 1, respectively).

Figure 4-2 Integrated Site Conceptual Model



Adapted from Figure 4.1-1 in SRR-CWDA-2010-00128, Rev. 1.

In the ICM, the closure cap is considered a surface feature and the region beneath the base of the closure cap is considered the vadose zone. The majority of the waste sources (i.e., 21 tanks and ancillary equipment) are located within the vadose zone, along with undisturbed soil and engineered backfill. In addition, eight waste tanks (all Type I and II tanks), along with some ancillary equipment, are either fully or partially submerged in the saturated zone. DOE considered five alternative scenarios in the HTF PA, termed Cases A through E, to evaluate potential system behavior for the waste tanks, along with a single configuration for the ancillary equipment. Section 4.4.2 of SRS-CWDA-2010-00128, Rev. 1 describes the alternative waste tank Cases A–E and the configuration of the ancilliary equipment. 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 alternative configurations should not be interpreted as representing a specific mechanism for liner/grout/concrete degradation or understood to mean that a given condition or set of conditions would actually be present in a closed waste tank system at some point in the future. DOE considers Case A, the base case, to be the best estimate of the behavior of the closed HTF 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 (Schroeder et al., 1994a; Schroeder et al., 1994b). 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 evaluates several release scenarios of potential importance. Section 4.2.4 of this TER summarizes DOE modeling of the closure cap and Section 4.2.18 discusses alternative cases considered by DOE, including a case without a closure cap.

In the vadose zone modeling, DOE estimates degradation rates and failure times for cementitious materials and steel liners for each of the five tank types (i.e., Types I, II, III, IIIA, and IV) that comprise the HTF. The estimated behavior of the cementitious materials and steel liners affects the chemical environment in the tanks. The behavior of the cementitious materials and steel liners and the chemical environment also affect the release of radionuclides from the contaminated zone (CZ)[7], which lies within the primary liner of each tank, and from annular regions, which are outside the primary liners for tanks that have leaked HLW from the primary liner. All eight Type I and II tanks are modeled with contamination in the annulus between the primary and secondary liners[8]. Radionuclides released from the CZ or annulus are then available for transport out of the waste tank and into the vadose zone surrounding the tank. 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 HTF PA.

DOE deterministically models the vadose zone domain for each HTF waste tank type individually within PORFLOW™ (ACRi, 2008) to determine the tank-specific, time-dependent radionuclide flux entering the water table or exiting the bottom of the tank basemat for fully and partially submerged tanks. PORFLOW™[,9] is also used to estimate average velocity profiles of water infiltrating through the HTF vadose zone for (1) the base case (Case A), (2) all alternative configurations (Cases B, C, D, and E), and (3) tank types simulated in the probabilistic analysis using the GoldSim™ modeling platform[10]. DOE also uses GoldSim™ (GoldSim Technology Group LLC, 2010) to evaluate radionuclide release from the waste tanks in the probabilistic analysis. The transfer line source term is modeled by distributing the assumed inventory equally over four zones of the HTF area around the waste tanks and transfer facilities. Nine pump tanks, two condensate transfer system tanks, and three evaporator pots are modeled as individual point sources. Figure 4-3 depicts the modeled sources, both discrete (as in waste tanks and ancillary equipment) and distributed (as in transfer lines) . 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.

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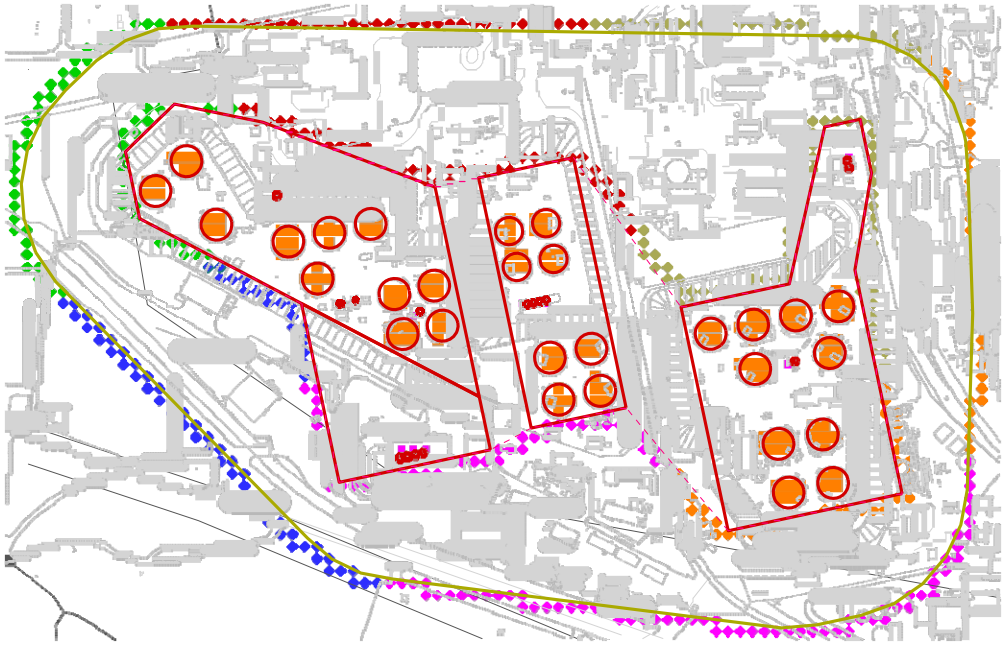
[7] The overlying tank fill grout conditions the chemistry of the water contacting the CZs in Cases A, B, and D. However, in Cases C and E, the chemistry of the water contacting the waste in the CZ is not conditioned by the overlying tank fill grout.

[8] Type I Tanks 9-12 and Type II Tanks 13-16 are modeled with contamination initially present in the annular region between the primary and secondary liners. Type I tanks are modeled with contamination in the annulus between the primary and secondary liners, while Type II tanks are modeled with contamination in the primary sand pad that is located between the primary and secondary liners. Because of its history of overtopping the secondary liner, Tank 16 is also modeled with contamination in the secondary sand pad which is located beneath the secondary liner (see Figures 1-1 and 1-2 for a depiction of the location of the annulus in Type I tanks and primary and secondary sand pads for Type II tanks.)

[9] PORFLOW is a registered trademark of Analytic & Computational Research, Inc.

[10] GoldSim is a registered trademark of GoldSim Technology Group LLC.

Figure 4-3 HTF/PORFLOW™ Saturated Zone Model Domain Showing Both Discrete and Distributed Sources, Stream Traces from Waste Tank Sources, and 1-m (3-ft) and 100-m (330-ft) Compliance Evaluation Sectors (A–F)



Adapted from Figure RAI-NF-14.2 in SRR-CWDA-2013-00106, Rev. 1.

Modeled discrete source locations are shown as orange squares for waste tanks and smaller, magenta squares for ancillary structures. Distibruted sources are shown as large red polygons for the areal extent of the transfer lines in the model.

DOE inputs the releases from these scenarios into environmental transport and exposure pathway models to evaluate radiological exposures to members of the public. Groundwater is the primary exposure pathway DOE considers. DOE uses the PORFLOW™ code 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 far-field contaminant flow and transport in the saturated zone.

Concentrations of radionuclides in groundwater at the compliance points are used to calculate doses to receptors through a number of residential and agricultural pathways Section 4.2.12 summarizes DOE's dose methodology. 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 4.2.14 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 estimate the potential dose from releases of radioactivity into the environment to demonstrate compliance with 10 CFR 61.41. PORFLOW™ is used in the HTF PA to calculate vapor phase radionuclide diffusion to the ground surface for use in air transport calculations. DOE includes nine radionuclides that were selected from a screening analysis. The Clean Air Act Assessment Package—1988 (CAP-88) computer model (Beres, 1990) is used to estimate dose from radionuclide emissions

to air. CAP-88 was used in the FTF PA to estimate the annual dose to maximally exposed individuals considering plume and ground shine (i.e., gamma radiation), inhalation, and foodstuff ingestion pathways using the results of the vapor phase radionuclide diffusion to the surface from PORFLOW™, as discussed above.

DOE employs 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 including 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 and to assess the sensitivity of model results to input parameters. DOE used the GoldSim™ code, with water flow input from a parametric flow study conducted in 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 Performance Assessment Approach

The NRC staff evaluated DOE's scenario analysis approach including identification, screening, and implementation of FEPs into the HTF PA. The details of the NRC staff's evaluation of DOE's identification, screening, and implementation of FEPs are documented in Section 4.2.3.1. The following sections first present a summary of DOE's implementation of conceptual models and parameter justifications followed by the details of the NRC staff's evaluation of DOE's approach for the following models:

- The closure cap model (Sections 4.2.4 and 4.2.5 of this TER),
- The radionuclide inventory (Sections 4.2.6 and 4.2.7 of this TER),
- The vadose zone model (Sections 4.2.8 and 4.2.9 of this TER), and
- The saturated zone model (Section 4.2.10 and 4.2.11 of this TER).

The NRC staff also reviewed DOE's selection of compliance boundaries. The NRC staff finds that DOE's selection of the compliance boundaries includes several sources that are expected to be of lesser risk significance (e.g., ancillary equipment) that, as a result of their inclusion, extend the boundary, as well as the travel time for radionuclides from some of the tanks, which are generally considered more risk significant. The NRC staff's evaluation of DOE's selection of compliance boundaries is discussed further in Section 4.2.11 of this TER.

The NRC staff also reviewed DOE's dose modeling methodology and finds that (1) the HTF PA presents a reasonable development of exposure pathways important for receptors located at SRS, and (2) the receptor characteristics and exposure scenarios are generally reasonable, as described in Sections 4.2.12 and 4.2.13. The NRC staff reviewed DOE's application of dose limits for demonstrating that the performance objectives for protection of the general population and individuals from inadvertent intrusion are met and finds that the dose limits are appropriately applied. Sections 4.2.14 and 4.2.15 discuss the application of dose limits for protection of the general population and Sections 4.2.16 and 4.2.17 disuss the application of dose limits for protection of individuals from inadvertent intrusion.

DOE evaluated a range of alternative conceptual models, performed uncertainty and sensitivity analyses (see Sections 4.2.18 and 4.2.19), or used other means to assess the uncertainty in the demonstration of compliance with the performance objectives of 10 CFR Part 61, Subpart C.

Table 4-3 provides an overview of DOE's HTF PA results for comparison with the performance objectives of 10 CFR Part 61, Subpart C. Provided all of DOE's key modeling assumptions are met, the results of DOE's HTF PA indicate that the performance objectives will most likely be met during a 10,000-yr compliance period.

For the protection of the general population performance objective, HTF PA results indicate that doses would exceed 0.25 mSv (25 mrem) TEDE at some point in the future from Ra-226. Given the delayed ingrowth of Ra-226 from its parent radionuclides, Pu-238, U-234, and Th-230, it is likely that the peak dose from Ra-226 would occur after 10,000 years. On the other hand, the uncertainty and sensitivity analyses show the potential for other radionuclides to dominate the peak dose earlier in time and at levels potentially above the dose limit should assumptions and approaches relied on for Case A prove to be invalid.

For protection of individuals from inadvertent intruders, results from the HTF PA probabilistic analysis indicate that projected doses could exceed 500 mrem/yr within 10,000 years. Because DOE's intruder analysis also evaluates potential doses to an inadvertent intruder at a well located at the 1-m (3-ft) compliance boundary, and at seven wells adjacent to specific waste tanks, concerns related to the 10 CFR 61.41 compliance demonstration also hold true for the 10 CFR 61.42 analyses. Due to the 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 analyses.

4.2.3.1 Scenario Analysis

Scenario analysis involves the formation of potential scenarios for demonstrating compliance with the performance objectives. The NRC staff focused its review of DOE's scenario analysis on the identification, screening, and implementation of FEPs for the HTF PA.

Identification of FEPs

The NRC staff evaluated DOE's methodology for identification of FEPs, screening of FEPs for further consideration, and implementation of FEPs in the HTF PA using Section 4.1.1.3 of NUREG-1854 and Section 3.2.1 of NUREG-1573. The NRC staff initially identified four FEPs related to climate change that are identified by the International Atomic Energy Agency (IAEA, 2004) that DOE has not appeared to consider:

- Hydrological/hydrogeochemical response to climate changes
- Ecological response to climate changes
- Human response to climate changes
- Other geomorphological changes

However, in response to an RAI from the NRC staff (CC-PA-1; Mohseni, 2013a [ML13196A135]) regarding FEP screening, which is discussed further below, DOE's response (SRR-CWDA-2013-00106, Rev. 1) provides a Microsoft® Excel spreadsheet, entitled "SRS_LW_FEPs_Rev0.xlsx" which identifies the four FEPs from IAEA (2004) and appears to roll them into DOE's FEP 2.7.07, "Climate Change" although this consolidation is not transparent to the NRC staff in SRR-CWDA-2012-00011, Rev. 0. Therefore, the NRC staff's review of DOE's identification of FEPs finds that DOE's identification is adequate.

Table 4-3 Summary of HTF Performance Assessment Results (Values in Red Text are Greater Than 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† can be met within 10,000 years compliance period.	PA results show doses are expected to be significantly above performance objective at some uncertain time in the future.	Performance objective‡ can be met.	Robust compliance conclusion for drill cuttings. Uncertainty with dose from intruder well.
Deterministic Analyses	Case A§ 4 mrem/yr 124 mrem/yr	Peak dose at 100-m boundary line within: 10,000 yrs 100,000 yrs	Chronic Exposure 51 mrem/yr 74 mrem/yr 100 mrem/yr 260 mrem/yr	Peak dose at 1-m boundary line within 10,000 years where intruder assumed to drill through: 3-in transfer line§ 4-in transfer line Waste tank 13,¶ Peak dose at 1-m boundary line within 20,000 yrs where intruder assumed to drill through a 3-in transfer line.
	Case B 14 mrem/yr 125 mrem/yr	Peak dose at 100-m boundary line within: 10,000 years 100,000 years		
	Case C 16 mrem/yr 124 mrem/yr	Peak dose at 100-m boundary line within: 10,000 years 100,000 years		
	Case D 18 mrem/yr 123 mrem/yr	Peak dose at 100-m boundary line within: 10,000 years 100,000 years		
	Case E 239 mrem/yr 239 mrem/yr	Peak dose at 100-m boundary line within: 10,000 years 100,000 years	Acute Exposure 1.4 mrem/yr 2.6 mrem/yr 14 mrem/yr	Peak dose at 1-m boundary within 10,000 yrs where inturder assumed to drill through: 3-in transfer line§ 4-in transfer line Waste tank 13

Table 4-3 (continued) Summary of HTF Performance Assessment Results (Values in Red Text are Greater Than the Compliance Limit)*

	Protection of the General Population 10 CFR 61.41	Notes	Protection of Individuals from Inadvertent Intrusion 10 CFR 61.42	Notes
Probabilistic Analyses§,□	Case A 13 mrem/yr 85 mrem/yr	Within 10,000 yrs Peak of mean doses Mean of peak doses	Chronic Exposure 762 mrem/yr 961 mrem/yr	Peak of the mean doses inside 1-m boundary at hypothetical drilling locations adjacent to specified waste tanks (adjacent to Tank 12): Within 10,000 years Within 20,000 years
	Case D 35 mrem/yr 210 mrem/yr	Within 10,000 yrs Peak of mean doses Mean of peak doses		
	All Cases 15 mrem/yr 220 mrem/yr 205 mrem/yr 530 mrem/yr	Within 10,000 yrs Peak of mean doses Mean of peak doses Within 100,000 yrs Peak of mean doses Mean of peak doses		

* To convert mrem/yr to mSv/yr, multiply by 0.01.
† Performance objective for protection of the general population from releases of radioactivity evaluated at 25 mrem/yr total effective dose equivalent.
‡ Performance objective for protection of individuals from inadvertent intrusion is evaluated at 500 mrem/yr total effective dose equivalent.
§ Case, scenario, or analysis evaluated for compliance with the respective performance objective.
□ Case, scenario, or analysis evaluated for understanding uncertainty in the demonstration of compliance with the respective performance objective.
¶ Scenario evaluated exposures from drill cuttings beginning at 500 years after closure.

Screening of FEPs

The NRC staff's review of DOE's screening methodology finds the methodology, which focuses on the perceived likelihood and impact as criteria for screening, is reasonable. The NRC staff also finds that DOE provides documentation of rationales that are relied upon for the screening decision in the case of excluded FEPs. However, the NRC staff notes that the justification provided for excluding a number of FEPs during the second phase (See Table 4.3-1; SRR-CWDA-2012-00011, Rev. 0) of screening appears to mix the likelihood and impact criteria employed by DOE. For instance, FEP 2.6.03, Orogeny, is screened out with the following justification: "This FEP is screened out based on the professional judgment of the FEPs Screening Team members that it would be beyond extremely unlikely for this to affect PA results." The NRC staff finds this justification confusing as to whether the FEP was perceived as unlikely to occur during the 10,000-year period, or unlikely to affect the results of the PA. In some cases, the meanings could be equivalent, but not necessarily so, and this may lead to confusion and a loss of transparency. The NRC staff recommends that DOE clarify for FEPs with the aforementioned justification whether the FEP is perceived to be unlikely during the assessment period or the impact is perceived to be insignificant according to DOE's decision criteria.

The NRC staff also reviewed the use of expert judgment to inform the screening of FEPs, including the qualifications of the DOE screening team and the methodology the team used, using guidance from NUREG-1563. The NRC staff reviewed the qualifications of the DOE team that performed the screening using their expert judgment. The team consisted of nine members. The qualifications are provided in Section 5.0 of SRR-CWDA-2012-00011, Rev. 0. The NRC staff notes that NUREG-1563 recommends that subject matter experts, even for informal professional judgments be knowledgeable individuals in engineering and science, who, by nature of their experience and academic achievement, can speak to the understanding of certain scientific laws and principles. The membership of DOE's screening team minimally meets this requirement, however, the NRC staff notes that the expertise appears to be focused on experience conducting PAs and appears to be somewhat limited in the areas of the geosciences, corrosion, and cementitious materials behavior. The NRC staff recommends that DOE include subject matter experts on the screening team in the specific engineering and scientific disciplines that are pertinent to the expert judgments being made. In some cases, subject matter experts may not be available due to the nature of the FEP or resource limitations. In these cases, DOE should document more clearly the specific expertise of the chosen experts and how their experience relates to the expert judgment being made. Regardless of whether appropriate subject matter experts are utilized, DOE should clearly document a transparent technical basis for each screening decision that also provides a traceable link to the information considered and relied upon to arrive at the screening decision.

NUREG-1563 also indicates that the NRC staff will accept for review the results of formal or informal judgment provided the rationale associated with the judgment is adequate, transparent, and sufficiently documented. The NRC staff notes that DOE did not include documentation of an individual subject matter expert's basis for judgment and the data relied upon for their judgment during the first phase of the screening, in which subject matter experts assigned values for perceived likelihood and impact for each FEP. Further, documentation in SRR-CWDA-2012-00011, Rev. 0 of the discussions from the second phase of screening and why subject matter experts may have changed their initial opinions on the criteria to arrive at a

consensus opinion were not provided. In response to an RAI from the NRC staff (CC-PA-2; Mohseni, 2013a [ML13196A135]), DOE provided in SRR-CWDA-2013-00106, Rev. 1 a Microsoft® Excel spreadsheet, entitled "SRS_LW_FEPs_Rev0.xlsx". However, DOE's response provided sparse additional information regarding the rationales for individual subject matter expert's judgments or on the data considered during the initial and second screening phases. The NRC staff recommends that DOE enhance the transparency and traceability of its expert judgment process to be consistent with the guidance provided in NUREG-1563 for informal expert judgment. Specifically, subject matter experts should provide reasoning for their opinions and the data, if any, from which those opinions are formed. Further, sufficient documentation should exist to allow external examination of what the judgments are, how the judgments are arrived at (their basis), and how the judgments are used.

Finally, the NRC staff has questions regarding the screening decisions for the following FEPs. The outstanding questions are identified for each FEP below:

- 2.7.04, Acid Rain—(Excluded) It is not clear to the NRC staff why this FEP was determined to be outside the scope of the HTF PA. The NRC staff believes that information on acid rain and its effects on the performance of the closed tanks could be considered in the PA should the anticipated impact be expected to be sufficiently significant.

- 2.2.07, Pollution—(Excluded) It is not clear to the NRC staff why this FEP was determined to be outside the scope of the HTF PA. The NRC staff believes that information on existing pollution and its effects on the performance of the closed tanks could be considered in the PA should the anticipated impact be expected to be sufficiently significant.

- 3.5.12, Chelating Agents Effects—(Excluded) It is not clear to the NRC staff how DOE considered the effects of chelating agents on contaminant mobility, such as oxalates that result from the use of oxalic acid during tank cleaning. Other examples of chelating agents may include humic or fulvic acids in SRS soils or agents used in the solvent extraction processes from H-canyon that remain in the waste received at HTF.

- 3.6.01, Thermal Processes and Conditions the Engineered System—(Excluded) It is not clear to the NRC staff how DOE considered the effects of heat of hydration and how changes in stresses resulting from the heat of hydration could change both hydrologic and mechanical properties of the engineered components.

- 3.6.04, Thermo-Mechanical Stresses Alter Characteristics of Engineered Barrier System Components—(Excluded) See discussion for FEP 3.6.01 above.

- 6.2.04, Seismicity Associated with Igneous Activity—(Excluded) It is not clear to the NRC staff whether the rationale for screening this FEP out is related to the subject of this particular FEP. The justification provided in Table 4.3-1 of SRR-CWDA-2012-00011, Rev. 0 for screening out this FEP indicates that the closure system does not have enough heat to significantly impact the results. This rationale appears unrelated to the subject of this particular FEP.

Implementation of FEPs

The NRC staff also reviewed DOE's evaluation of whether the HTF PA appropriately incorporates FEPs that passed the screening process and were to be included in the HTF PA, as documented in SRR-CWDA-2012-00044, Rev. 1. The report SRR-CWDA-2012-00044, Rev. 1 provides a crosswalk for each FEP to the relevant sections of the HTF PA or supporting references and a summary of DOE's findings of whether included FEPs were appropriately incorporated in the PA. The NRC staff's review finds that the crosswalk has the potential to enhance transparency and traceability. However, the NRC staff notes that the crosswalk is generally not comprehensive. Many of the selected FEPs are crosswalked to either an erroneous section of the HTF PA or a section that provides an example of where the particular FEP is implemented in the HTF PA, but is generally not comprehensive. In some cases, the crosswalk (1) doesnot link to every instance the FEP is implemented, or (2) links to an erroneous instance where the FEP is implemented into the HTF PA. In other cases, the level of detail that is provided by SRR-CWDA-2012-00044, Rev. 1 regarding the implementation of a FEP in the PA is inadequate.

For example, the NRC staff points to FEP 3.7.09, "Concrete Shrinkage/Expansion". The report SRR-CWDA-2012-00044, Rev. 1 crosswalks the implementation of this FEP to Section 4.2.2.2.4 of the Revision 0 HTF PA (SRR-CWDA-2010-00128, Rev. 0). SRR-CWDA-2012-00044, Rev. 1 (see Table 3.0-1) indicates that the "PA Model assumes concrete shrinkage will have a negligible effect on the grout and does not model this process." DOE appears to have confused a screening argument for a description of how this FEP, which is screened in (i.e., included) based on the screening methodology, is implemented in the HTF PA. The NRC staff finds that Revision 1 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1) provides no discussion in Section 4.2.2.2.4 of how shrinkage is implemented in the HTF PA, however, the NRC staff notes that shrinkage is discussed in other sections of the HTF PA (SRR-CWDA-2010-00128, Rev. 1), notably in regards to alternative Cases B-E, such as Sections 4.4.2.2 through 4.4.2.5. Therefore, it appears that DOE does not consider shrinkage of cementitious materials in its base case even though it had concluded that this FEP should be included (i.e., screened in this FEP). Other examples among the sampling of FEPs reviewed by the NRC staff that are reportedly screened in, but in fact not implemented in the HTF PA are: FEP 3.5.11, "Reaction Kinetics" and FEP 3.5.07, "Colloidal Generation".

Regarding the level of detail, the NRC staff points to FEP 3.5.09, "Rind (Chemically Altered Zone) Forms in the Near-Field" as an example. The NRC staff notes that this is a general FEP that could cover a potentially large range of chemical alteration processes, but DOE only identifies the leaching of cementitious materials into the soil and the effects this material has on the chemistry of the soil. DOE provides no documentation in SRR-CWDA-2012-00044, Rev. 1 regarding the implementation of other potential chemically altered zones in the HTF PA, such as weathering or coating of fracture surfaces. Other examples among the sampling of FEPs reviewed by the NRC staff that are not considered in sufficient detail are FEP 3.4.04, "Resaturation and Desaturation," which only considers the effects on closure cap and not on cementitious materials or steel liners, and FEP 3.5.01, "Chemical/Geochemical Processes and Conditions," which does not provide traceability to the various chemical/geochemical processes considered (e.g., oxalic acid impacts on waste release) in the HTF PA.

The result of these examples is a lack of transparency and traceability and reduced confidence that all relevant FEPs are included in the HTF PA. The NRC staff recommends that DOE improve the transparency and traceability of its implementation of FEPs as described in SRR-CWDA-2012-00044, Rev. 1 to ensure comprehensive, accurate, and traceable links to clear descriptions of how included FEPs are actually implemented in the HTF PA.

4.2.3.2 Barriers Important to the Compliance Demonstration

DOE's HTF PA models the performance of multiple barriers, both engineered and natural, that either reduce the movement of water or radionuclides through the environment or reduce the release rate of radionuclides from the tanks. Barriers that reduce the flow of water are the closure cap, the tank vault concrete, the carbon steel tank liner, and the tank fill grout. Barriers that limit the release or movement of radionuclides are the carbon steel tank liners, tank fill grout, basemat concrete, vadose zone soils (for unsubmerged tanks), and saturated zone flow and soils. Release of radionuclides initially present in annular regions of Type I and II tanks is also reduced by the annular fill grout. The barriers' capabilities are modeled over varying timeframes and degrees of effectiveness.

The NRC staff focused its review on those barriers that most serve to limit the timing and magnitude of the peak dose in order to better understand the uncertainty in DOE's demonstration of compliance with the 10 CFR Part 61 performance objectives. Because DOE estimates the magnitude of the peak dose over longer periods of performance to exceed 1 mSv/yr (100 mrem/yr), the NRC staff carefully reviewed DOE's analysis and support for the performance of two key barriers that result in the delay of key radionuclide releases and peak dose for thousands to tens of thousands of years. These two barriers, which were also the focus of the NRC staff's FTF TER (Camper, 2011 [ML112371751]) are (1) carbon steel tank liners and (2) chemical barriers (present in the tank grout or contaminated zone).

When carbon steel tank liners are intact they function as the primary barrier to radionuclide release from the tank. They are responsible in the base case (Case A) for precluding the release of radionuclides from 25 of the 29 HTF tanks with initially intact liners for thousands, in the case of Type IV tanks, to tens of thousands of years for the remaining tank types. These carbon steel liners assist in eliminating releases of shorter-lived radionuclides and delaying releases of longer-lived radionuclides for nearly all of the tanks beyond the 10,000-year period over which compliance with the performance objectives are demonstrated. While faster steel liner failure times can be realized for tanks with initially failed liners (see alternative Cases B through E), these alternative cases account for only 25 percent of the realizations in DOE's HTF probabilistic assessment as discussed in Section 4.2.19. Thus, steel liner failure times are dominated by Case A that has relatively longer steel liner failure times for initially in-tact tanks (e.g., only a fraction of a percent of Type I, II, and III/IIIA tanks with initially in-tact liners fail within 10,000 years in Case A).

Sections 4.2.8.2 and 4.2.9.2 summarize DOE's technical basis for the estimated steel liner failure times used in the HTF PA and the NRC staff's concerns with DOE's model support for steel liner performance, respectively. Although assumptions regarding steel liner performance are risk-significant, the NRC staff concludes, as it did for the FTF TER (Camper, 2011 [ML112371751]) that methods to obtain additional support for steel liner lifetimes may not be readily available and other key technical issues such as solubility limits for key radionuclides may be more tractable.

Once carbon steel tank liners have failed, DOE expects the fill grout to chemically buffer the water contacting the waste within the primary liner. The chemistry of the water contacting the waste is used to determine the solubility limits of radionuclides, which affect their release rates. The effect of the chemical transitions on radionuclide release varies according to the radionuclide. For some key radionuclides, DOE expects the presence of reducing conditions to inhibit their release (e.g., Np-237) through low solubility limits. Other key radionuclides are modeled with releases that are relatively insensitive to the chemical transitions (e.g., Pu-239 and Tc-99). The NRC staff is concerned that DOE's estimates for radionuclide solubility limits in the HTF PA, which are summarized in Section 4.2.8.3, may underestimate the releases of key radionuclides that could actually occur by one or more orders of magnitude, particularly for key radionuclides such as Tc-99, Pu-239, and Np-237, depending on actual conditions at the HTF. The NRC staff details its concerns with DOE's support for solubility limits for key radionuclides in Section 4.2.9.3. Similar to the FTF TER (Camper, 2011 [ML112371751]), given the importance of solubility limits on the magnitude of peak dose, the NRC staff primarily recommends that DOE develop additional support for the solubility limits for key radionuclides.

DOE, as modeled in its base case (Case A), estimates that the chemistry of the water contacting the CZ in every tank (including those with liners assumed to be failed initially) will remain reducing for more than 7,000 years. A reducing environment maintains low solubility limits for some key radionuclides that are sensitive to the reduction-oxidation potential of the water contacting the waste, such as Np-237. In the HTF PA, once the chemistry transitions to an oxidizing environment, solubility limits increase for these redox-sensitive radionuclides (e.g., Np-237), but remain very low for others that are modeled as relatively insensitive to redox conditions (e.g., Pu-239, Tc-99). When the chemistry transitions to a lower pH, the solubility limits of some pH-sensitive radionuclides (e.g., Np-237) rise. DOE estimates this will occur at around 20,000 years or beyond for all tank types except Type I tanks, which transition after approximately 8,000 years (no liner) and 11,000 years (liner),. Transition times occur earlier in alternative cases due to the possibility of bypassing pathways through which water contacting the waste is not conditioned by the entire fill grout mass.

The NRC staff concludes in Section 4.2.19 that the uncertainty in timing of the chemical transitions is likely understated given (1) the potential for bypassing pathways through the tank grout, (2) uncertainty in geochemical modeling, and (3) uncertainty in the ability of the tank fill grout to chemically condition water contacting the waste. The NRC staff is concerned that DOE's estimates for chemical transition times, similar to its estimates for radionuclide solubility limits, (see Section 4.2.8.3) may lead DOE to underestimate, in the HTF PA, the releases of key radionuclides that could actually occur by one or more orders of magnitude, particularly for key radionuclides such as Tc-99, Pu-239, and Np-237, depending on actual conditions at the HTF. The NRC staff details its concerns regarding DOE's support for solubility limits of key radionuclides in Section 4.2.9.3.

Other barriers that influence the timing and magnitude of the peak dose are tank vault concrete (in particular the tank basemat), annular grout, and saturated zone travel times to the compliance boundary. In the HTF PA, radionuclides that are released from the CZ after failure of the steel liner generally travel downward through the basemat before being released to the vadose zone soil below the tank or to the saturated zone soil in the case of submerged tanks. Cementitious material K_ds are important for estimating the migration of key radionuclides, such as Np-237 and potentially Pu-239 (depending on actual annular release or solubility limit in the

CZ), through the concrete basemat. Annular grout, which is estimated in the HTF PA to degrade slowly both hydraulically and chemically, limits the flow of water and delays the transport of radionuclides originating in the annular inventories of Type I and II tanks. Delayed release of annular inventories, which are outside of primary containment, is particularly important to reducing the risk from (1) shorter-lived radionuclides such as Sr-90 and Cs-137 and (2) longer-lived radionuclides such as Tc-99, whose K_d significantly decreases when the annulus grout transitions from reducing to oxidizing conditions, and Pu-239. However, the doses from Tc-99 originating from the annulus may be limited, due to inventory, and delayed release may actually result in a higher peak dose from Tc-99 due to superposition with releases of Tc-99 from the CZ. In Sections 4.2.9.3 and 4.2.9.4, the NRC staff details concerns with the adequacy of the technical support for the cementitious K_ds of key radionuclides, as well as the adequacy of the evaluation of releases from the annuli of Type I and II tanks. Finally, travel times of radionuclides through the saturated zone are a function of the Darcy velocities, as well as K_ds of key radionuclides, such as Pu-239 and Ra-226, in saturated zone soils. The NRC staff has identified issues associated with the HTF groundwater model that are discussed in Sections 4.2.9.4 and 4.2.11.2.

These barriers are the focus of the 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 HTF PA and supporting references, the NRC staff makes several recommendations related to additional follow-up work DOE could conduct during the monitoring period to strengthen its compliance demonstration.

4.2.3.3 Performance Assessment Overview Review Results and Recommendations

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

- In the FTF TER (Camper, 2011 [ML112371751]), the NRC staff recommended DOE perform a systematic scenario analysis process. DOE's inclusion of an analysis to identify, screen, and disposition FEPs, *ex post facto*, improves DOE's HTF PA documentation. However, due to the lack of transparency and traceability regarding screening decisions and disposition of included FEPs in the HTF PA, it remains difficult for the NRC staff to determine whether FEPs are comprehensively evaluated in DOE's HTF PA.

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

With respect to DOE's HTF PA approach, the NRC staff makes the following recommendations, along with a rating of risk significance and priority[11]:

1. DOE should improve its documentation of the screening and disposition of FEPs in the HTF PA to enhance transparency and traceability including the use of expert judgment during the screening and implementation of included FEPs in the HTF PA. (Low-to-Medium Risk-Significance, Intermediate-Term)

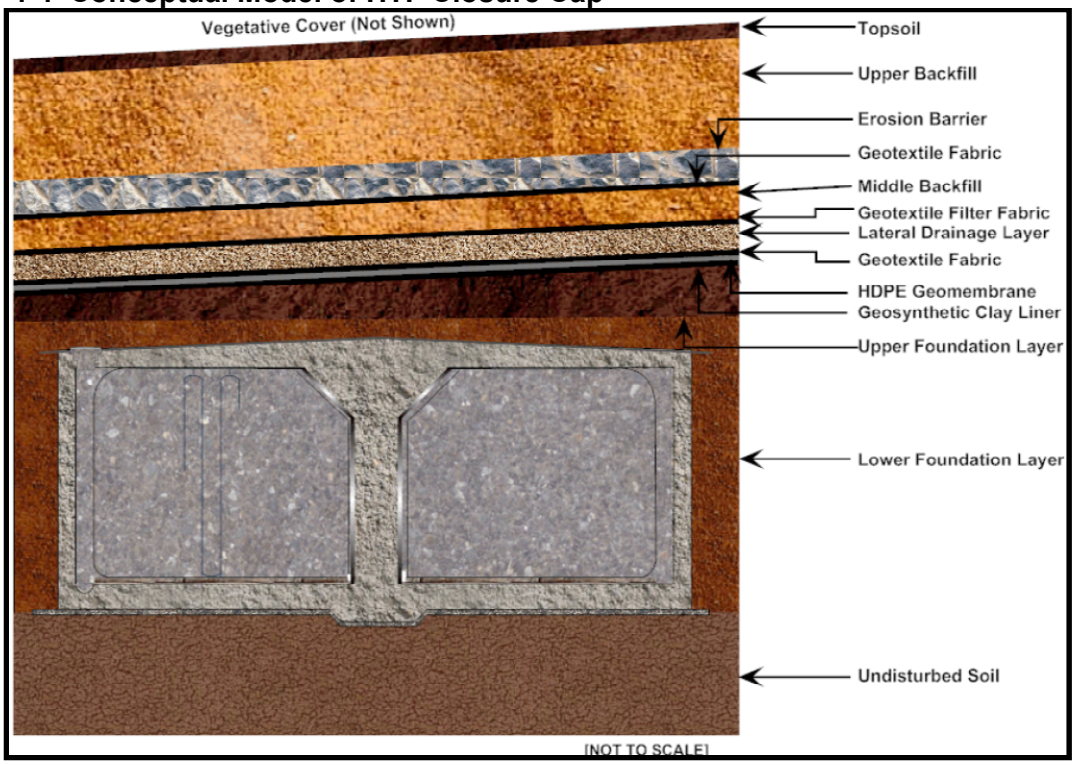
2. DOE should initiate additional follow-up work during the monitoring period to provide support for key modeling assumptions and barriers relied on for long-term performance in DOE's HTF PA as discussed in more detail in the Sections that follow (see individual recommendations for risk-significance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

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. The design of the closure cap for the HTF is similar to that for the FTF (Figure 4-4) Performance is dependent on the various closure cap layers that are shown in Figure 4-4 and described in Section 4.3 of WSRC-STI-2007-00184, Rev 2 for the FTF. Although the internal structure of the closure cap design for the HTF is the same as that for the FTF, the overall layout of the HTF closure cap is different than that for the FTF. Whereas the closure cap for the FTF will consist of a single engineered soil cover, the closure cap for the HTF will consist of two engineered covers (referred to as the West Cap and East Cap) with a drainage valley separating the two covers, as shown in Figure 4-5. After installation of the closure cap, (1) 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; and (2) the 100-yr institutional control period will be 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 HTF closure.

[11] 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 (1) optional or (2) contingent on results of other analyses.

Figure 4-4 Conceptual Model of HTF Closure Cap



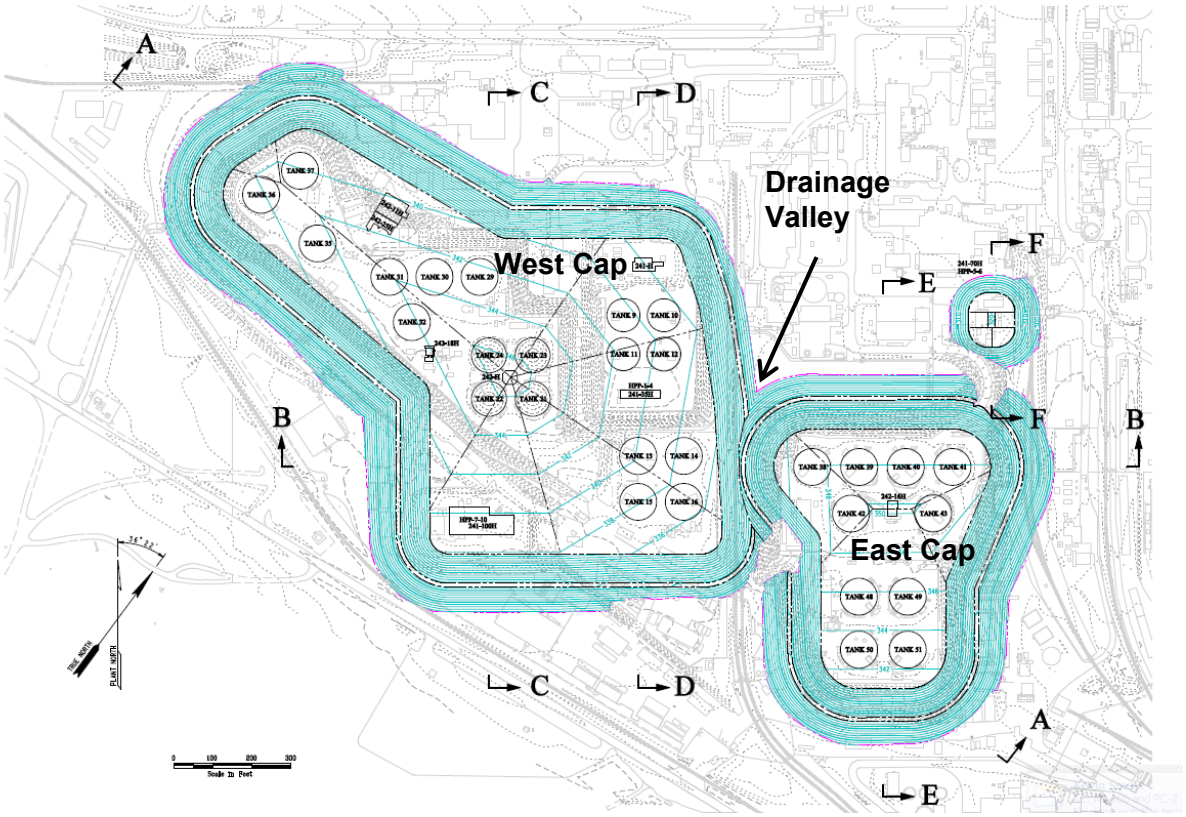
Adapted from Figure 3.2-84 in SRR-CWDA-2010-00128, Rev. 1.

4.2.4.1 Infiltration

Limiting infiltration reduces the potential for advective transport of radionuclides. However, in the HTF PA sensitivity and barrier analyses (see Section 4.2.18.4 and 4.2.18.5), the closure cap is shown to be important to limiting flow through the tanks for the first few thousand years, but to have a minimal impact on peak radionuclide doses within 10,000 yrs. Infiltration rates reach steady-state conditions by approximately 2,600 yrs after closure with infiltration ranging from 29.1 to 29.6 cm/yr (11.5 to 11.7 in/yr).

DOE tested several closure cap configurations with the HELP model and selected a final conceptual design configuration that results in the least amount of net infiltration and provides 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 the invasion of loblolly pine trees. The proposed composite hydraulic barrier is a high-density polyethylene (HDPE) geomembrane underlain by a geosynthetic clay liner (GCL). The GCL is assumed to plug all

Figure 4-5 HTF Closure Cap Conceptual Design Footprint



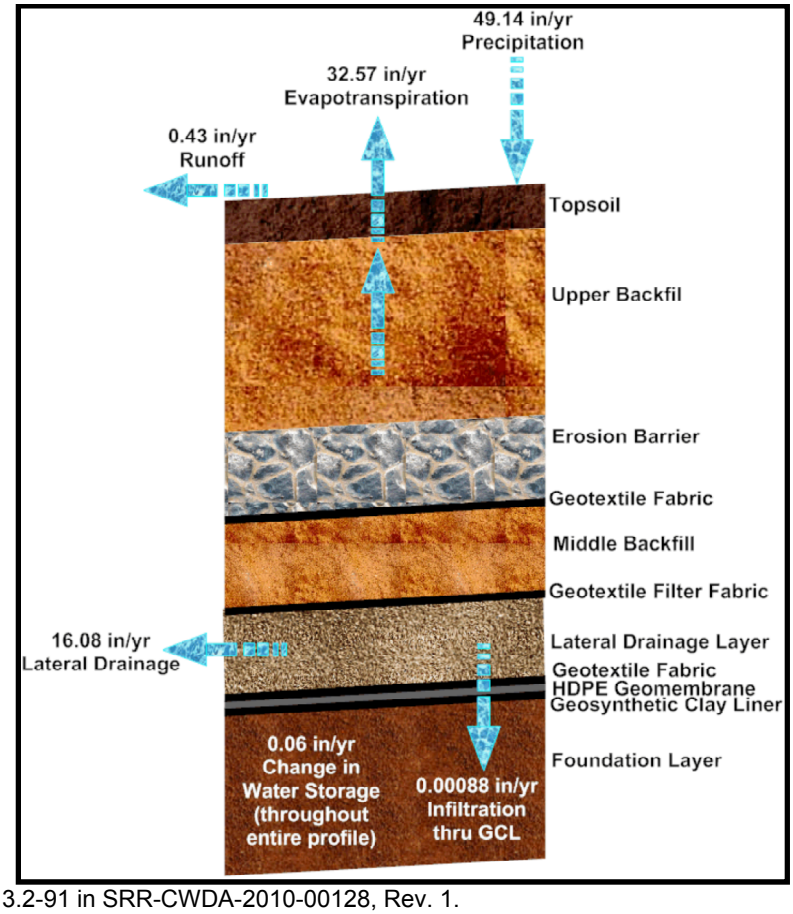
Adapted from Figure 3.2-85 in SRR-CWDA-2010-00128, Rev. 1.

the holes that may develop in the geomembrane until the holes have been penetrated by loblolly pine tap roots. DOE considers a range of potential closure cap degradation mechanisms which are discussed in Section 3.2.4.7 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1).

DOE performed 100 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 4-6. 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. The results of this simulation are shown in Table 3.2-14 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1). Estimated average annual net infiltration through the closure cap is then used as an upper boundary condition to the HTF vadose zone model, which is described further in Section 4.2.8. DOE also simulates 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 to the historical background infiltration rate of 37.7 cm/yr (14.9 in/yr).

The infiltration rates DOE uses in the HTF PA are all based on water balance simulations performed for the FTF conceptual closure cap design. As noted previously, although the basic cover construction for the HTF is the same as that for the FTF, a drainage valley is planned between the East and West Caps of the HTF closure cap, allowing a portion of the water

Figure 4-6 Closure Cap Water Balance



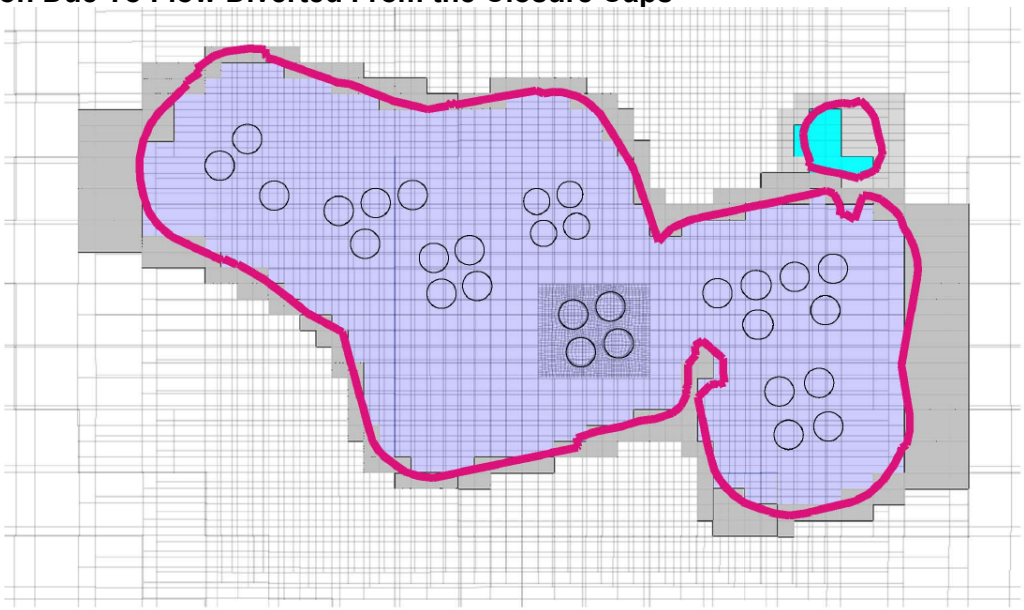
Adapted from Figure 3.2-91 in SRR-CWDA-2010-00128, Rev. 1.
To convert in/yr to cm/yr, multiply by 2.54.

entering the cover drainage layers over the East and West Caps to flow into the drainage valley. The FTF closure cap, on the other hand, would cover a single area and all water flow through the drainage layer would be diverted away from the cap footprint. Some portion of the water flow through the drainage valley at HTF may infiltrate into the underlying vadose zone. Portage, Inc. (PORTAGE-08-022, Rev. 0) modeled the effect of focused infiltration along the perimeter of the HTF caps as part of their HTF modeling exercise, but they did not model focused infiltration into the drainage valley (Figure 4-7). The authors assumed that one-third of the diverted water would infiltrate and the rest would leave the model domain by surface flow.

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 (10 ft) of material above the waste tanks and significant ancillary equipment. Typically, agricultural and residential intruder scenarios consider a nominal excavation depth of 3 m (10 ft). Therefore, proper design, construction, and performance of the erosion barrier should limit surface water erosion and direct contact with the waste by potential inadvertent intruders. The erosion barrier is designed to limit erosion to the underlying cap layers, however, the vegetative cover, topsoil,

Figure 4-7 PORTAGE Model Grid and Gray Shaded Areas Assigned Focused Perimeter Infiltration Due To Flow Diverted From the Closure Caps



Adapted from Figure 3-16 in PORTAGE-08-022, Rev. 0.

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 evaluates the physical stability of the closure cap with respect to a probable maximum precipitation (PMP) event, consistent with NUREG-1623. The PMP is defined as the theoretically greatest depth of precipitation that is possible during a given time period over a given area at a particular geographic location. Based on the PMP, DOE evaluates the design criteria for the vegetative cover, erosion barrier, side slopes, and toe of the side slopes. Although the methodology presented in NUREG-1623 addresses a 1,000-year timeframe, DOE states in an RAI response for the FTF review that the SRS-specific PMP event provides assurance of closure cap stability against gullying for the 10,000-year compliance period (Response to FTF Comment IE-7; SRR-CWDA-2009-00054, Rev. 0).

DOE's 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, DOE assumes that by that time the HDPE and GCL layers would be degraded by other mechanisms. Accordingly, animal burrowing is not considered a significant risk.

The projected long-term topsoil loss is determined according to the Universal Soil Loss Equation for both vegetative cover conditions (i.e., Bahia grass and pine forest). DOE predicts approximately 3.7 cm (1.45 in) of soil loss for the topsoil and no reduction in the upper backfill layer within 10,000 years, as discussed in Section 8.2 of WSRC-STI-2007-00184, Rev. 2.

Riprap for the integrated drainage system ditches has not yet been sized because the project is still in its early phases and there is a lack of a detailed closure cap drainage system. DOE expects to select the riprap material for the erosion barrier, side slope, and toe of the side slope from local granite or mylonitic quartzite quarries.

4.2.5 NRC Evaluation of Infiltration and Erosion Control

The closure cap design information that DOE provided to the NRC staff is preliminary and will be finalized closer to the actual time of closure of HTF. The designs 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 ensure that 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).

4.2.5.1 NRC Evaluation of Infiltration

Infiltration is usually a sensitive parameter value in PAs because it is directly related to the flux of contaminants into the groundwater. However, the performance of a series of additional barriers can limit the sensitivity of the closure cap performance to the overall performance of the site. As discussed by the NRC staff in the FTF TER (Camper, 2011 [ML112371751]), the sensitivity and uncertainty analyses in the FTF PA (SRS-REG-2007-00002, Rev. 1) did not identify any risk-significant closure cap parameters as a result of the modeled performance of the steel tank liners. DOE assumes that by the time that the earliest steel liners at the FTF are assumed to fail, degradation of the closure cap has already resulted in steady-state infiltration. Accordingly, the cap would have a minimal impact on peak doses. However, several of the HTF steel liners are assumed to be initially failed and several of the tanks have contamination outside of the primary steel liners. Accordingly, the closure cap could be an important barrier to the release of radionuclides and the timing of peak dose at HTF. Short-lived radionuclides could be mobilized prior to decaying to insignificant levels. The near-field model does not explicitly consider lateral groundwater flow, thereby making early modeled infiltration through the closure cap potentially important to the release of short-lived radionuclides from tanks that are in contact with the groundwater. If groundwater flow is more explicitly considered in the near-field model, then early infiltration may not be as significant. Also, the peak dose could be shifted earlier in time if the closure cap does not limit infiltration during the first couple thousand years.

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 (12.5 in/yr). Consequently, the long-term steady-state infiltration rate, which ranges from 27.1 to 29.6 cm/yr (10.7 to 11.7 in/yr), is less than the median background value of 37.7 cm/yr (14.9 in/yr). DOE states that the assumed saturated hydraulic conductivity for the upper foundation layer is reasonable, as the value is typical for soil-bentonite blends (SRR-CWDA-2010-00033, Rev. 1). 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 0.3-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 assumes that the degradation of the drainage layer (i.e., a reduction in hydraulic conductivity) will be controlled by colloidal infilling of the pore space within the lateral drainage layer from the overlying backfill. DOE's approach does 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. As such, infiltration through the closure cap could increase earlier than expected by DOE.

DOE modeling indicates that saturated conditions will occur above the composite layer (Response to FTF RAI IEC-7; SRR-CWDA-2011-00044, Rev. 1). An initial hydraulic head of 15.60 cm (6.14 in) above the HDPE geomembrane is predicted to increase until year 2,623 when it ranges from 72.44 to 77.09 cm (28.52 to 30.35 in) throughout the remainder of the 10,000-year compliance period. In Section 3.2.4.7 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1), DOE states that conservative modeling assumptions (e.g.,silting-in without clay mobilization and deep root penetration of the HDPE) tend to restrict modeled lateral drainage from the closure cap and result in an average annual head on the HDPE geomembrane that is greater than DOE anticipates. Should the buildup of hydraulic head occur, DOE does not think it would adversely impact the physical stability of the closure cap, vegetation, erosion, or the performance of the composite layer (Response to FTF RAI IEC-7; SRR-CWDA-2011-00044, Rev. 1). Based on limited model support, it is difficult for the NRC staff to assess (1) the likelihood of hydraulic head buildup within the cover or (2) its potential impact on closure cap performance. A more realistic representation of infiltration and saturation within the proposed closure cap is needed to assess the potential for buildup of hydraulic head. If an analysis containing a more realistic representation determines that the buildup of hydraulic head is likely, an explicit evaluation of the physical stability of cover materials under this condition would be needed.

Portage, Inc. (PORTAGE-08-022, Rev. 0) modeled the effect of focused infiltration along the perimeter of the HTF caps as part of their HTF modeling exercise. Based on guidance from SRS staff, they assume that one-third of the diverted water would infiltrate and the remainder would leave the model domain by surface flow. The area of focused infiltration simulated, as documented in PORTAGE-08-022, Rev. 0 (Figure 4-7) is the outer perimeter of the combined East and West Caps, and does not include the drainage valley between the caps. The modeling in PORTAGE-08-022, Rev. 0 also assumes that the perimeter drainages would continue to receive ambient recharge at rates estimated from the SRS Nominal GSA model. The ambient recharge rates are similar to the rates estimated for diverted flow, so the net recharge around the perimeter of the caps is approximately twice that for the case without a cap.

In the absence of details about the construction of the perimeter drainage system, particularly in the area where the East and West Caps will merge, and without simulations of the effect of focused infiltration from the drainage features, the risk significance of perimeter infiltration is difficult for the NRC staff to evaluate. Conceptually, the HTF PA model results in less total flow

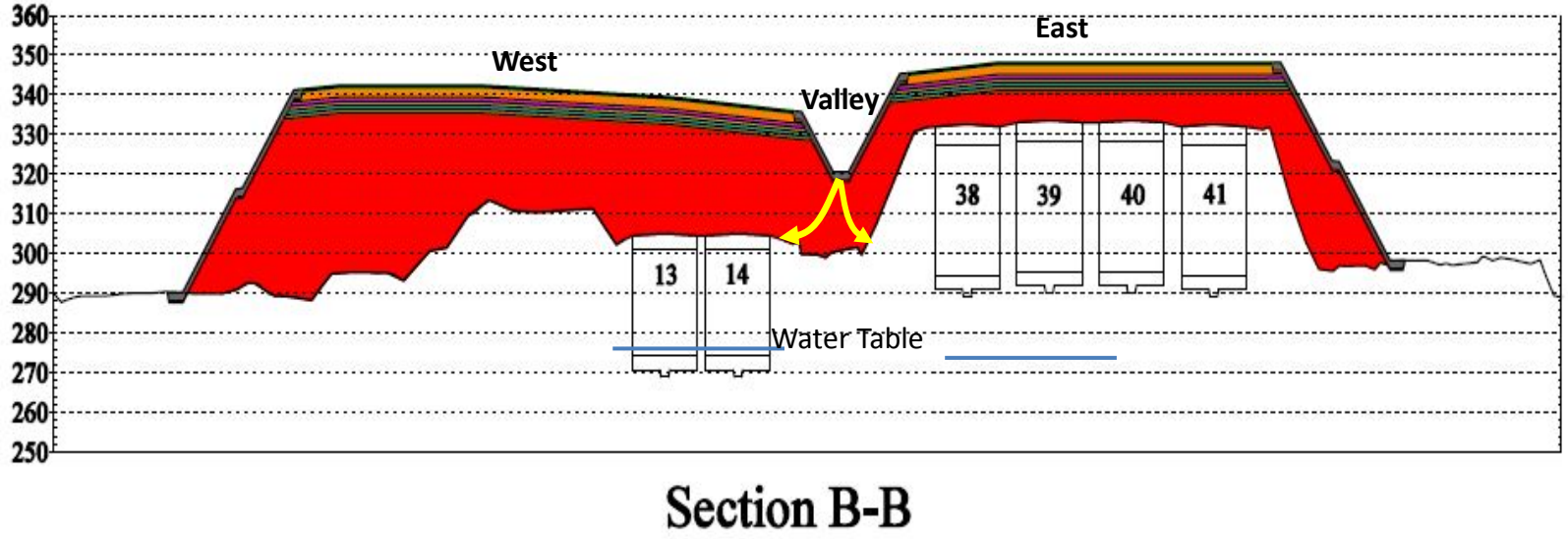
into the vadose and saturated zones during the period when the caps reduce net infiltration, because the modeling assumes that all of the water that is diverted by the caps leaves the model domain. Adding the effect of perimeter infiltration, which could conceivably be greater than that simulated in PORTAGE-08-022, Rev. 0, would increase the groundwater flow rate outside the footprint of the caps, but would not increase the vertical flux directly affecting releases from the tanks. Increased perimeter infiltration would also likely change flow patterns between the source areas and the 100-m (330-ft) boundary (Shaffner, 2013b [ML13126A127]). Focused recharge in the drainage valley between the East and West Caps could flow to the east and west, as illustrated by the yellow arrows in Figure 4-8. This focused recharge would tend to increase the water table elevation under portions of the West Cap where several of the tanks are already partially or fully below the water table, and decrease the distance from the tank basemats to the water table for the other tanks. DOE should evaluate the potential effects of focused perimeter recharge, including that in the drainage valley. The analysis should include appropriate refinement of the grid cells receiving recharge and a well-supported value for the diversion of flow. The assumption that only one-third of the diverted flow infiltrates into the subsurface does not appear to have a technical basis. Based on the HELP modeling reported in the HTF PA, most of the water flowing to the perimeter of the caps will be through the drainage layer after passing through the upper backfill; thus, flow to the perimeter drainage system will be gradual (in comparison with episodic surface runoff), allowing more time for water to infiltrate in the perimeter drain.

In the HTF PA, the closure cap limits the flow, and therefore, the release of radionuclides in the first few thousand years. However, the NRC staff expects closure cap performance to be more risk-significant if the performance of additional barriers (e.g., tank and annulus grout, intact steel liners, natural system) is determined to be optimistic. The 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 systems adequately represent the real-world systems influenced by the presence of the closure cap including both the near-field and far-field domains. 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 its 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 for protection of inadvertent intruders and to provide suitable conditions for the vegetative cover and topsoil. To mitigate the potential effects of erosion by surface water, erosion protection designs must 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 are consistent with NUREG-1623. This guidance document specifically addresses a 1,000-year timeframe rather than the 10,000-year compliance period. DOE states that the SRS-specific PMP has a low probability of occurrence and is a bounding event, thereby providing assurance of the physical stability of the closure cap design for the 10,000-yr compliance period (Response to FTF Comment IE-7; SRR-CWDA-2009-00054, Rev. 0). The NRC staff determined, in that review, that DOE's estimates for the PMP are reasonable and that the probability of such an event being equaled or exceeded is very low. Accordingly, the PMP is considered by the NRC staff to provide a reasonable design basis.

Figure 4-8 Cross Section Through HTF Closure Caps Illustrates Drainage Valley



Adapted from Figure 3.2-86 SRR-CWDA-2010-00128, Rev. 1.
Note: Elevations are in feet AMSL. To convert feet to meters, multiply by 0.3048.

Long-term maintenance of the topsoil and vegetative cover is important to closure cap performance as evapotranspiration [82.7 to 84.2 cm/yr (32.6 to 33.2 in/yr)] dominates the modeled water balance distribution for SRS precipitation [124.8 cm (49.1 in)]. DOE uses the Universal Soil Loss Equation to predict topsoil losses throughout the compliance period. However, those predictions may not adequately account for complex and uncertain processes that may contribute to soil loss and the initiation of gullying over long time periods. An evaluation of the cumulative effects from precipitation events over long time periods with respect to gully formation is needed to support predictions of the long-term performance of the topsoil and vegetative layers.

The NRC staff concludes that the closure cap, as designed, can provide adequate long-term erosion protection. However, it remains for DOE to demonstrate that certain aspects of the designed performance can be achieved (e.g., evaluation of an acceptable rock source, the ability of an integrated drainage system to accommodate design features). Although the design will not be made final until closer to the time of site closure, verification that certain design features can be implemented as designed is needed in advance of site closure to allow sufficient time to change the closure cap design or the assumptions regarding long-term erosion protection, if necessary. These modifications may be important to the overall evaluation of closure cap performance.

4.2.5.3 Infiltration and Erosion Review Results and Recommendations

Similar to what is noted in the NRC staff's FTF TER (Camper, 2011 [ML112371751]) and FTF Monitoring Plan (Camper, 2013a [ML12345A322]), with respect to infiltration and erosion, the NRC staff notes the following for HTF: (1) DOE's approach to assessing closure cap performance is reasonable for planning purposes; and (2) DOE has provided sufficient information regarding long-term erosion protection of the closure cap.

Notwithstanding (1) and (2) above, the NRC staff notes 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.

Similar to what is noted in the NRC staff's FTF TER (Camper, 2011 [ML112371751]) and FTF Monitoring Plan (Camper, 2013a [ML12345A322]); the NRC staff's recommendations for infiltration and erosion include the following along with a rating of risk significance and priority (see the footnote in Section 4.2.3.3 for a description of the risk significance and priority of the recommendations):

1. DOE should provide additional model support for (1) the long-term hydraulic conductivity of the upper foundation layer and lateral drainage layer, and (2) the long-term erosion of the topsoil layer (see FTF Monitoring Factors 5.1 and 5.2; Camper, 2013a [ML12345A322]). (Low Risk Significance, Intermediate Term [Prior to Final Closure])

2. DOE should provide additional model support to understand the effects of perimeter infiltration and focused infiltration in the drainage valley between the East and West

Caps on near-field and far-field groundwater flow patterns and radionuclide transport. (Low Risk Significance, Intermediate Term [Prior to Final Closure])

3. Prior to completing the final closure cap design, DOE should conduct 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; see FTF Monitoring Factor 5.2; Camper, 2013a [ML12345A322]). (Low Risk Significance, Intermediate Term [Prior to Final Closure])

4.2.6 Radionuclide Inventory

Section 3.1 first discusses the HTF 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 HTF inventory, (2) the development of the inventories for those radionuclides expected to be present in HTF 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 a reasonable approach to developing tank and auxiliary equipment inventories for the HTF facility using information obtained from previous FTF experience. For example, DOE assumes a reasonably conservative volume informed by the final estimated volumes of FTF tanks that have been cleaned. DOE estimated concentrations based on data in the WCS or averaged across similar tanks for those radionuclides where WCS data were not available. For conservatism, DOE assumes the maximum concentrations for radionuclides within a given tank group. DOE adjusted the initial projected inventories for specific radionuclides based on the final sampling results from FTF tanks. Staff notes that since some of the inventory adjustments are based on the cleaning experience obtained with baseline technologies, DOE may need to revisit those assumptions should the baseline technologies change.

Inventory can be more or less risk significant depending on the mix of radionuclides present in the waste and the expected chemical environment of the contaminated zone. For example, inventory is generally more risk-significant for radionuclides that are not solubility limited. The potential dose contributions of solubility-limited radionuclides may not be sensitive to the assumed inventory, although inventory may be important in certain cases. For example, radionuclide mass may slowly leach out of the tanks with a low, non-risk-significant solubility for long periods of time. Later, when chemical transition to a higher solubility occurs, the radionuclide inventory may be depleted, thereby limiting or completely eliminating the impact of the later, more risk-significant release at higher solubility that could have been realized with a higher initial inventory. The inventory of solubility limited radionuclides could also be important if a down-stream barrier (e.g., concrete vault basemat) controls the release of the radionuclides. It is important to note that many key radionuclides at HTF are 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., Tc and Pu). One risk-significant radionuclide, Ra-226, is not expected to be present initially in the tanks at any appreciable quantity but is expected to be produced over time by its predecessors: Th-230, U-234, and Pu-238. These predecessors of Ra-226 may be present, initially or over time, in risk-significant quantities in HTF tanks. Thus, little uncertainty may be associated with the Ra-226 inventory, although the inventory of its

parents may be an important consideration when assessing compliance. Of course, there is always the possibility that the waste in HTF has a radionuclide 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 HTF PA.

For most HTF tanks, DOE develops a projected inventory using the maximum concentration from any tank within a tank type grouping (e.g., Type I&II, III/IIIA, and IV). Concentrations of radionuclides in the tanks are determined from the WCS or via special calculation. With respect to the expected residual volume to remain following waste retrieval from HTF tanks, DOE assumes that 15 m^3 (4,000 gal) will remain in the tanks. Finally, the probabilistic analysis also considers uncertainty in the inventory estimates. DOE assumes a log-uniform distribution that ranges from a factor of 100 times less to a factor of 10 times greater than the inventory, depending on tank type and radionuclide. The distribution is skewed low for tanks that have not yet been cleaned because DOE expects that the final volume will be less than the volume assumed in the inventory projection.

After each tank is cleaned, the amount of residual material remaining in the tanks and annuli will be determined using visual inspection techniques to estimate volumes and sampling to estimate concentrations. DOE has not completed final characterization of any HTF tanks. The process for final characterization for the HTF tanks will be similar to that used for the completed FTF tanks. Sections 3.1 and 3.2 of this TER contain 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 for residual radioactivity remaining in cleaned tanks, DOE attempts 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 HTF PA results, as many of the radionuclides driving facility risk are solubility limited. DOE assumes some measure of risk in a Type II decision error (i.e., assuming that the performance objectives can be met, when in fact they cannot) if one or more of the following conditions are true:

- DOE significantly underestimates inventory for key radionuclides whose inventory significantly impacts peak dose.

- DOE cannot defend solubility limits (e.g., Tc-99) for solubility-limited key radionuclides that have limited impacts on peak dose.

- Significant mass depletion of HRRs occurs before transition to a higher solubility limiting phase, particularly if the lengthy chemical transition times assumed in the PA modeling prove overly optimistic.

However, compliance risk is mitigated if DOE fulfills its commitment to develop post-retrieval inventory estimates based on sampling and evaluates the dose impacts associated with the final HTF inventory while considering uncertainty in the final inventory and PA modeling assumptions

(e.g., solubility assumptions). Therefore, the NRC plans to monitor DOE's commitment to characterize cleaned tanks following waste retrieval.

In Section 3.2 of this TER, the NRC staff reviews DOE's approach to developing the final inventory for tanks after they have been cleaned. Section 3.2 also lists the NRC staff's review results and recommendations with respect to DOE's plans to sample HRRs and to consider uncertainty in final inventory estimates for tanks that have yet to be cleaned or for which final inventory estimates have not yet been developed. See Section 3.2 for additional information.

The NRC staff finds that DOE's approach to developing inventories for tanks that have yet to be cleaned is reasonable for the purposes of assessing HTF risk, prior to development of final inventories following waste retrieval activities. As a result of the NRC staff's finding, the NRC staff recommends that DOE continue its commitment to sample each tank following waste retrieval activities. NRC staff will follow-up with DOE on the 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 by considering two primary sources of contaminants: waste tanks and ancillary equipment. Waste tanks refer to the 29 subsurface tanks in the HTF that have been used to store aqueous wastes; ancillary equipment refers to equipment used in the HTF 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 that fills the emptied tanks and the HTF 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 ICM (see Section 4.2.2) to simulate the release of radiological and chemical constituents from the 29 underground waste tanks and associated ancillary equipment, and to simulate the migration of the contaminants through soil and groundwater (see Figure 4-2). The ICM includes simplified representations of the actual physical infrastructure of the waste tanks. These simplified representations include:

- The depiction of each discrete waste tank segment or area as homogeneous (i.e., ignoring interior elements, such as columns, cooling coils, in-place mixers and other equipment likely to be left within the tanks and/or penetrations through the area, such as waste tank risers, transfer lines);

- The use of minimum segment thicknesses in the baseline analysis for areas with variable thicknesses (e.g., waste tank walls, tank tops); and

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

In the ICM, 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 concrete is assumed to be initially intact and degrade over time through cracking (SRNL-STI-2012-00465). The concrete hydraulic conductivity and moisture characteristic curves (MCCs) are subsequently modified by blending intact matrix and fracture properties over time. DOE uses information from a cementitious materials degradation analysis, summarized in Section 4.2.8.1, to derive the hydraulic properties of concrete degrading over time.

After passing through the concrete waste tank top, the water travels into (1) the tank grout (for Type IV tanks and after liner failure for Type I, II, III, and IIIA tanks); (2) the annulus grout (for Type I, II, III, and IIIA tanks); or (3) reaches the steel tank top liner (before liner failure for Type I, II, III, and IIIA tanks). Before steel liner failure, the tank liner is modeled as impermeable to both advection and diffusion and water flow is diverted around the liner. After failure, DOE states that the liner no longer serves as a barrier to advective or diffusive flow and has no further effect on the model results. DOE estimates the steel liner failure time for each tank type from corrosion analyses, except for Tank 12 (Type I) and Tanks 14-16 (Type II), which are assumed to have failed liners at the time of HTF facility closure because they have a significant number of leak sites. The corrosion analyses are summarized in Section 4.2.8.2.

Water that enters the waste tank grout travels downward to the 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 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 time to reach a fully degraded state is derived from a cementitious materials degradation analysis. Degradation of the grout over time is accounted for by blending intact matrix and fracture properties to derive grout hydraulic conductivities and MCCs (SRNL-STI-2012-00465). In some alternative scenarios, fast flow paths through the tank grout are modeled that result in a higher flow rate through the grout.

The ICM 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 (or fast flow path) leaches radionuclides from the CZ. For Type I and II tanks at HTF, there are also contaminants located outside of the primary liners in the annuli between the primary and secondary liners. This additional contamination, resulting from leaks in the primary tanks during operations, is modeled within the ICM as a thin layer within the bottom of the annulus grout for Type I tanks. For Type II tanks, the additional contamination is modeled in the ICM within a sand pad layer between the primary and secondary liners (referred to as the primary sand pad) beneath the primary tank. Because Tank 16 experienced an overflow of its secondary liner, DOE also models contamination within a sand pad layer beneath the secondary liner (referred to as the secondary sand pad). Sections 3.2.2, 4.2.8.3 and 4.2.8.4 summarize the inventories and releases from the annuli of Type I and II tanks.

DOE assumes the radionuclide release rate from the CZ to be (1) controlled by a fixed concentration limit (or solubility, in the broad sense of the term), (2) dependent on the chemical properties (e.g., pH, E_h) of the pore fluid passing through and interacting with the overlying tank grout, and (3) independent of the grout or CZ K_ds. The model accounts for the evolution over time of the physical and chemical conditions within the grouted waste tanks and how they affect radionuclide leaching from the CZ. 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.

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 (i.e., number of pore volumes) that passes through the cementitious material over time. The grout K_ds affect 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.

Transport of contaminants from the CZ upwards into the overlying tank grout through diffusion or advection, prior to liner failure or prior to the onset of appreciable flow (for tanks with failed liners), is controlled by grout K_ds (see Section 4.2.8.4). Although no solubility control is assumed for cementitious materials, solubility control in the CZ limits the diffusion of contaminants. Contaminants released from the CZ do not leave the tank until the tank liner fails. DOE estimates 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 except for Tank 12 (Type I) and Tanks 14-16 (Type II), which are assumed to have failed liners at the time of HTF facility closure because they have a significant number of leak sites. The analysis assumes that the primary liner and secondary liner (if present) fail simultaneously.

Contaminants within the annulus of Type I tanks and sand pads of Type II tanks can leave the tank prior to and after liner failure. Prior to liner failure, contaminants must migrate upward through the grout between the primary and secondary liners and past the top of the secondary liner to enter the tank vault concrete. Transport of contaminants from the annulus of Type I tanks and sand pads of Type II tanks upwards into the overlying annular grout and vault concrete through diffusion or advection is controlled by grout K_ds (see Sections 4.2.8.3 and 4.2.8.4); no solubility control is assumed for cementitious materials. Contaminants within Type II tank sand pads can also migrate upward into the CZ and overlying primary tank grout and are modeled similarly to contaminants that originate from the CZ, as discussed above.

After contaminants exit the outermost waste tank liner, they primarily 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, with the evolution of hydraulic properties of the basemat being similar to that of the grout. Degradation of the basemat over time is accounted for by blending intact matrix and fracture properties to derive basemat hydraulic conductivities and MCCs (SRNL-STI-2012-00465). In some sensitivity scenarios, 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", with respect to 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 (1) the vadose zone (e.g., soil) beneath the waste tank for the tanks above the water table (i.e., Types III/IIIA and IV), or (2) the saturated zone for the submerged and partially submerged tanks (i.e., Type I and II tanks). For Type III/IIIA and IV tanks, after contaminants exit the vadose zone, they enter the aquifers beneath the HTF. The contaminant fluxes for the submerged and partially submerged tanks are loaded into the far-field model source elements at the elevation of the Type I and II tank basemats (see Sections 4.2.10.2 and 4.2.11.1). Section 4.2.8.4 summarizes DOE's modeling of the vadose zone.

DOE models the release of radionuclides from the transfer lines and ancillary equipment separately from the waste tanks. The timing of radionuclide releases is dependent on the rates of general corrosion and pitting corrosion of the steel, which is discussed in Section 4.2.8.2. DOE applies a failure time of 510 years to all steel transfer lines and ancillary equipment in the HTF PA modeling. Once this equipment fails, the radionuclide inventory is assumed to be available for direct release into the surrounding soil. Transfer line inventory is modeled by distributing the assumed inventory equally over the entire HTF. DOE models the ancillary equipment (i.e., pump tanks, CTS tanks, and evaporators) as point sources located in the HTF at a central point of the individual components. Other ancillary equipment is not modeled explicitly.

4.2.8.1 Cementitious Materials Degradation

In the ICM, the rate and timing of degradation of the waste tank cementitious materials can vary with tank type and hydrologic condition, as based on analyses presented in SRNL-STI-2010-00035, Rev. 0 and SRR-CWDA-2010-00019, Rev. 0. The report SRNL-STI-2010-00035, Rev. 0 uses information from scientific literature to assess the potential for various degradation mechanisms to impact the HTF concrete and grout. The degradation mechanisms considered to be potentially important for the HTF cementitious materials are external sulfate attack, leaching, and carbonation-induced corrosion of rebar or cooling coil steel. DOE does 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 does 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 indicates that data from the literature are insufficient to address the likelihood that alkali-aggregate reaction will occur or the consequence this mechanism will have on the degradation of the HTF cementitious materials. Finally, the analysis does 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 (≤11 parts per million [ppm]).

To model the degradation of HTF cementitious materials, DOE uses simple empirical relationships and diffusion equations taken from published literature and reviewed in SRNL-STI-2010-00035, Rev. 0. These relationships and equations allow the calculation of the depth of penetration of potentially deleterious species into the cementitious material. The species DOE considers in modeling the chemical degradation of cementitious materials are SO_4^{2-} and Mg^{2+} for sulfate attack, Ca^{2+} for leaching, $CO_2(g)$ for carbonation-induced steel corrosion under unsaturated hydrologic conditions, and Ca^{2+} and CO_3^{2-} for carbonation-induced steel corrosion

under saturated hydrologic conditions. Carbonation-induced steel corrosion in unsaturated hydrologic conditions initiates sooner and propagates faster than in saturated conditions because gaseous diffusion of CO_2 is faster than aqueous diffusion of Ca^{2+} and CO_3^{2-}. However, DOE applies the higher—thus, more conservative—carbonation rate derived for unsaturated conditions in calculating the degradation rate of saturated concrete and grout, specifically those of Type I tanks that are completely submerged below the water table and Type II tanks with their base below the water table (SRR-CWDA-2010-00019, Rev. 0).

For Type I, II, III, and IIIA tanks, which contain cooling coils, carbonation is identified as the most aggressive chemical 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 SRR-CWDA-2010-00019, Rev. 0). The tank grout is assumed to be fully degraded once carbonation reaches half the grouted height. Concrete vault degradation is calculated to begin as early as 1,350 years, 2,550 years, 2,550 years, and 2,500 years for Type I, II, III, and IIIA tanks, respectively, with full degradation occurring after 2,700 years, 5,100 years, 5,100 years, and 5,000 years, respectively. Tank grout degradation is calculated to begin as early as 2,700 years, 5,100 years, 5,100 years, and 5,000 years for Type I, II, III, and IIIA tanks, respectively, with full degradation occurring after 13,200 years, 16,700 years, 19,200 years, and 19,100 years, respectively.

For Type IV tanks and annuli of the Type I, II, III, and IIIA tanks, which have no cooling coils, leaching is considered as the major chemical degradation mechanism for the tank grout, although carbonation-induced rebar corrosion is an important process for vault concrete degradation. Similar to Type I, II, III, and IIIA tanks, Type IV tank grout degradation is assumed to start once the carbonation effect has traversed the vault concrete and reached the grout. DOE takes no credit 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. Concrete vault degradation is calculated to begin as early as 400 years with full degradation occurring after 800 years. Type IV tank grout degradation is calculated to begin as early as 800 years with full degradation occurring after 64,400 years.

The HTF PA assumes that the concrete and tank grout exist as intact matrices initially, and then physically degrade over time through cracking (SRNL-STI-2012-00465). This approach differs from the FTF PA, where the hydraulic conductivity of the cementitious materials is assumed to increase linearly with time from the initial to fully degraded state (SRS-REG-2007-00002, Rev. 1). For HTF, the assumed degradation through cracking results in a non-linear increase in hydraulic conductivity with time and this degradation alters the MCCs (MCCs) of the cementitious materials. The matrix is assumed to develop through-going fractures of 0.127 mm (0.005 in) at an initial fracture spacing of 10 km (6 mi) (i.e., no fractures) to a final configuration of one fracture every 0.1 m (4 in). The assumed start and end times for degradation of the cementitious materials are provided in Table 4.2-30 of SRR-CWDA-2010-00128, Rev. 1. The hydraulic conductivities and MCCs for the cementitious materials are derived by blending matrix and crack properties (SRNL-STI-2012-00465). It is important to note that this approach is

intended to incorporate the increased flow due to fractures into an effective continuum (i.e., this approach is not modeling fractures and fracture flow explicitly). Flow through a fracture, or fast pathway, is represented in alternative cases B-E. In SRNL-STI-2012-00465, Figure 9 illustrates an example of the assumed unsaturated hydraulic conductivities for a Type IV tank over the period of degradation divided into 40 time intervals. The resultant flow through the cementitious materials is, in part, a function of the degree of saturation of the cementitious materials (i.e., suction head), which is discussed in Section 4.2.8.4. 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 base case (Case A) deterministic assessment in which no fast flow path exists through the waste tank system. Four other waste tank cases are analyzed to simulate the potential conditions in the HTF closure system over the modeling period. Table 4.4-1 in the HTF PA (SRR-CWDA-2010-00128, Rev. 1) provides a summary of the waste tank cases. In Cases B and D, DOE assumes that degradation of the cementitious materials begins at year 500, with full degradation occurring at year 501, with water flowing through the grout. Whereas in Cases C and E, DOE assumes that the cementitious materials degrade gradually, which is similar to Case A. In addition, a fast flow path is assumed to exist between the waste tank top and contaminated zone (e.g., from riser through cooling coil) in Cases B and C. The fast flow path is modeled with the hydraulic properties of gravel (Table 4.4-10; SRR-CWDA-2010-00128, Rev. 1). In Cases D and E, a fast flow path is assumed to exist through the entire closed system (i.e., through a tank riser, through a cooling coil, through the tank fill grout, and through the basemat) with water flowing primarily through the fast pathway. The reducing capacity of the overlying grout is assumed to chemically condition the infiltrating water in Cases A, B, and D. In Cases C and E, DOE assumes that the grout does not impart any chemistry changes to the water.

4.2.8.2 Steel Degradation

The ICM 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. After failure, waste release occurs assuming the steel is absent or otherwise not a hindrance to advection and diffusion. The time of initial waste release is tied to the integrity of the waste tank primary liner and the transfer lines. Secondary liners (for all but Type IV tanks, which lack secondary liners) are 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 that were initially conducted for the FTF PA and are presented in WSRC-STI-2007-00061, Rev. 2 for tank liners and in WSRC-STI-2007-00460 for transfer lines.

DOE extends the analysis documented in WSRC-STI-2007-00061, Rev. 2 to Type II tanks in SRNL-STI-2010-00047. DOE also evaluates the impact of differing environmental conditions between FTF and HTF on Type I tanks and transfer lines in SRNL-STI-2010-00047. The Type I tanks and some of the transfer lines at HTF are fully submerged, while at FTF the Type I tanks and transfer lines are in the unsaturated zone. As noted in C-ESR-G-00003, Rev. 5, Tanks 9 through 16 (Types I and II) have existing leak sites in their primary steel liners. For tanks with a significant number of leak sites or leak sites located near the bottom of the liner, DOE assumes

that the primary and secondary liners are failed at the time of closure and are not a barrier to flow. These tanks include Tanks 12 and 14-16. DOE does not consider the quantity of leak sites or their locations for Tanks 9-11 and 13 to be significant; therefore, DOE models these tanks according to the results of the analysis presented in SRNL-STI-2010-00047.

Carbon Steel Waste Tank Liners

As was the case for the FTF PA (SRS-REG-2007-00002, Rev. 1), DOE estimates steel tank liner failure times for the HTF PA using both deterministic and stochastic approaches. DOE uses the stochastic approach to account for potential uncertainty in the tank liner failure times used to demonstrate compliance with the performance objectives. In the end, a partial stochastic approach is used to define steel tank liner failure times for use in the deterministic and probabilistic HTF PA models. The partial stochastic approach fixes diffusion coefficients for the various cases while varying other uncertain parameters. The steel tank liner failure times for various tank types that are derived from the partial stochastic analysis for use in the deterministic and probabilistic HTF PA modeling are summarized in Table 4.2-32 of SRR-CWDA-2010-00128, Rev. 1. The deterministic HTF PA modeling uses the median failure times from the partial stochastic analysis, while the probabilistic modeling uses the distribution of failure times developed from the partial stochastic analysis.

Similar to the FTF PA, DOE continues to use information from scientific literature in the HTF PA to evaluate the potential effect of corrosion on the integrity of the carbon steel tank liners and to estimate the tank liner failure time (WSRC-STI-2007-00061, Rev. 2; SRNL-STI-2010-00047). A notable difference from the FTF PA involves the modeling of Type I and II tank liners in the HTF PA. DOE assumes that the primary and secondary liners for four tanks (i.e., Tanks 12, 14, 15, and 16) are failed initially. DOE indicates that the liners for these four tanks, therefore, do not impede advective or diffusive flow in the HTF PA at any time after closure.

DOE continues to assume a passive corrosion rate of 1 μm/yr (0.04 mil/yr) prior to carbonation- or chloride-induced steel depassivation, as it did for the FTF PA. Localized corrosion including pitting corrosion continues to be excluded by DOE in the HTF PA because DOE, as discussed in response to FTF RAIs (See response to FTF RAI-NF-2; SRR-CWDA-2011-00054, Rev. 1) believes that even if pitting were to occur, pit growth would decrease with time. DOE also indicates that its probabilistic analysis encompasses accelerated corrosion mechanisms such as pitting corrosion.

In DOE's HTF PA, as in the FTF PA, the entire liner is considered to have failed when the thinnest segment is penetrated by general corrosion. DOE uses this simultaneous liner surface failure model, instead of an alternative conceptualization such as gradual corrosion of the liner surface over time. This discrete failure of the entire steel liner tends to maximize peak doses. By contrast, a gradual degradation model would spread over time the releases of radionuclides (See response to FTF RAI-NF-2; SRR-CWDA-2011-00054, Rev. 1).

Similar to the modeling of FTF liners, DOE models the HTF liners, under grouted conditions, to fail due to either chloride depassivation or carbonation activation. In the case of chloride depassivation, the assumed corrosion rate is low and DOE computes that it would take thousands of years for through-wall liner corrosion to occur after the initiation of chloride depassivation. For activation by carbonation, DOE computes that through-wall corrosion would

occur in a few decades after carbonation activation. Both processes are assumed to affect the liner in parallel. The failure time is dictated by the fastest process causing through-wall penetration of the liner (see discussion in Section 4.2.9.2 of this TER).

Based on an empirical equation from technical literature (Clear, 1976), DOE continues to compute the chloride-induced corrosion initiation time as a function of the concrete thickness, water-to-cement ratio, and groundwater chloride concentration, as it did for FTF (WSRC-STI-2007-00460; SRNL-STI-2010-00047). After initiation, DOE assumes the corrosion propagation rate is controlled by oxygen diffusion through the concrete. DOE computes the chloride-enhanced corrosion rate as a function of the oxygen diffusion coefficient in concrete, oxygen concentration in groundwater, and concrete thickness. DOE postulates a minimum value of 1 μm/yr (0.04 mil/yr) for the chloride-enhanced corrosion rate for the deterministic corrosion analysis, based on typical values of corrosion rates of carbon steel under passive conditions.

Based on published carbonation analyses (Papadakis and Fardis, 1989; NUREG/CR-5542), DOE computes the carbonation initiation time as the time for a carbonation front to propagate through the concrete vault thickness. This approach is consistent with the approach used for the FTF PA. The penetration time is computed as a function of the carbon dioxide diffusion coefficient in concrete, the total inorganic carbon in groundwater or soil moisture, calcium hydroxide bulk concentration in concrete solid, and the concrete thickness. After activation by carbonation, the carbon steel is assumed to be in an active corrosion propagation state with a corrosion rate of 254 μm/yr (10 mil/yr).

Steel liner failure times for the HTF PA rely on results from the partial stochastic corrosion analysis in WSRC-STI-2007-00061, Rev. 2 (Type III, IIIA, and IV tanks) and in SRNL-STI-2010-00047 (Type I and II tanks) that consider discrete values of the oxygen and carbon dioxide diffusion coefficients. The analysis in SRNL-STI-2010-00047 is an extension of the steel degradation analyses for the FTF (WSRC-STI-2007-00061, Rev. 2). The extension accounts for the full or partial submersion of Type I and II tanks at HTF in groundwater. The main change in the analysis in SRNL-STI-2010-00047 from the previous FTF analyses is the consideration of galvanic and macrocell corrosion of the liner under partially submerged conditions. These corrosion processes are considered feasible for the partially submerged tanks because of the heterogeneous availability of oxygen for saturated and unsaturated soil conditions. DOE implements the galvanic corrosion concept by adopting a higher oxygen diffusion coefficient for immersion conditions (10^{-4} cm^2/s), compared to a diffusion coefficient for unsaturated conditions (10^{-6} cm^2/s). This approach is summarized in Table 4.2-32 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1), which shows that the assumed value of the oxygen diffusion coefficient for Type I and II tanks is 10^{-4} cm^2/s and 10^{-6} cm^2/s for Type III, IIIA, and IV tanks. It is understood, however, that such an approach is a simplification to simulate higher corrosion rates associated with galvanic corrosion. Physically, the effective diffusion coefficient for oxygen in concrete in saturated conditions is expected to be smaller than the oxygen diffusion coefficient in unsaturated conditions. DOE states that the results of the HTF PA are relatively independent of the value of the oxygen diffusion coefficient, because the dominant mode causing failure of the liner is the activation of corrosion by carbonation. DOE states in SRR-CWDA-2013-00106, Rev. 1 (Response to RAI-NF-2) that the use of the 10^{-4} cm^2/s diffusion coefficient for submerged conditions is an overestimation to also account for the potential formation of cracks in the concrete vault.

According to Table 4.2-32 in the HTF PA (SRR-CWDA-2010-00128, Rev. 1), DOE adopts general corrosion rates of the carbon steel activated by chloride in the range of 1 to 2 μm/yr (0.04 to 0.09 mil/yr) [Figure 40 and 41, $D(O_2)=10^{-4}$ cm^2/s; SRNL-STI-2010-00047] for Type I and II tanks and corrosion rates equal to 1 μm/yr (0.04 mil/yr) for Type III and IV tanks [Tables 28 and 29, $D(O_2)=10^{-4}$ cm^2/s, WSRC-STI-2007-00061, Rev. 2]. The magnitude of these corrosion rates corresponds to passive corrosion. As a comparison, a liner 0.5-in (1.3-cm) thick would corrode in 12,500 years at a rate of 1 μm/yr (0.04 mil/yr). Because of this slow rate, DOE concludes that the liner failure time is mostly controlled by the penetration time of a carbonation front , and that HTF PA results are almost independent of the magnitude of the oxygen diffusion coefficient, provided that it is less than 10^{-4} cm^2/s.

As previously stated, for Type I and II tanks the assumed value of the oxygen diffusion coefficient is 10^{-4} cm^2/s to determine liner failure times, independent of the configuration (i.e., Cases A, B, C, D, or E). For Type III/IIIA and IV tanks, DOE assumes a value of the oxygen diffusion equal to 10^{-6} cm^2/s to determine liner failure times for the HTF PA modeling, also independent of the configuration. The numerical distinction on the HTF PA results of selecting an oxygen diffusivity of 10^{-4} cm^2/s or 10^{-6} cm^2/s is negligible, as associated liner corrosion rates are computed by DOE to be in the passive range (due to the assumption of oxygen diffusion control of the corrosion rate). DOE establishes a distinction between the base case (Case A) and the alternative configurations (Cases B, C, D, E) by the use of a carbon dioxide diffusion coefficient of 10^{-6} cm^2/s for the former configuration, and 10^{-4} cm^2/s for the latter configurations, independent of the tank type. The numerical outcome of this distinction is that failure times are roughly one order of magnitude shorter for the configurations with fast flow paths (Cases B, C, D, and E) compared to the base case (Case A). Failure times are on the order of 1,000-3,000 years for Cases B, C, D, and E and on the order of 11,000–13,000 years for the base case (Case A), for all tank types but Type IV. Failure times for Type IV tanks are much shorter because of the thinner concrete vault and liner (3,638 yrs for Case A, and 75 years for the fast flow configurations).

Several uncertain parameters are considered in the partial and comprehensive stochastic analyses including tank and concrete thicknesses, diffusion coefficients for oxygen and carbon dioxide, 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 10^{-10} to 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.

Stainless Steel Transfer Lines

The reports WSRC-STI-2007-00460 and SRNL-STI-2010-00047 provide estimates for failure of the stainless steel transfer line core piping at HTF. Because less than ½ percent of the transfer lines in HTF are carbon steel, the HTF 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 as defined for the FTF PA in WSRC-STI-2007-00460. Both general corrosion and pitting corrosion are considered to be the degradation mechanisms. Data from technical literature (NBSIR 81-2228) show that both the general corrosion rate and pitting corrosion rate decrease with time. Based on published corrosion rate data, DOE uses 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 (BNL-71537-2003), the area breached by pitting corrosion is computed as 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 when pitting corrosion breaches 25 percent of the line wall surface. The earliest transfer line failures are estimated to occur 510 years after HTF closure [for transfer lines with a minimum wall thickness of 0.295 cm (0.116 in)]. The latest transfer line failures are estimated to occur 532 years after HTF closure [for the transfer lines with the minimum wall thickness of 0.480 cm (0.189 in)].

The document SRNL-STI-2010-00047 extends the FTF transfer line analysis to account for soils relevant to the HTF. Based on corrosion rates compiled by the National Bureau of Standards, DOE computed failure times longer than thousands of years to tens of thousands of years with a stochastic methodology, as summarized in Table 4.2-34 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1). These longer failure times are the result of slower corrosion rates associated with soils in the HTF and the use of a power law expression to compute the penetration depth as a function of time. The power law equation predicts corrosion rates that decrease with time, eventually falling below passive corrosion. The thousands of years lifetime estimates are more an indication of passive corrosion than localized corrosion. As a conservative approach, DOE adopts the failure time of 510 years, computed in WSRC-STI-2007-00460, for the transfer lines in the HTF PA.

4.2.8.3 Source-Term Release

As discussed in Section 4.2.8, the ICM assumes that the residual waste within the primary liner 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. Radionuclide leaching from the CZ depends on the flow and chemical composition of the pore fluid passing through the zone. Infiltration from the surface (for all tanks) or groundwater (for fully and partially submerged tanks) that passes through the waste tanks interacts with the grout, which changes the grout mineralogy and causes fluids emerging from the grout that flow into the CZ to have a composition that reflects these interactions. Contaminant release from the CZ is controlled primarily by the solubility of assumed contaminant-bearing solid phases in the varying fluid composition. As the grout-filled waste tanks “age,” the pore fluid chemistry also changes, which affects contaminant release

from the CZ. For the HTF waste tanks, there is also contamination located within the annuli and sand pads of the Type I and II tanks. The annulus and sand pad waste is outside of primary containment and assumed to be soluble. In the Type I tanks, DOE models the annulus contamination as a thin layer within the bottom of the reducing annulus grout. In Type II tanks, DOE models the annulus contamination as contained within the sand pads. In the modeling for both tank types, the release of residual waste from the annulus is controlled by sorption because DOE assumes the material is soluble Transport of the contaminants from the Type I tank annuli and Type II tank sand pads is discussed in Section 4.2.8.4.

To account for the evolution of pore fluid chemistry and its effect on radionuclide release from the CZ, DOE uses a conceptual model that considers four chemical states of the grout as it ages: (1) Reduced Region II, (2) Reduced Region III, (3) Oxidized Region II, and (4) Oxidized 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) (i.e., reducing and oxidizing). 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 by DOE to exhibit chemical conditions represented by Reduced Region II as a result of the addition of blast furnace slag to the grout. DOE assumes that because the waste tank grout and concrete will be aged at the time of HTF closure, none of the cementitious materials would be characterized as Region I. 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 and carbon dioxide. The geochemical model simulates the reaction of infiltrating water with grout components that include calcium-silicate-hydrate phases (jennite, tobermorite), hydrotalcite, gypsum, gibbsite, pyrite, and hematite.

Geochemical modeling results indicate that the grout pore water E_h rises from a reducing −0.47 volts, poised by the presence of pyrite, to a value about +0.56 volts at 523 pore volumes when pyrite is completely oxidized, corresponding to the transition from Reduced Region II to Oxidized Region II. The grout pore water pH initially is 11.6, controlled by the mineral jennite, then decreases to 11.1 at 67 pore volumes when jennite is converted to tobermorite. At 2,119 pore volumes, tobermorite is consumed and the pH decreases to 9.2, at which point the E_h also increases to +0.68 volts, poised by equilibrium with dissolved oxygen, corresponding to the transition from Oxidized Region II to Oxidized Region III. Thus, in the ICM, the waste tank pore water chemistry changes from an initial Reduced Region II chemical state to (1) Oxidized Region II conditions after 523 pore volumes have passed through the tank grout, and (2) Oxidized Region III after 2,119 pore volumes have passed through the tank grout. Because the E_h rises to oxidizing conditions prior to the pH decreasing to a value of 9.2, Reduced Region III is bypassed.

For submerged and partially submerged waste tanks, DOE models the evolution of pore fluid chemistry assuming four different chemical conditions (Conditions A, B, C, and D) that reflect varying degrees of groundwater influence and consequently, varying chemical conditions. Condition A assumes that groundwater flows laterally, directly into the CZ with no effect from the concrete vault. Condition B assumes that groundwater equilibrates with the concrete vault before passing through the CZ where it mixes with the small amount of Oxidized Region II grout pore fluid. Condition C assumes that groundwater flows laterally, directly into the CZ with no effect from the concrete vault and mixes with a small amount of Reducing Region II grout pore

fluid. Condition D assumes that groundwater flows directly into the residual waste layer with no effect from the concrete vault and mixes with a small amount of Oxidizing Region II grout pore fluid. DOE stated that the most probable progression of chemical conditions in submerged waste tanks would be from Condition C to Condition D to Oxidized Region III. DOE uses Geochemist's Workbench® to simulate the mixing of groundwater with grout pore fluid, and selects the composition at 90 percent groundwater and 10 percent pore fluid for Conditions C, and D. The mixture compositions are used as input to Geochemist's Workbench® modeling of grout degradation—similar to that conducted for the non-submerged waste tanks—to determine the change in pore water composition (and associated radionuclide solubilities) over time, as the water reacts with the reducing grout. The grout pore water E_h transitions from −0.47 volts to +0.54 volts after 1,826 pore volumes of fluid reacted (i.e., Condition C to Condition D) and the pH transitions from 11.3 to 9.3 at 2,445 pore volumes (i.e., Condition D to Oxidized Region III).

In the ICM, release of radionuclides from the CZ is either instantaneous or is controlled by imposing a concentration limit (i.e., solubility limit). As shown in Table 4.2-11 from the HTF PA (SRR-CWDA-2010-00128, Rev. 1), instantaneous release is assumed for several elements: berkelium, carbon (Condition A), californium, cesium, iodine, niobium, protactinium, rhodium, selenium (under some conditions), and tellurium. 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 the solubility of a pure solid phase or on co-precipitation of the radionuclide as a trace element in an iron oxide solid (SRNL-STI-2012-00404, Rev. 0). The assumed phases and solubilities for several key radionuclides under the varying chemical conditions is provided in Table 4-4.

When the concentration isbased on the solubility of a pure solid phase, DOE states that the solubility controlling solid is 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, amorphous phases and hydroxides may be chosen over crystalline phases and oxides, respectively, because amorphous phases and hydroxides 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 iron oxide (SRNL-STI-2012-00404, Rev. 0). With this approach, the concentration ratio between the element and iron 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 (Fe_3O_4; reduced regions) and maghemite (Fe_2O_3; oxidized regions) in the particular chemical state and uses the resulting dissolved iron concentration and the solid phase ratio to calculate the dissolved concentration of the radionuclide. Solubility limits for technetium, uranium, neptunium, and plutonium based on iron co-precipitation are shown in Table 4.2-14 of SRR-CWDA-2010-00128, Rev. 1. Although Table 4.2-11 in SRR-CWDA-2010-00128, Rev. 1, does show either pure-phase or no solubility control for technetium, a footnote to the table indicates that iron co-precipitation is used for technetium in deterministic models. For uranium, neptunium, and plutonium, the iron co-precipitation model is not used in deterministic models, but it is used to calculate solubility limits for establishing probability distributions and also used in sensitivity analyses (Sections 5.6.3.3 and 5.6.7.3; SRR-CWDA-2010-00128, Rev. 1). The solubility limit for the iron co-precipitation phases of technetium, uranium, neptunium, and plutonium are lower than the pure-phase limit.

Table 4-4 Phases and Solubilities of Key Radionuclides Modeled in HTF Performance Assessment*

	Non-submerged Tanks					
	Reduced Region II		Oxidized Region II		Oxidized Region III	
	Phase	mol/L	Phase	mol/L	Phase	mol/L
Np	NpO_2 (am,hyd) †	1.0×10^{-09}	NpO_2 (am,hyd) †	3.0×10^{-07}	NpO_2 (am,hyd) †	2.0×10^{-06}
Pu	PuO_2 (am,hyd) †	3.0×10^{-11}	PuO_2 (am,hyd)	3.0×10^{-11}	PuO_2 (am,hyd) †	3.0×10^{-11}
Tc	Fe co-ppt‡,§	1.0×10^{-14}	Fe co-ppt‡,§	1.0×10^{-13}	Fe co-ppt‡,§	2.0×10^{-15}
U	UO_2 (am,hyd) †	5.0×10^{-09}	$UO_3\cdot 2H_2O$	5.0×10^{-05}	$UO_3\cdot 2H_2O$	4.0×10^{-06}

	Submerged Tanks						
	Condition A		Condition B	Condition C		Condition D	
	Phase	mol/L		Phase	mol/L	Phase	mol/L
Np	NpO_2 (am,hyd) †	3.0×10^{-05}	N/A	NpO_2 (am,hyd) †	1.0×10^{-09}	NpO_2 (am,hyd) †	2.0×10^{-05}
Pu	PuO_2 (am,hyd) †	2.0×10^{-10}	N/A	PuO_2 (am,hyd) †	3.0×10^{-11}	PuO_2 (am,hyd) †	3.0×10^{-11}
Tc	Fe co-ppt‡,§	1.0×10^{-12}	N/A	Fe co-ppt‡,§	3.0×10^{-12}	Fe co-ppt‡,§	1.0×10^{-15}
U	$UO_3\cdot 2H_2O$	4.0×10^{-05}	N/A	UO_2 (am,hyd) †	4.0×10^{-09}	$UO_3\cdot 2H_2O$	2.0×10^{-06}

* Table 4.2-11 in SRR-CWDA-2010-00128, Rev. 1 unless otherwise attributed.
† (amorphous, hydroxide)
‡ Iron co-precipitation (Fe co-ppt)
§ Table 4.2-14 in SRR-CWDA-2010-00128, Rev. 1.

4.2.8.4 Near-Field Flow and Transport Modeling

DOE models the near-field flow and transport for each HTF waste source individually within PORFLOW™ version 6.30.2 to determine the source-specific, time-dependent radionuclide contaminant flux entering the UTRA-UZ. 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 individual waste tanks and near-field transport in PORFLOW™ within a two-dimensional, axisymmetric radial wedge. Using symmetry, DOE places no-flow and no-flux boundary conditions at the waste source centerlines and at the outer radius of the model domains, which extend 9 m (30 ft) beyond the perimeter of each waste tank. The near-field PORFLOW™ model domains use 5,000 to 7,000 cells to represent each of the different types of waste sources.

For non-submerged tanks, vadose zone thickness between the concrete slabs (Tank Types III/IIIA) or basemats (Tank Type IV) underlying the tanks and the water table ranges from 2.2 to 5.5 m (7.1 to 18.2 ft); however, this thickness is modeled as a constant for each tank type, based on the average vadose zone thickness beneath a tank group. 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 PORFLOW™ model; zero concentration flux is also prescribed at the upper boundary. For waste tanks that are entirely above the water table, 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.

For partially or fully submerged waste tanks (Tank Types I and II), DOE relies on a cross-flow factor to account for lateral flow because the saturated zone is not represented in the near-field model. The cross-flow factor is applied to submerged materials to calculate the number of pore volumes that, in turn, determines the timing of the chemical degradation transitions. A ratio of 10 to 1 is assumed to represent the ratio of the total cross flow through the material components to the downward flow due to infiltration. The contaminant fluxes from the partially or fully submerged tanks are loaded into the far-field model source elements at the elevation of the tank basemats (see Sections 4.2.10.2 and 4.2.11.1).

Contaminated transfer lines in the HTF area are widely distributed over a broader region than that represented by the waste tank clusters alone (see Figure 4-3). DOE divides process piping into four polygonal regions for separate modeling runs based on common vertical distances between the equipment and the underlying mean water table (SRNL-STI-2012-00465). DOE models this inventory as broadly distributed sources and the 100-m (330-ft) compliance point is generally measured from the edges of these four polygons (Shaffner, 2013b [ML13126A127]). Additionally, DOE models eleven pump tanks and three evaporator pots as point sources. The associated inventory was assumed to be in direct contact with the soil after the assumed failure time of 510 years (SRR-CWDA-2010-00128, Rev. 1).

Near-Field and Vadose Zone Transport Material Properties

Compacted excavated soil backfill was emplaced surrounding, as well as a minimum of 2.7 m (9 ft) above Type I tanks. Soil backfill emplaced around Type II tanks was brought level with the top of the tanks and extended laterally for a minimum of 6.4 m (21 ft). Compacted excavated soil backfill was emplaced around Type III and IIIA tanks to within 0.3 m (1 ft) of the elevation of the top of the tanks. Type IV tanks were immediately surrounded with vermiculite bags prior to backfilling with compacted excavated soil to a minimum height of 1.1 m (3.7 ft) above the top of the tanks. DOE neglects the material properties of vermiculite in the near-field model because it considers the vermiculite volume to be insignificant (WSRC-STI-2006-00198, Rev. 0). 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 derives material property model parameters for compacted excavated soil backfill based on laboratory measurements of sandy-clay–dominated compacted backfill from the Old Radioactive Waste Burial Ground in E-Area (WSRC-STI-2006-00198, Rev. 0). Backfill material properties are listed in Table 4-5. DOE assumes a saturated effective diffusion coefficient based upon non-GSA literature values

Table 4-5 Material Properties used in PORFLOW™ Near-Field Modeling, Including Those of Intact Cementitious Materials*

Vadose Zone Material	Saturated Effective Diffusion Coefficient ($cm^2/s^{\dagger}$)	Average Total Porosity (%)	Average Dry Bulk Density ($g/cc^{\ddagger}$)	Average Particle Density ($g/cc^{\ddagger}$)	Saturated Horizontal Hydraulic Conductivity ($cm/s^{\S}$)	Saturated Vertical Hydraulic Conductivity ($cm/s^{\S}$)
Concrete (Vaults/ Basemat/ /Roof)	8.0×10^{-7}	16.8□	2.06□	2.51□	3.5×10^{-8}	
Grout Fill LP#8-016 (Annuli/Tanks)	5.0×10^{-8}	21¶	1.97¶	2.49¶	2.1×10^{-9}¶	
Gravelly Fast Flow Path	9.4×10^{-6}§§	30§§	1.82§§	2.60§§	1.5×10^{-1}§§	
Sand Pad	8.0×10^{-6}	38	1.65	2.66	5.0×0^{-4}	2.8×10^{-4}
Backfill	5.3×10^{-6}	35	1.71	2.63	7.6×0^{-5}	4.1×10^{-5}
Upper Vadose Zone	5.3×10^{-6}	39	1.65	2.70	6.2×0^{-5}	8.7×10^{-6}

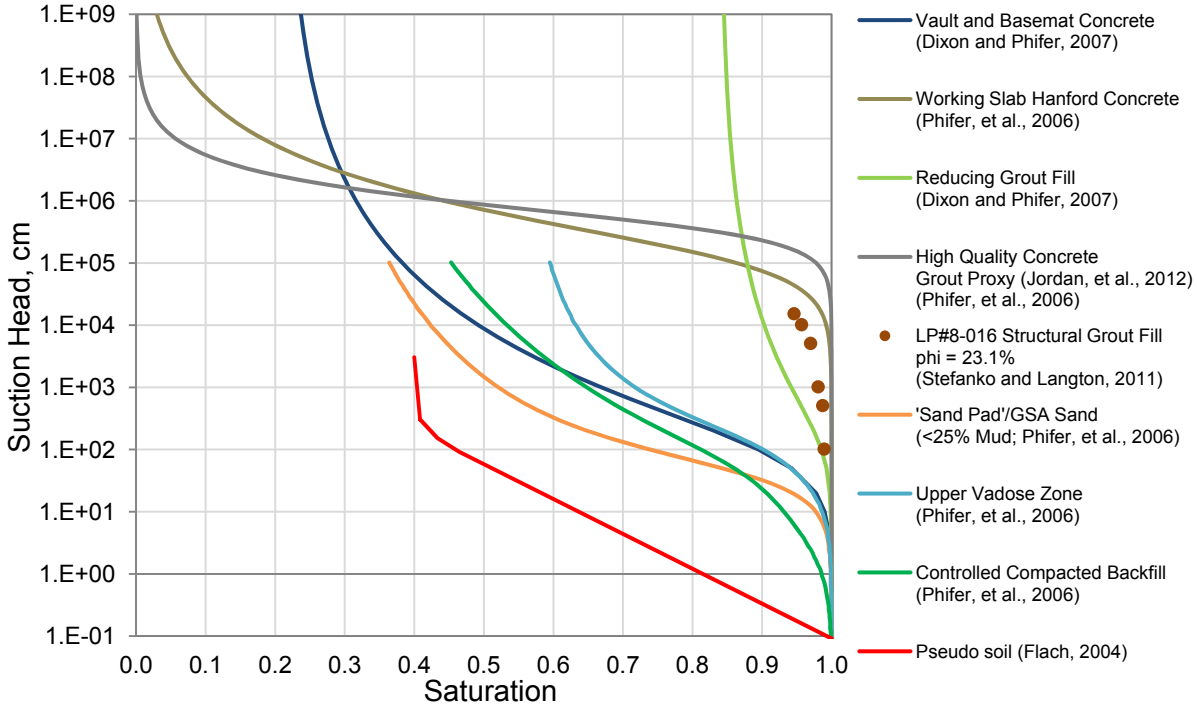
* WSRC-STI-2006-00198, Rev. 0 unless otherwise attributed.
† To convert cm^2/s to in^2/s, mcultiply by 0.16.
‡ To convert g/cc to lb/ft^3, multiply by 62.4.
§ To convert cm/s to ft/s, multiply by 3.3×10^{-2}.
□ WSRC-STI-2007-00369, Rev. 0
¶ SRNL–STI–2011–00551
§§ SRR-CWDA-2010-00128, Rev. 1

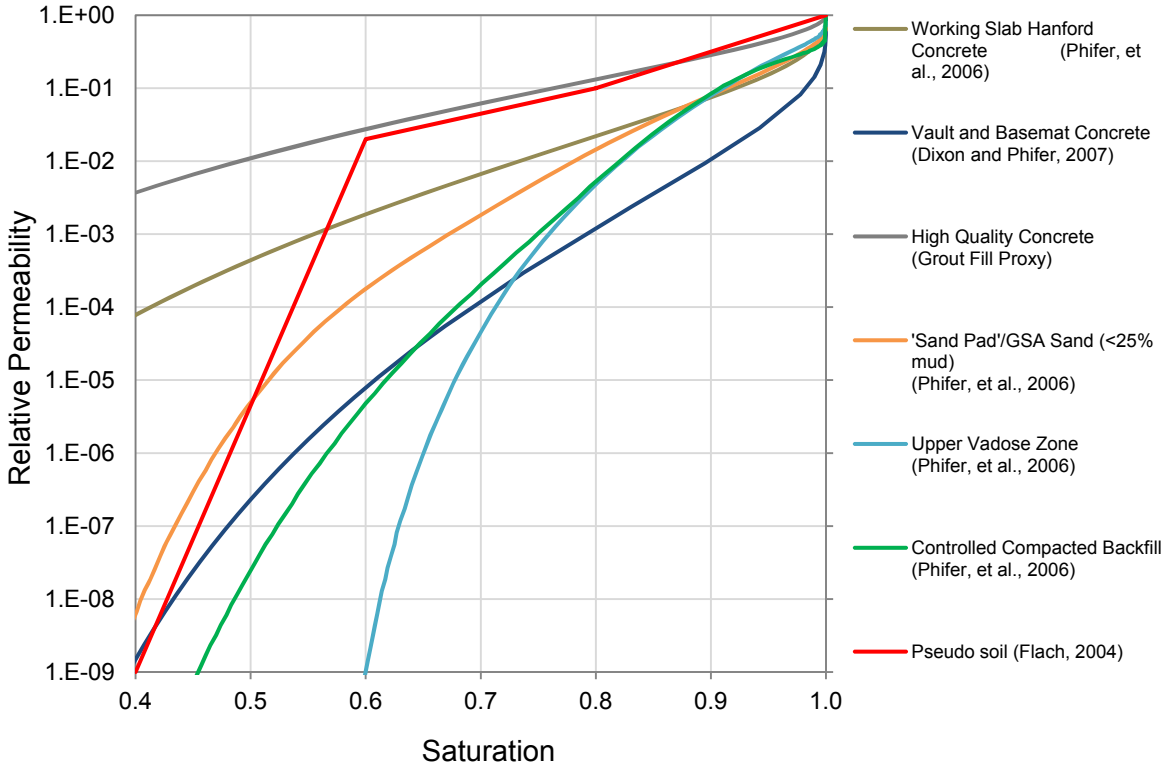
for molecular diffusion coefficients and tortuosity (WSRC-STI-2006-00198, Rev. 0) and neglects hydrodynamic dispersion. DOE assumes backfill material properties do not change with time.

DOE derives model parameters that represent undisturbed vadose zone material properties below the tanks using data from the E-Area Low-Level Waste Facility (WSRC-STI-2006-00198, Rev. 0), which is located approximately 900 m (3,000 ft) northwest of the HTF. E-Area subsurface materials are assumed to be similar to H-Area materials in geology and physiography, although relatively few borings were available in H-Area to confirm this assumption. Undisturbed vadose zone material beneath the H-Area waste tanks is silty- to clayey-sand in the upper vadose zone (SRR-CWDA-2010-00128, Rev. 1). The lower vadose zone material identified in E-Area occurs below the water table in H-Area (Shaffner, 2013b [ML13126A127]).

Estimated material properties for the E-Area vadose zone soils are given in Table 4-5 and Figure 4-9. K_{sat} values reported in Table 4-5 are increased beyond initial estimates by 2σ so that an E-Area vadose zone model would adequately agree with field saturation and suction data (WSRC-STI-2006-00198, Rev. 0). DOE assumes the same saturated effective diffusion coefficient for vadose zone soils that is used for backfill, and neglects hydrodynamic dispersion. DOE assumes vadose zone material properties do not change with time.

Figure 4-9 Characteristic Curves Used in PORFLOW™ Near-Field Modeling, Including Those of Intact Cementitious Materials





The materials that the steel liners sit atop vary somewhat by tank type. Figures 1-1 through 1-5 depict the construction of the various tank types and their concrete basemats. Summaries of the construction of the tank foundations beneath the tank steel liners are discussed by each tank type in the following bullets:

- Type I tanks, which have a 76-cm (30-in) reinforced-concrete basemat, were constructed atop a 10-cm (4-in) thick working slab. Layers of construction grout separate the primary liner from the secondary liner and the secondary liner from the basemat.

- Type II tanks, which have a 1.1-m (3.5-ft) thick reinforced concrete basemat, were constructed atop a single 15-cm (6-in) thick working slab. Layers of 2.5-cm (1-in) thick commercially procured sand separate the primary liner from the secondary liner and the secondary liner from the basemat.Type III tanks were constructed atop a single 15-cm (6-in) thick working slab. The tanks have a 1.1-m (3.5-ft) thick reinforced concrete basemat, except at the drop panel in the center of the waste tank where the thickness is 1.6 m (5.3 ft).

- Type IIIA tanks are grouped into three separate clusters. Each cluster was constructed above a 10-cm (4-in) thick working slab, which was either broken up or punched through with 10-cm (4-in) diameter holes on 45.7-cm (18-in) centers to prevent water from perching above them. The Type IIIA basemat thickness is 1.1 m (3.6 ft) of reinforced concrete, except at the drop panel in the center of the waste tank where its thickness is1.9 m (6.3 ft).

- Type IV tanks, which have a 10-cm (4-in) reinforced concrete basemat, do not rest on a working slab; however, there is an 8-cm (3-in) thick drainage and maintenance slab between these tanks.

Because working and maintenance slabs were not constructed with reinforcing steel, DOE assumes that heavy working equipment, tank construction, and tank filling activities cracked the slabs such that they will not function to laterally divert water in the near-field.

The Type II tanks were built with sand pads to provide a level surface atop the basemat and support for steel liners. Type II tanks contain two sand pads: (1) a pad between the basemat and secondary liner (referred to as the secondary sand pad) and (2) a pad between the secondary and primary liners (referred to as the primary sand pad). The assigned material properties for the commercially procured sand of these sand pads are given in Table 4-5 and Figure 4-9. DOE assumes that the procured sand has the same moisture retention properties as that of naturally occurring GSA sand. However, the procured sand has been cleaned, whereas the GSA sand contains fine-grained particles. Accordingly, the MCCs of these two sands would likely differ; however, DOE indicates that this assumption has no impact on HTF PA results because the sand pads are located below the water table and modeled as fully saturated (Shaffner, 2013a [ML13106A338]). Although the near-field model does not include the saturated zone, DOE assumes that the relative permeability is equal to one at all saturation levels for the primary and secondary sand pad layers (Shaffner, 2013a [ML13106A338]). Therefore, the MCCs for the sand pad layers have no effect on the HTF PA results.

As radionuclides are released from the CZ within the primary liner, they migrate by advection or diffusion through a cementitious material—either tank grout or basemat concrete. In addition to the tank inventories contained within the primary liners, there is contamination located outside of the primary containment in the Type I and Type II tanks (see Sections 3.1.2 and 3.2.2 of this TER). For Type I tanks, DOE models the annular inventory as a thin layer within the bottom of the annulus grout. For Type II tanks, DOE models the inventory within the sand pads beneath the primary tank. Transport of contaminants from the annulus of Type I tanks and sand pads of Type II tanks upwards into the overlying annular grout and vault concrete through diffusion or advection is controlled by grout K_ds; no solubility control is assumed for cementitious materials.

Cementitious material K_ds are used in the ICM for radionuclide transport out of the tank vaults. Table 4.2-29 in DOE's HTF PA (SRR-CWDA-2010-00128, Rev. 1) shows the selected K_d values for the various pore water chemical states reflecting the age of the cementitious materials. Selection of most cementitious K_ds is documented in SRNL-STI-2009-00473, Rev. 0. For purposes of assigning K_d 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_d values shown in Table 4.2-25 of DOE's HTF PA (SRR-CWDA-2010-00128, 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. 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_d discussion in the HTF PA, therefore, applies to the saturated zone K_d discussion.

The sediment K_d values are selected, as documented in several reports cited in Table 4.2-25 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1), on the basis of site-specific data where possible, supplemented by analyses of data from technical literature. Selection of most K_ds is documented in SRNL-STI-2009-00473, Rev. 0. The technetium K_ds for sandy and clayey soils are based on the medians of 47 measurements of local samples (Table 1; SRNL-TR-2009-00019, Rev. 0). Relative to values used for the FTF PA, neptunium K_ds are updated in the HTF PA using new site-specific data (SRNL-STI-2009-00634, Rev. 0). A substantial volume of site-specific data exists for plutonium. For all elements in the deterministic analysis, DOE chooses what are termed the "best" values, rather than conservative values.

For the HTF PA, DOE uses K_ds for "cement leachate impacted soils media" (Table 4.2-25; SRR-CWDA-2010-00128, Rev. 1). The adjusted K_ds are intended to reflect the effect on radionuclide sorption when the chemistry of the transporting water is affected by leachate from the tank grout. The adjusted K_ds are used for vadose zone transport in sediments until the contamination zone has reached Oxidized Region III conditions. The K_d adjustment uses a factor calculated from data from the Hanford site and the approach is considered provisional until site-specific data are obtained (Sections 4.2.4 and 4.2.5; SRNL-STI-2009-00473, Rev. 0).

DOE uses log-normal distributions with minimum and maximum values and geometric standard deviations based on a factor of the geometric mean in its probabilistic assessment for (1) cementitious materials, (2) vadose zone, and (3) saturated zone sorption coefficients (Section 5.6.3.4, Table 5.6-13; SRR-CWDA-2010-00128, Rev. 1). The parameter distributions are calculated from a statistical analysis of samples from a single E-Area core. In that analysis (WSRC-STI-2008-00285), K_ds were measured for several radionuclides in 27 samples that represented the upper vadose, lower vadose, and aquifer zones in the E-Area.

Near-Field and Vadose Zone Transport Model Validation

DOE uses characterization and monitoring data collected from E-Area under normal infiltration (uncapped) conditions to validate aspects of PORFLOW™ near-field models. These data included soil suction and water content, tracer test pore velocity, and tritium plume concentrations. E-Area soil suction data range from 50 to 200 cm (20 to 80 in), while 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) (SRR-CWDA-2010-00128, Rev. 1). E-Area soil suctions provide the technical basis supporting the use of a low-suction range in the HTF PA modeling. Such results are not attained directly from the best estimate of K_{sat}; rather, because of excessive water hold-up in the original simulations, K_{sat} is increased to the high end of its uncertainty range (+2σ) to obtain an acceptable match of model output to field data (WSRC-STI-2006-00198, Rev. 0). Water content field data suggest saturation ranges from 35 to 75 percent, while PORFLOW™ vadose zone modeling produces upper and lower vadose zone saturation values of 91 and 72 percent (SRR-CWDA-2010-00128, Rev. 1). Field and laboratory tracer test experiments indicate a pore velocity of 114 cm/yr (45 in/yr), while PORFLOW™ vadose zone modeling produces upper and lower vadose zone pore velocities of 86 cm/yr (34 in/yr) and 109 cm/yr (43 in/yr) (SRR-CWDA-2010-00128, Rev. 1).

Near-Field Flow and Transport Data and Model Uncertainty

DOE considers uncertainty in the near-field flow field through the assignment of velocity profiles for five modeling cases, using eight near-field tank models (i.e., Tank Type I, Tank Type I–no liner, Tank Type II, Tank Type II–no liner, Tank Type III, Tank Type IIIA North/South, Tank Type IIIA West, and Tank Type IV) and 72 parametric flow scenarios (see Section 4.2.18.1 of this TER). Tanks IIIA West are modeled separately from Tanks IIIA (North and South) because of differences in vadose zone thickness beneath the tanks (SRNL-STI-2012-00465). The 72 flow scenarios consist of three fast flow cases (partial fast flow path, full fast flow path, and no fast flow path), four liner failure times (time zero, early, moderate, or late), three cementitious material degradation rates (fast, nominal, or slow), and two infiltration cases (capped or background rates). Velocity profiles are abstracted from the two-dimensional PORFLOW™ near-field models for use in the GoldSim™ probabilistic model for individual waste tank cases and tank types.

DOE considers the possibility for fast flow through preferential pathways versus flow through the grout matrix through alternative scenarios. The fast flow pathways are represented in Cases B-E (see Section 4.2.18.1 of this TER). DOE assumes that the fast flow pathways develop in the grout prior to bulk grout degradation, allowing flow through the tank to be dominated by these pathways. The fast flow pathways provide a means to bypass the chemical conditioning of the bulk grout, allow earlier entry into the CZ, and allow accelerated release of HRRs compared to

the base case (Case A) in which flow is assumed to occur through the grout matrix. In Cases B–E, flow through the system is governed by underlying assumptions regarding liner integrity and fast flow path continuity (modeled by a discrete high conductivity "gravelly" column and row of cells in PORFLOW™; Table 4-5), cementitious material degradation, and infiltration rate. The appropriateness of this fast flow path modeling treatment is discussed in Section 4.2.9.1. In addition to uncertainty in the flow field, there is also uncertainty associated with flow and transport parameters assigned to materials in DOE's near-field model that are a function of radionuclides (i.e., K_ds) and the nature and extent of grout degradation (i.e., diffusivities, hydraulic conductivities, and characteristic curves).

4.2.9 NRC Evaluation of Release and Near-Field Transport

4.2.9.1 NRC Evaluation of Cementitious Materials Degradation

The NRC staff reviewed DOE's assessment of cementitious material degradation as described in Section 4.2.2.2.4 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1) and in supporting analyses presented in SRNL-STI-2010-00035, Rev. 0; SRR-CWDA-2010-00019, Rev. 0; and SRNL-STI-2012-00465. Based on information in technical literature on cement and concrete degradation, the NRC staff finds that DOE's assessment has appropriately considered the two important cementitious material chemical degradation mechanisms. DOE identifies carbonation as the dominant degradation mechanism for grout in Types I, II, III, and IIIA tanks because of enhanced corrosion of the steel cooling coils present in these tanks, whereas it identifies leaching as the major degradation mechanism for grout in Tank IV tanks and the annuli of Type I, II, III, and IIIA tanks, which have no cooling coils. Carbonation-induced corrosion of reinforcement steel is identified as the dominant degradation mechanism for the concrete vaults in all tank types. Mechanical degradation mechanisms (e.g., differential settlement, shrinkage, thermal loading) could result in additional cracking; however, the effects of additional degradation mechanisms should be bounded by the consideration of bypassing fast pathways.

With respect to the hydraulic conductivity of cementitious materials in the HTF PA, DOE assumes a non-linear change in hydraulic conductivity rather than assuming a linear increase in hydraulic conductivity from intact to fully degraded conditions, as DOE assumed in the FTF PA. To compute the hydraulic properties of degraded concrete and grout in the HTF PA, DOE blends intact matrix properties with the hydraulic properties of fractures, which are assumed to develop over time (SRNL-STI-2012-00465). Note that fast pathways are treated independently from this assumed hydraulic degradation due to fracturing, as discussed in Sections 4.2.8.4 and 4.2.9.4 of this TER. DOE then relies, in part, on a series of MCCs designed to represent various stages of concrete and grout degradation (SRNL-STI-2012-00465). Depending on the level of saturation in the model for the concrete and grout (i.e., suction head), the hydraulic conductivity is assumed to vary by several orders of magnitude. Figure 9 in SRNL-STI-2012-00465 shows that if the suction head is greater than approximately 100 cm (40 in) then the assumed hydraulic conductivity for the "degraded" concrete and grout will be equivalent to the intact cementitious materials.

The NRC staff's review of the PORFLOW model files indicates that many of the cementitious materials for the various tank types are fully saturated throughout the modeled performance period for Cases A-E. The model files do indicate that there are instances when the saturation levels for cementitious materials are as low as 99.3 percent. Because these cementitious

materials are almost completely saturated, the NRC staff would not expect the relative permeability to differ from that of fully saturated cementitious materials. However, DOE's assumed MCCs can still result in a significant reduction in flow. The underlying MCC used to represent the cementitious materials in SRNL-STI-2012-00465 is assumed to be that of a high quality concrete from WSRC-STI-2006-00198, Rev. 0. Figure 4-9 illustrates that at a saturation of 99.3 percent, the high quality concrete curve leads to a suction head of greater than 10^5 cm (10^4 in). A suction head of this magnitude can be seen in Figure 9 of SRNL-STI-2012-00465 to result in the degraded curves coinciding with the curve for intact grout. In other words, the effective hydraulic conductivity of the grout that DOE assumes to be degraded would not differ from the hydraulic conductivity of intact grout, at any saturation levels less than 100 percent. The NRC staff questions the reasonableness of this assumed unsaturated flow behavior through degraded concrete and grout; however, it does not appear to have a significant impact on the flow results in the HTF PA. First, the PORFLOW model files show that a large majority of the cementitious materials are 100 percent saturated for all of the cases, tank types, and time periods. Second, for the cementitious materials that are not fully saturated, the PORFLOW model files do not indicate a significant reduction in flow, as would be expected by the assumed MCCs. This unexpected result may be a model artifact due to convergence of the numerical model. Regardless, the assumed MCCs for degraded cementitious materials do not appear to have a significant effect on the flow results. However, depending on future modeling assumptions, this approach to flow through degraded cementitious materials could result in a significant reduction in flow and may, therefore, require more model support.

The saturation in the fast flow paths, which are represented in alternative Cases B-E through the tank grout and vault concrete, is modeled as being much lower than through the intact cementitious materials. The PORFLOW model files show that the fast flow paths are often 30–50 percent saturated. The HTF PA does not provide the assumed MCC for gravel; however, the PORFLOW model files show what appears to be a reasonable amount of water flowing through the fast flow paths. Depending on future modeling assumptions, the use of MCCs may be more risk significant and may require more model support.

DOE assumes, as they did in the FTF PA, that the timing of cementitious material degradation is controlled by the diffusive transport of deleterious species (SRNL-STI-2010-00035, Rev. 0; SRR-CWDA-2010-00019, Rev. 0). Degradation calculations for the base case (Case A) suggest relatively long lifetimes for the cementitious materials. For example, the concrete vaults in Type I, II, III, IIIA, and IV tanks do not start degrading until 1,350, 2,550, 2,550, 2,500, and 400 yrs, respectively. As the NRC staff discusses in Section 4.2.9.1 of the FTF TER (Camper, 2011 [ML112371751]), Appendix D of the FTF Monitoring Plan (Camper, 2013a [ML12345A322]), and in the HTF RAIs (CC-NF-2; Mohseni, 2013a [ML13196A135]), these long lifetimes may be overly optimistic based on available information from existing concrete vaults at the SRS. An evaluation of the FTF waste release by a peer review panel states that the slow carbonation of cement-based material by diffusion of carbon dioxide seems like a low probability scenario; preferential flow through cracks would appear to be a more likely scenario (LA-UR-12-00079). Furthermore, DOE likely overestimates the initiation time of concrete vault degradation by carbonation and rebar corrosion. DOE assumes in its degradation analysis that concrete degradation starts once the carbonation effect reaches 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. Earlier concrete vault degradation

times will result in earlier initiation times for tank grout degradation. As DOE discusses in response to RAI-NF-2 in SRR-CWDA-2013-00106, Rev. 1, DOE believes that the assumed diffusion coefficient captures the effects of increased permeability due to cracking, independent of the rebar location (see Section 4.2.9.2 of this TER for additional discussion). However, based on its risk significance, the NRC staff will continue to review additional information used to support DOE's assumptions regarding Concrete Vault Performance as discussed in Monitoring Factor 3.1 of the FTF Monitoring Plan (Camper, 2013a [ML12345A322]). Additional discussion related to the FTF Monitoring Plan, Monitoring Area 3—Cementitious Material Performance is provided in Sections 4.2.9.2, 4.2.9.3 and 4.2.9.4 of this TER.

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, II, 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, II, III, and IIIA tanks. Section 4.2.9.2 discusses the NRC staff's concerns about the effect of uncertainties in cementitious material degradation on steel liner performance.

4.2.9.2 NRC Evaluation of Steel Degradation

The NRC staff reviewed DOE's assessment of carbon steel tank liner and stainless steel transfer line failure, as described in the HTF PA (SRR-CWDA-2010-00128, Rev. 1), Sections 4.2.2.2.6, and 4.4.2, and in supporting analyses presented in WSRC-STI-2007-00061, Rev. 2, WSRC-STI-2007-00460, and SRNL-STI-2010-00047.

NRC Evaluation of Carbon Steel Waste Tank Liner Failure

First, because chemical species that induce corrosion, such as water, chloride, oxygen, and carbon dioxide, need to be transported through the cementitious materials, the NRC staff finds, as it did in the FTF TER (Camper, 2011 [ML112371751]) that the uncertainty in carbon steel tank liner longevity is related primarily to the integrity and hydraulic properties of the cementitious materials in contact with the steel liner. The long estimated lifetime of the liner depends upon the concrete vault to be a long lasting diffusive barrier, controlling the transport of species such as oxygen and carbon dioxide.

The document SRNL-STI-2010-00035, Rev. 0 discusses chemical degradation of cementitious materials and indicates significant penetration (20.8 cm [8.2 in]) of a carbonation front in 1,000 years for unsaturated conditions (Table 3-3 in SRNL-STI-2010-00035, Rev. 0). Such analysis may apply to Type II, III, IIIA, and IV tanks whose liners are exposed to unsaturated conditions over their entirety or over some portion of the liner. Penetration of carbon dioxide by diffusion is linked to carbonation of the concrete and depassivation of the rebar. Corrosion of the rebar leads to formation of corrosion products. The expansion in volume of the rebar due to the formation of corrosion products can crack reinforced concrete and cause spalling. DOE postulates that failure of the reinforced concrete vault occurs when the carbonation front penetrates 50 percent of the wall thickness. As discussed in the FTF TER (see Section 4.2.9.1; Camper, 2011 [ML112371751]), the technical basis for this failure criterion is not clear, especially since failure criteria in technical literature are more commonly defined on the basis of

the thickness of the rebar cover (Parrot, 1990; Andrade et al., 1993; Molina et al., 1993; Neville, 1996). Gradual cracking of the reinforced concrete vault may cause feedback on the rate of transport of carbon dioxide causing a local lowering of the porewater pH, activation of the rebar, and eventually activation of the liner. Also, as discussed in Section 4.2.9.2 of the FTF TER, DOE continues to rely on diffusion coefficients through cementitious materials that differ between the cementitious materials degradation analysis and steel liner corrosion analysis. Thorough integration of the chemical degradation of cementitious materials analysis with the liner corrosion analysis can potentially result in shorter liner lifetimes. The NRC staff asked DOE to provide a technical basis for the diffusion coefficients for carbon dioxide and oxygen and to assess whether the coefficients used in the projections of steel liner lifetimes appropriately account for permeability changes of the concrete due to carbonation (see RAI-NF-2; Mohseni, 2013a [ML13196A135]). In response, DOE states that its use of 10^{-6} cm^2/s for both carbon dioxide and oxygen is bounding because a range of 10^{-7} to 10^{-9} cm^2/s was empirically determined for the intrinsic diffusion coefficient for intact concrete as a function of the water-cement ratio (Response to RAI-NF-2; SRR-CWDA-2013-00106, Rev. 1). Further, DOE indicates that it expected the permeability to increase by a factor of 100 due to cracking, thereby increasing the diffusion coefficient for oxygen and carbon dioxide by approximately 10,000. The NRC staff finds that DOE's use of higher oxygen diffusivities through submerged concrete (i.e., 10^{-4} cm^2/s) appears to be reasonable, if not bounding. However, the NRC staff continues to question whether DOE's use of a fixed 10^{-6} cm^2/s for carbon dioxide diffusivity through concrete in the base case (i.e., Case A) and for oxygen diffusivity through concrete for non-submerged tanks is appropriate. It is unclear whether this fixed diffusivity adequately accounts for the observed or expected degradation of 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 bypassing pathways through the grouted vault system and more aggressive steel liner service conditions than assumed in DOE's modeling.

Similar to concerns raised in the FTF TER (Camper, 2011 [ML112371751]), the NRC staff is concerned that alternative reasonable assumptions regarding the hydraulic performance of the concrete vaults over time, and the related effectiveness of the concrete vault as a diffusive barrier, would lead to faster failure times for the steel liners than estimated in the HTF 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 HTF 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; DOE/SRS-WD-2013-001, Rev. 0; SRNS-STI-2008-00096; and SRR-CWDA-2010-00128, Rev. 1) indicate a limited ability of the vault concrete to act as a fully effective and uniform barrier to the transport of species controlling steel corrosion (e.g., water, oxygen, chloride, carbon dioxide) over the vault's lifetime. 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 cementitious material performance in DOE's corrosion modeling are supported or appropriate.

Second, although DOE considered the possibility of liner corrosion activated by chloride, the resulting corrosion rates used to derive tank liner failure times are in the range of 1 to 2 μm/yr (0.04 to 0.09 mil/yr) as inferred by the NRC staff from values reported in Table 4.2-32 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1) and are not clearly distinguishable from passive corrosion. These low corrosion rates are dependent on (1i) the limited rate of transport of oxygen through the concrete vault and (2) the implicit assumption that oxygen reduction and enhanced iron dissolution occur on the same location on the liner surface. These assumptions appear questionable because (1) the heterogeneity in the system causes varying availability of oxygen; (2) the rebar, liner, and transfer lines are likely electrically connected; (3) the oxidation and reduction reactions controlling corrosion can be macroscopically separated; and (4) rebar corrosion could cause cracking of the concrete and later influence the transport of species affecting corrosion (carbon dioxide, chloride, oxygen). The NRC staff asked DOE to provide a basis for this implicit conclusion that the tank liners would remain essentially passive (See RAI-NF-1; Mohseni, 2013a [ML13196A135]). In its response (Response to RAI-NF-1; SRR-CWDA-2013-00106, Rev. 1), DOE states that it considered a broader range of corrosion rates beyond 1 μm/yr (0.04 mil/yr) for Type I and II tanks to account for the possibility of galvanic corrosion and degraded concrete vaults in its comprehensive stochastic methodology to estimate liner failure times.

The comprehensive stochastic methodology is used to develop the distribution of failure times reported in Table 4.2-31 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1). The average and median of the distribution of corrosion rates for saturated soil from the comprehensive stochastic methodology are on the order of 1.3 μm/yr (0.05 mil/yr) and remain within the passive corrosion regime. DOE estimates the impact of early liner failure on peak dose using early liner failure times reported in Table 5.6-47 as described in Section 5.6.7.6 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1). DOE reports that it observes no direct correlation between early liner failure and an increase in peak dose. The early liner failure times appear to approximate the lower, or first, quartile of results from the comprehensive stochastic methodology of liner failure times reported in Table 4.2-31 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1) except for Type IV tanks. However, DOE does not provide a basis for the estimates of early liner failure times used in the sensitivity analysis. While it is not clear to the NRC staff that the early liner failure times considered by DOE are reasonable, the NRC staff agrees with DOE that the early tank liner failure times that are evaluated by DOE are not likely to significantly affect the magnitude of the peak dose. Results from the HTF PA indicate that peak doses would exceed 0.25 mSv (25 mrem) TEDE at some point in the future from Ra-226. Given the delayed ingrowth of Ra-226 from its parent radionuclides, Pu-238, U-234, and Th-230, it seems more certain that the peak dose from Ra-226 would occur after 10,000 years. On the other hand, other uncertainty and sensitivity analyses (as discussed in Section 4.2.19) show the potential for radionuclides other than Ra-226 to dominate the peak dose earlier in time and at levels potentially above the performance objectives, should other assumptions and approaches relied on for Case A prove to be invalid (e.g., solubility limits as discussed in Section 4.2.9.3). Thus, the NRC staff, as also expressed in Section 4.2.9.2 of the FTF TER (Camper, 2011 [ML112371751]), continues to be concerned that early tank liner failure could result in doses that exceed the performance objectives within the period of compliance. For this reason, issues raised in the FTF TER regarding the potential for early liner failure (e.g., modeling partial failure of the tank liners as in a patch model) remain pertinent for HTF.

In summary, DOE selects a diffusion coefficient of 10^{-6} cm^2/s for carbon dioxide as a reasonably conservative value to represent intact concrete. However, it is not clear to the NRC staff that this value sufficiently and adequately accounts for the observed, predicted, and expected degradation of the concrete vaults, including cracking of reinforced concrete by rebar corrosion. It is also not clear to the NRC staff whether assuming corrosion rates controlled by oxygen diffusion properly accounts for the presence of joints and gaps within the concrete vaults or between the steel liner and the concrete vaults that might lead to bypassing pathways through the grouted vault system. Also, as discussed, metal locations where oxygen reduction occurs can be macroscopically separated from locations with enhanced iron dissolution. In other words, limited supply and transport of oxygen may not necessarily limit corrosion rates.

Although assumptions regarding steel liner performance are expected to be risk-significant, the NRC staff concludes that methods to obtain additional support for steel liner lifetimes may not be readily available and other key technical issues such as solubility limits for key radionuclides may be more tractable (Camper, 2013a [ML12345A322]). Therefore, pending results of solubility or leaching experiments planned by DOE (SRR-CWDA-2013-00049, Rev. 1; SRNL-RP-2013-00203, Rev. 0), as recommended in the FTF TER (Camper, 2011 [ML112371751]) and Section 4.2.9.3 of this TER, the NRC staff may also recommend additional experiments or alternative modeling treatment of the steel liner to strengthen the compliance demonstration.

Because chemical species that induce corrosion (i.e., water, chloride, carbon dioxide, and oxygen) need to be transported through the cementitious materials, the NRC staff finds that the uncertainty in steel tank liner longevity is related primarily to the hydraulic properties of the cementitious materials and their effect of the persistence of a chemical and physical environment that will limit corrosion of the steel liner. For FTF, the NRC staff identified the performance of the concrete vault as it pertains to steel liner corrosion as an area to be monitored (Monitoring Factor 3.1; Camper, 2013a [ML12345A322]). Similarly, the NRC staff plans to monitor the performance of the cementitious materials to limit steel liner corrosion at HTF.

NRC Evaluation of Stainless Steel Transfer Line Failure

Based on literature information on stainless steel degradation in underground environments, especially in contact with soil, the 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. The NRC staff finds DOE's use of a stainless steel general corrosion rate of 1 μm/yr (0.04 mil/yr) and the pitting corrosion rate of 25 μm/yr (1 mil/yr) to be reasonable as bounding values for estimating the failure times. Moreover, staff finds DOE's assessment that pitting corrosion is the controlling mechanism for transfer line failure to be appropriate.

4.2.9.3 NRC Evaluation of Source-Term Release

The NRC staff reviewed DOE's analysis of water chemistry contacting and leaching contaminants from the CZ, as described in Section 4.2.1.1 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1) and in supporting analyses presented in SRNL-STI-2012-00404, Rev. 0. DOE's analysis of the evolution of chemical conditions 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. The NRC staff notes that

there is little experimental support for the Geochemist's Workbench® modeling results. The NRC staff's concerns include the long-term evolution of chemical conditions (i.e., pH and E_h) and the solubilities of the various radionuclides in the CZ, as well as the modeled location and release of radionuclides located in the annuli and sand pads of the Type I and II tanks.

NRC Evaluation of pH

The pH of the grout system can be an important factor not only on the solubility of radionuclides but also 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 the evolution of pH. In the HTF PA, the geochemical modeling results indicate that the solubility of the HRRs are generally not sensitive to the transition from Oxidized Region II to Oxidized Region III (see Table 4-4 in Section 4.2.8.3 of this TER). However, if future revisions to the HTF PA and updated geochemical modeling indicate that certain radionuclide solubilities are sensitive to pH, DOE may need to provide additional support for the modeling results.

In the FTF TER (Camper, 2011 [ML112371751]), the NRC staff discusses concerns related to the assumed mineralogy of the grout and the geochemical modeling of the pH evolution of the grout (see Monitoring Factors 2.2 and 3.2 in the NRC Monitoring Plan; Camper, 2013a [ML12345A322]). In the review of the HTF PA, the NRC staff requested additional information regarding the assumed normative mineralogy used in the geochemical modeling (CC-NF-2; Mohseni, 2013a [ML13196A135]). The mineralogy determines the pH evolution of the grout pore water, which can subsequently affect radionuclide solubility. Accordingly, an incorrect mineralogy could lead to a modeled release of radionuclides that is non-conservative. In response to CC-NF-2, DOE discusses that the assumed mineralogy is consistent and appropriate with the chemical composition of the reducing grout (SRR-CWDA-2013-00106, Rev. 1).

In the HTF PA, as in the FTF PA, the pH evolution of the tank grout is modeled as occurring in discrete steps (Figure 4; SRNL-STI-2012-00404, Rev. 0). In the FTF TER, the NRC staff discusses that an experimentally-determined pH curve drops almost linearly with time for grouts, such as SRS grout, that contain low percentages of portland cement (Figure 4-5; Camper, 2011 [ML112371751]). Although the NRC staff continues to disagree with the assumed step-wise pH evolution of the grout, DOE acknowledges that the shape of the curve is a limitation of the geochemical model (Shaffner, 2012 [ML12236A370]). As a consequence of this model simplification, the pH is overestimated at some times and underestimated at other times. However, DOE also discusses that the solubility of key radionuclides is not significantly sensitive to these slight variations in pH. The NRC staff will continue to evaluate information related to chemical transitions and groundwater conditioning during monitoring.

NRC Evaluation of Reduction-Oxidation Potential

Consistent with NRC staff's review of the FTF PA, the 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. In review of the HTF PA, the NRC staff continues to have concerns about the ability of the overlying reducing grout to condition the CZ to help maintain the low solubility of key radionuclides. Although DOE assumes in the HTF PA

that two key radionuclides—plutonium and technetium— are not sensitive to the transition from reducing to oxidizing conditions (see Table 4-4), the NRC staff discusses later in this section (see NRC Evaluation of Radionuclide Solubility) that DOE has not provided sufficient support for the assumed phases of several key radionuclides, including plutonium and technetium. The NRC staff's concerns regarding the ability of the backfill grout to condition the CZ are documented in the FTF TER (Camper, 2011 [ML112371751]) and FTF Monitoring Plan under Monitoring Factors 2.2, 3.2, and 3.3 (Camper, 2013a [ML12345A322]), including (1) the ability of the overlying grout to condition water infiltrating through preferential pathways, and (2) the formation of a passivation layer, which may limit the ability of the blast furnace slag to condition the infiltrating water. In addition to these aforementioned concerns, recent research raises additional questions regarding the ability of the grout to condition the CZ.

The pore water E_h in slag-bearing grout is derived from Geochemist's Workbench® analyses that assume equilibrium with pyrite to account for the reducing capacity of the grout. 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 and 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. 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 cracked 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 [ML101160513]; Pabalan et al., 2012 [ML12089A319]) or limited by depletion of buffering phases along the crack–matrix interface (Harris et al., 1997). In the second, at high pH, iron oxyhydroxides may precipitate on the surface of pyrite or other low-redox-buffering mineral phases, armoring the surface, and protecting it from further oxidation.

If flow occurs through cracks, 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. The Center for Nuclear Waste Regulatory Analysis research shows that if flow rates are high through the system, the groundwater may not be fully conditioned by the reductants (i.e., disequilibrium between pore water and solid matrix; Pabalan et al., 2012 [ML12089A319]). In response to RAI-NF-4, DOE states that the residual waste will be conditioned by the overlying reducing grout through diffusion, regardless of whether or not water travels through reducing grout to reach the CZ. However, conditioning of the CZ by the overlying grout, either by diffusion or flow through the matrix, might be slow relative to the rate at which infiltrating water, migrating via preferential pathways, contacts the CZ. Accordingly, the longevity of reducing chemical conditions and the retention of redox-sensitive elements may be overestimated in the HTF PA.

With respect to the potential for armoring of the surface, recent research suggests that the method by which DOE calculates the reduction capacity may be optimistic. DOE relies on a Ce(IV) titration method (SRNL-STI-2009-00637, Rev. 0), which utilizes sulfuric acid and results in the complete reaction of the reducing phases (PNNL-22977). An alternative approach is the Cr(VI) method, which is conducted under neutral or alkaline conditions (PNNL-22977). The latter approach more closely represents the alkaline conditions that are anticipated for the grouted tanks. Under these alkaline conditions, the Pacific Northwest National Laboratory

researchers noted that a passivation likely occurred, which could limit the effective reduction capacity of the blast furnace slag. As a result, it may be that the availability of reducing minerals is quite limited. This is especially the case in static conditions (Perez-Lopez et al., 2005). In both cases (i.e., fracture flow and surface armoring), the use of an equilibrium based approach to E_h control is very likely not appropriate.

The HTF PA (SRR-CWDA-2010-00128, Rev. 1) states that the use of pyrite is simply a method to account for the reducing capacity of the grout and is not meant to imply that pyrite grains in the grout are the primary source of reducing capacity. However, the lack of empirical data supporting the selection of pyrite to represent the grout reducing capacity implies that the calculated E_h transition time is uncertain. For example, a recent study by Um et al. (PNNL-22977) stated that CaS (oldhamite) is a major source of reduced sulfur in blast furnace slag. If this is the case, the grout would provide less redox buffering capability than assumed in the HTF PA because CaS is relatively soluble and could be released by infiltrating groundwater prior to reacting with dissolved oxygen in the infiltrate. In another study (PNNL-21723), researchers observed reduced sulfur species in leachates from saltstone simulant experiments that also utilized blast furnace slag to impart reducing conditions (RAI-NF-3; Mohseni, 2013a [ML13196A135]). If the reduced sulfur phase(s) in blast furnace slag is more soluble than pyrite, DOE's geochemical modeling would overestimate the longevity of reducing conditions in the grout and the release of redox sensitive radionuclides could occur sooner than assumed (Alexander, 2013 [ML12272A082]). In response to RAI-NF-3 (SRR-CWDA-2013-00106, Rev. 1), DOE discusses that reduced sulfur in the blast furnace slag may be more soluble than pyrite; however, due to its highly reactive nature it will react with iron and potentially other metals to form metal sulfides once it is released from the slag. Based on the prevalence of iron in the reducing grout and the long period of time considered in the HTF PA, DOE believes that iron would react with the reduced sulfur species and eventually form pyrite. Based on the proposed grout formulation in Table 2 of SRNL-STI-2012-00404, Rev. 0, the NRC staff agrees that there is sufficient iron (254 mol/m^3) to form the amount of pyrite assumed for the HTF PA. However, there is significantly more calcium (approximately 1,700 mol/m^3) present in the system, which could form calcium-sulfide phases that are more soluble than pyrite. It is also not clear to the NRC staff that relatively soluble reduced sulfur phases (e.g., amorphous metal sulfides, oldhamite) will necessarily transition to less soluble forms of metal sulfides (e.g., pyrite, pyrrhotite) during the period of performance. The less soluble phases may be thermodynamically favored; however, kinetic limitations may preclude the formation of the less soluble phases in the tank grout during the performance period.

Additional research that has recently been conducted on salt-containing grouts raises questions regarding the use of reduction capacity as a parameter in determining the long-term chemical retention of redox-sensitive radionuclides (SRNL-STI-2013-00541, Rev. 0; CBP-TR-2013-02, Rev. 0). The researchers observed that the reduction capacity measurements, as determined by the Ce(IV) method, did not correlate to the release of redox-sensitive species. Because there was not a correlation between the measured reduction capacity and the release of the studied radionuclides (Cr, Tc), the authors concluded that reduction capacity is not an appropriate or meaningful parameter for determining or predicting chromium and technetium and oxidation states, speciation, and solubilities in cementitious materials. The NRC staff notes that these reports differ from the tank farms in two regards. First, this research evaluated cast stone and saltstone simulated grouts. Second, the conceptual models for waste release from the salt-containing grouts and tank backfill grout also differ. For the salt-containing grout

wasteforms, the contaminants are co-located with the grout versus a discrete CZ beneath the grout in the tanks. However, these salt-containing cementitious materials rely on blast furnace slag as the primary reductant, similar to the reducing tank backfill grout. Although this research did not directly evaluate tank backfill grout, it further highlights the need for support (e.g., laboratory and field experiments) for key modeling assumptions. The NRC staff will continue to follow research related to the use of reduction capacity as a parameter in determining the long-term chemical retention of redox-sensitive radionuclides during the monitoring period.

DOE conducted several analyses to provide insight into the uncertainty related to the chemical transitions. DOE states in response to RAI-NF-4 (SRR-CWDA-2013-00106, Rev. 1), that the treatment of the uncertainty in the grout's reactive surface area and reducing capacity are adequate in the HTF PA. DOE evaluates the uncertainty in the grout reactive surface area and reducing capacity in sensitivity analyses through one of two methods: (1) varying the redox transition times by ±30 percent; or (2) in extreme cases, assuming that the pore water E_h is unaffected by the slag and applying solubility limits associated with oxidizing conditions. DOE also states that both portland cement and fly ash have substantial reduction capacity, as shown in Table 2 of SRNL-STI-2009-00637, Rev. 0, in addition to blast furnace slag, which could contribute to the grout reducing capacity but are not included in the base case calculations. Additional DOE model results indicated that adding magnetite to represent the reducing capacity of fly ash in the grout degradation calculations would result in redox transition times later than assumed in the base case calculations. Nonetheless, the NRC staff is not convinced that the ±30 percent in the chemical transition times adequately bounds the uncertainty in the first chemical transition time. The potential for (1) flow bypassing the reducing grout, (2) the formation of a passivation layer, and (3) the early release of soluble reduced sulfur phase(s) results in substantial uncertainty regarding the ability of the overlying grout to condition the CZ and limit the release of redox-sensitive radionuclides.

In addition, DOE provides a sensitivity analysis in response to RAI-NF-3 (SRR-CWDA-2013-00106, Rev. 1) that varies the number of pore volumes required for the chemical transitions by a factor of 0.05 to 20. The results of the sensitivity analysis indicate that the variation in the number of pore volumes to chemical transition increases the peak dose in the HTF PA by less than a factor of 10. However, several aspects limit the information provided by the sensitivity analysis. First, there are multiple factors that may decrease the effective reduction capacity (e.g., early release of reduced sulfur phases, passivation of the reduced sulfur phases, bypassing flow) and it is not clear whether the variation in the number of pore volumes would account for these uncertainties. Second, DOE utilizes the base case (i.e., Case A) in the sensitivity analysis; however, the NRC staff has multiple concerns with the conceptualization and parameterization of the base case. These concerns are often interrelated and may result in nonlinear effects on the dose projections. Accordingly, the use of one-off sensitivity analyses based on the base case provides limited additional information. To increase confidence in the modeling approach and results, detailed experimental studies are needed, as the NRC staff discusses under Monitoring Factors 2.2 and 3.2 in the FTF Monitoring Plan (Camper, 2013a [ML12345A322]).

NRC Evaluation of Radionuclide Solubility

The NRC staff reviewed DOE's ICM abstraction for solubility limits, as described in Section 4.2.1.2 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1) and supporting documents, most notably SRNL-STI-2012-00404 and RAI responses in SRR-CWDA-2013-00106, Rev. 1. The review focused on the more risk-significant isotopes of technetium, radium, uranium, neptinum, and plutonium. In the ICM, DOE assumes (1) that the CZ is located immediately above the tank bottom and (2) that the chemistry of the tank grout and the CZ are intimately related, but (3) that solubility-controlled release from the tanks is not necessarily dependent on transport of radiological constituents through the tank grout overlying the CZ because the radionuclide inventory is not initially assumed to be bound in the tank grout. The solubility limits are defined for Reduced Region II, Oxidized Region II, and Oxidized Region III for tanks above the water table, and for Conditions A, C, and D for submerged tanks (Condition B values are not tabulated because the Condition D chemistry is nearly identical; Table 4.1-11, SRR-CWDA-2010-00128, Rev. 1). The 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 some key radionuclides, the modeled pure phase solubility limits seem reasonable. For uranium, the solubility limits of around 5×10^{-9} mol/L for reducing conditions, controlled by amorphous, hydrated UO_2, are reasonable when considered against an independent evaluation of solubility limits in cementitious systems (Pabalan et al., 2009 [ML090980426]). Likewise, the uranium values at around 10^{-5} for oxidizing conditions, controlled by schoepite, are acceptable. Furthermore, DOE model results in Figure 32 of SRNL-STI-2012-00404 suggest that the uranium solubility limit is not sensitive to E_h above −180 mV. The uranium solubility limit, therefore, is not subject to uncertainties in the assumption of E_h once oxidizing conditions are established. Assignment of radium solubility control to $RaSO_4$ is appropriate, and the radium solubility limit is relatively insensitive to E_h.

The neptunium solubility limit of 1×10^{-9} mol/L under reducing conditions (i.e., Reduced Region III and Condition C) controlled by amorphous, hydrous NpO_2 is reasonably consistent with Pabalan et al., (2009 [ML090980426]). DOE uses the same controlling solid to determine neptunium solubility limts for Oxidized Regions II and III, which are 3×0^{-7} mol/L and 2×10^{-6} mol/L, respectively, and are higher in the lower-pH oxidizing waters of Conditions A and D. While these values are considerably lower than corresponding solubility limits calculated for the FTF PA (SRS-REG-2007-00002, Rev. 1), they are reasonable considering that DOE uses a thermodynamic database based on the well-established Nuclear Energy Agency (NEA) data compilation. Pabalan et al. (2009 [ML090980426]), find that NEA-based data give similar neptunium solubility limits. However, DOE model results in Figure 24 of SRNL-STI-2012-00404 show that the neptunium solubility limit is sensitive to E_h for Oxidized Region III. A value of 3×10^{-5} mol/L (i.e., over an order of magnitude higher) is possible if E_h is 400 mV, rather than 290 mV. Similar E_h sensitivity is shown in SRNL-STI-2012-00404 for Condition A water. The neptunium solubility limit, therefore, is subject to the same uncertainties with respect to knowledge of the water oxidation-reduction potential as is discussed next for plutonium.

In the FTF TER (Camper, 2011 [ML112371751]) and FTF Monitoring Plan (Camper, 2013a [ML12345A322]), the NRC staff discusses concerns regarding the assumed plutonium solubility. In response to the NRC staff's TER, DOE conducted a plutonium solubility peer review study

(LA-UR-12-00079) and updated their geochemical model (SRNL-STI-2012-00087, Rev. 0). The NRC staff's review of these documents is discussed in further detail in a technical review report (TRR) on waste release and solubility (Alexander, 2013 [ML12272A082]). In the HTF PA, DOE extended the updated geochemical modeling to include additional radionuclides and oxalate. There are several notable differences associated with the assumed plutonium solubility in the HTF PA, versus the FTF PA, including revisions to: the solubility controlling phase (i.e., the HTF PA assumes that plutonium is present as an amorphous hydrous oxide, rather than being co-precipitated with iron), the geochemical database and modeling, the geochemistry associated with the various chemical conditions, and the presence of submerged contaminants. However, the NRC staff continues to be concerned about the assumed solubility of plutonium and the limited support that has been developed by DOE. Important details from the previous reviews and review of new aspects are provided below.

The plutonium solubility limits are discussed with respect to two areas of uncertainty: (1) E_h sensitivity and (2) solid phases controlling plutonium solubility. DOE calculated and adopted plutonium solubility limit for all waters (except the low-pH Condition A water) is 3×10^{-11} M, using amorphous, hydrous PuO_2, or PuO_2(am,hyd), as the controlling solid. This value is corroborated by the models using NEA data in Pabalan et al. (2009 [ML090980426]) for E_h below 400 mV. However, consideration of literature information by Pabalan et al. (2009 [ML090980426]) suggests that a value about an order of magnitude higher would be more inclusive of available data. (Note that the effect of colloids is neglected.) As discussed in SRNL-STI-2012-00404, plutonium solubility is not sensitive for Region II at E_h below 450 mV and is not sensitive for Region III at E_h below 530 mV. Above those thresholds, the plutonium solubility limit increases strongly with E_h. However, the NRC staff notes in a TRR (Alexander, 2013 [ML12272A082]) that there is significant uncertainty in DOE's model with respect to thermodynamic modeling calculations and the assumed threshold E_h. For the low-pH Condition A water, the plutonium solubility limit is E_h-sensitive below 230 mV and above 570 mV. It is important to consider, therefore, whether such higher E_h values are feasible under Region II or III conditions.

The E_h values used for oxidized solubility simulations in SRNL-STI-2012-00404—240 mV for Oxidized Region II and 290 mV for Oxidized Region III—are not based on grout degradation and oxidation models, but on an interpolation between a hypothetical pH 12.5 cement and SRS groundwaters on an E_h-pH plot (Figure 17; SRNL-STI-2012-00404). While it may be reasonable to adopt E_h values below the E_h-pH line representing equilibrium with atmospheric oxygen, questions can be raised regarding DOE's approach. The argument for adopting these E_h values rests on two lines of reasoning.

First, Figure 16 of SRNL-STI-2012-00404 reproduces a figure from Langmuir (1997) showing that E_h values in natural waters do not correspond to theoretical equilibrium with measurable dissolved oxygen. DOE draws red ovals corresponding to the pH for Regions II and III in the plot area corresponding to near-surface environments in contact with atmosphere. The center of the oval at pH ~11 (Region II) does correspond approximately to the DOE assumed E_h of 240 mV. Some caution, however, should be employed in posing this argument, because the E_h-pH field in Figure 16 uses natural waters and relies on no measurements at pH greater than about 10 (see Figure 11.3; Langmuir, 1997). The oval at pH ~9 (Region III) centers on 350 mV, which is higher than the 290 mV used by DOE for Region III.

The second line of reasoning employs an interpolation between assumed fresh cement E_h-pH and values measured in SRS groundwaters at the water table (Figure 17; SRNL-STI-2012-00404, Rev. 0). DOE cites data on fresh grouts to establish the high-pH end-member, but that end-member of the line in Figure 17 is not necessarily relevant to aged grout. Oxidized Regions II and III, by definition, apply to systems in which the reducing capacity of the grout has been exhausted. There is not a technical basis for assuming that shallow waters interacting with aged grout under oxidizing conditions will evolve along a straight E_h-pH line between fresh grout and groundwaters at the water table. In a TRR for FTF (Alexander, 2013 [ML12272A082]), the NRC staff also discusses that there is significant uncertainty with respect to the E_h of infiltrating water. DOE relies on a limited number of water table samples to establish the chemical conditions for water infiltrating the waste tanks; however, these groundwater samples are not necessarily representative of the infiltrating water. Several factors may result in the groundwater not being representative of the water infiltrating the waste tanks, including (1) uncertainty in the groundwater measurements, (2) potentially greater levels of dissolved oxygen in the vadose zone versus the water table wells, and (3) differences in the minerals present in the grout and vadose zone soils. Therefore, the E_h values established by DOE for Oxidized Region II and III solubility calculations must be considered uncertain. Additional information is needed to support the selection of plutonium solubility based on the expected E_h and updated geochemical modeling

As discussed above, this uncertainty may be significant at the scale of an order of magnitude for neptunium in Oxidized Region III. For plutonium, this uncertainty is only significant if it is feasible for E_h to exceed 450 mV in Region II and 530 mV in Region III (Figures 44 and 45; SRNL-STI-2012-00404, Rev. 0), provided that these E_h thresholds are reasonable. It should be mentioned that the positions of the phase boundaries in these plots are themselves uncertain. Even if it cannot be shown with confidence that these E_h values will be reached, DOE needs to consider these uncertainties in establishing what they consider to be conservative neptunium and plutonium solubility limits. In response to RAI-NF-6, DOE poses arguments having to do with the E_h of fresh cements (SRR-CWDA-2013-00106, Rev. 1) but, as noted above, Oxidized Region II and III cements have evolved in E_h-pH space since the time they were fresh.

Another area of uncertainty regarding plutonium solubility is in the choice of controlling solid, which is the main topic addressed in RAI-NF-6 (Mohseni, 2013a [ML13196A135]). DOE states in its response (SRR-CWDA-2013-00106, Rev. 1) that experiments are planned (also focused on technetium, uranium, and neptunium) that will explore what solids may be controlling plutonium behavior in actual tank waste residue samples. The NRC staff continues to encourage this type of study. In the absence of such data, DOE further defends its use of PuO_2(am,hyd) as the controlling solid in the tanks. For example, DOE points out that, although plutonium carbonates may form in Tank 18 during water-washing, they would eventually dissolve or be converted to PuO_2(am,hyd) after grouting. In spite of the DOE response, the NRC staff considers the characteristics of plutonium solids in tank waste to remain an open question. As the NRC staff discusses in an FTF TRR (Alexander, 2013 [ML12272A082]) and HTF RAI-NF-6 (Mohseni, 2013a [ML13196A135]), the NRC staff is concerned that the assumption of PuO_2(am,hyd) control is not consistent with SRS observations.

FTF residual waste samples from Tank 18 were indicative of a plutonium carbonate phase that had a significantly higher solubility than the phase assumed in the HTF PA (Alexander, 2013 [ML12272A082]). As the NRC staff discusses in the FTF TRR, there may be a thermodynamic

potential for plutonium carbonate phase(s) to transform into PuO_2(am, hyd); however, several factors may inhibit or preclude this transformation. If higher solubility plutonium phases are present in risk-significant quantities after the final cleaning of the tanks, additional information will be needed to demonstrate that the 10 CFR Part 61 performance objectives will be met. This information includes verification that any higher-solubility plutonium phases will convert to lower solubility phases under reducing-grout conditions.

In the FTF PA, DOE uses an iron (Fe) co-precipitation model to establish very low solubility limits under some conditions for technetium, urainum, and plutonium in the base case deterministic model. For the HTF PA, the iron co-precipitation model is retained in the base case deterministic model only for the technetium solubility limit, while iron co-precipitation values are used to set lower limits for probabilistic analyses for technetium, uranium, neptunium, and plutonium. The calculated solubility limits using the model for technetium, uranium, neptunium, and plutonium are shown in Table 4.2-14 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1), and can be compared with the modeled pure phase solubility limits in Table 4.2-11 of the HTF PA. Because these lower limits are assigned 50 percent probabilities in DOE's probabilistic analyses (Tables 5.6-10, 5.6-11, and 5.6-12; SRR-CWDA-2010-00128, Rev. 1), the iron co-precipitation model remains a significant issue for all four elements.

The iron co-precipitation model establishes very low solubility limits for technetium from 1×10^{-15} to 3×10^{-12} mol/L for the base case deterministic model, even for oxidizing conditions (Table 4.2-14; SRR-CWDA-2010-00128, Rev. 1). In contrast, Pabalan et al. (2009 [ML090980426]) calculate technetium solubility limits no lower than approximately 10^{-7} mol/L for a reasonable range of reducing conditions and they recommend that no limit be applied to technetium under mildly reducing to oxidizing conditions. The low iron co-precipitation values in the HTF PA strongly inhibit technetium release from the tanks. Because technetium is a potentially strong dose contributor in the HTF PA, the technical basis for the very low concentrations in the iron co-precipitation model should be well supported. Likewise, the use of the typically very low solubility limits for technetium, uranium, neptunium, and plutonium in the probabilistic models influences the results of those analyses and is, therefore, significant to understanding uncertainty in the compliance demonstration.

The iron co-precipitation model assumes (1) a ratio between the element and iron in the tank residue, (2) that the ratio in solution is the same, and (3) that the iron solution concentration is controlled by the solubility of magnetite (reducing) or maghemite (oxidizing; hematite for FTF models). In a series of FTF RAIs—FTF-NF-22 and FTF-NF-23 in SRR-CWDA-2009-00054, Rev. 0, and FTF-RAI-NF-8 and FTF-RAI-NF-9 in SRR-CWDA-2011-00054, Rev. 1—the NRC staff requested further information supporting the model and the concentration limits DOE developed from the model. The issue arose again in HTF RAI-NF-7, with the NRC staff stating that DOE had not demonstrated that (1) all soluble Tc-99 will be removed during tank washing and (2) all remaining Tc-99 will be co-precipitated with iron phases (Mohseni, 2013a [ML13196A135]). The NRC staff notes in their comments that there is the possibility that DOE would eventually be able to eliminate this concern if Tc-99 tank inventories are lowered enough by cleaning such that this radionuclide would not pose a risk, as the NRC staff discusses in Section 3.1 of the FTF Monitoring Plan (Camper, 2013a [ML12345A322]). DOE responds that, to date, this level of Tc-99 removal had been achieved in all washed tanks and that they would make a determination of tank Tc-99 inventories once all tanks have been cleaned (Response to RAI-NF-7; SRR-CWDA-2013-00106, Rev. 1). This determination will likely be followed by new

calculations to demonstrate compliance. The DOE response also presents sensitivity analyses that demonstrate that when a range of solubility limits are considered (including the case of no limit) the Tc-99 dose will not exceed the compliance limit within 10,000 years.

The NRC staff does not consider these sensitivity analyses definitive in resolving the question of technetium control by iron co-precipitation, because the magnitude and timing of Tc-99 dose depends on factors other than solubility (e.g., timing of initial radionuclide release) that may themselves be subject to uncertainty. Therefore, DOE needs to continue to bolster the technical basis for the model, unless it is demonstrated that the risk from Tc-99 has been sufficiently reduced by cleaning. In addition, the model's application to uranium, neptunium, and plutonium remains a technical issue because the low solubility limits calculated for those elements using this model affect the probabilistic results. As discussed in the FTF TER (Camper, 2011 [ML112371751]), evidence suggests that compounds other than iron oxides may be important for technetium, uranium, neptunium, and plutonium in SRS tank residues and could control solubilities. As shown in SRNL-STI-2209-00492, technetium in one tank seemed to follow iron removal, suggesting at least some co-precipitation. In another tank, even though 73 percent of the iron was removed, very little technetium 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 addition, DOE argues that soluble Tc-99 is removed during cleaning; however, technical challenges can occur that may compromise this assumption.

In another example, King et al. (2010) gives concentrations of uranium and plutonium in what appear to be solubility controlled solutions (both from under- and over-saturation). Five different actual supernate samples were analyzed and then spiked with uranium or plutonium to drive precipitation. Observed ranges of concentrations that represent solubility limits are 4×10^{-5} mol/L to 2.5×10^{-4} mol/L of uranium and approximately 8×10^{-6} mol/L of plutonium. The plutonium values are much higher than any of those given in Tables 4.2-11 and 4.2-14 in SRR-CWDA-2010-00128, Rev. 1. For uranium, the observed concentrations are higher than any in the reduced conditions and higher than most in the oxidized conditions in Tables 4.2-11 and 4.2-14. The authors identify clarkeite as the likely uranium 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 Scanning Electron Microscopy/Energy-Dispersive X-ray Spectroscopy from before and after oxalic acid cleaning of SRS tank waste in WSRC-STI-2007-00192, Rev. 1. Uranium is observed in the unwashed samples, often with iron and aluminum. Although a large fraction of uranium is dissolved by the acid, samples from filtrate after oxalic acid cleaning clearly show discrete uranium compounds present, apparently oxides or hydroxides. The fraction of these compounds to total uranium is unknown. It is unknown if plutonium is co-precipitated with iron 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 uranium) are retained in a number of compounds.

The report SRNL-STI-2012-00106, Rev. 0 discusses the issue of plutonium solids in waste from the bottom of Tank 18. Scanning Electron Microscopy analysis showed that plutonium was present as discrete particles; suggesting that plutonium was present in at least two forms. One form is particulate plutonium and the other may be as a co-precipitated phase. It is noted that the larger ionic radius of Pu^{+4} relative to Fe^{+3} makes it unlikely that plutonium would substitute for iron in an iron oxide lattice, although plutonium may be held as an occlusion in an iron lattice

or by adsorption on the iron oxide surface. The plutonium ionic radius makes it more likely to substitute into a $MnO_x \cdot xH_2O$.

DOE's probabilistic modeling includes the assumed co-precipitation of neptunium, uranium, plutonium, and technetium with low solubility iron-bearing minerals (magnetite and maghemite) 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 the NRC staff discusses in Monitoring Factor 2.1 of the FTF Monitoring Plan (Camper, 2013a [ML12345A322]). As discussed in DOE's response to FTF-RAI-NF-8 in SRR-CWDA-2011-00054, Rev. 1, each tank is to be sampled for determination of its final inventory. It was noted that "analysis may be performed to further the understanding on the waste release assumptions regarding iron co-precipitation and solubility controlling phases". The NRC staff strongly concurs with this statement.

The NRC staff concludes 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. This information would be relevant not only to the iron co-precipitation model, but also to other questions on controlling solid phases such as those raised in this section regarding plutonium. 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. Characterization studies have shown that on the micron scale the FTF tank waste solids are heterogeneous and that radionuclides are simultaneously retained in a variety of compounds (SRNL-STI-2012-00123, Rev. 0). Perhaps more importantly, washing the tanks with oxalic acid, will remove some large percentages of uranium (SRNL-STI-2009-00493, Rev. 0) 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., plutonium oxalate). The very low solubility radionuclide compounds or co-precipitated radionuclides will not control solubility until other higher-solubility compounds have become depleted. The 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. 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. DOE has included oxalate compounds in the geochemical modeling (SRNL-STI-2012-00404, Rev. 0). However, experimental evidence of the solubility of the residual compounds remains important in providing support for the geochemical modeling

Given the difficulty in quantifying the solid phases that contain radionuclides, the NRC staff has concluded—consistent with the FTF TER (Camper, 2011 [ML112371751])—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 that 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 (ASTM, 2008) or a similar static method. Sampling intervals can be informed by results in Hobbs (1999) and King et al. (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). The static tests will indicate maximum solution concentrations. Other leach tests as described in Krupka et al. (2006) and Cantrell et al. (2006) will be needed to determine concentrations of important elements in solution after the 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 soluble 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 calcium, iron, aluminum, silicon, and major anions. This information is needed as input to geochemical modeling of tank leachate chemistry. Test results should be compared to calculated water chemistries and releases determined by modeling (e.g., Tables 10, 11, and 14; SRNL-STI-2012-00404, Rev. 0).

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_d studies with various SRS soils. Verification of partitioning of key radionuclides onto natural sediment at the SRS 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).

DOE discusses uncertainties in solubility limit estimations briefly in the HTF PA (Section 4.2.1.2.8; SRR-CWDA-2010-00128, Rev. 1) and more thoroughly in SRNL-STI-2012-00404. For example, Figure 4.2-11 in the HTF PA (SRR-CWDA-2010-00128, Rev. 1) plots uncertainty ranges on model solubility limits based on thermodynamic data uncertainties; the uncertainties typically span one to two orders of magnitude. Likewise, SRNL-STI-2012-00404 provides a large number of graphical illustrations of solubility limit sensitivities to factors such as pH and E_h. These solubility limit uncertainties, however, were apparently not considered in DOE's HTF PA uncertainty and sensitivity analyses. The technetium, uranium, neptunium, and plutonium solubility limit parameter distributions used for the probabilistic GoldSim™ model are presented in Section 5.6.3.3 of SRR-CWDA-2010-00128, Rev. 1. DOE states that the uncertainty in solubility values is accounted for by conservatively selecting the solubility controlling phases. DOE also discusses that the uncertainty in thermodynamic data is addressed by modifying the chemical transition times in GoldSim™. It is not apparent that the solubility values with the probabilities shown in the tables in Section 5.6.3.3 of SRR-CWDA-2010-00128 have been conservatively selected, particularly since the NRC staff considers that other, higher-solubility limit phases have not been considered (e.g., discussion of plutonium earlier in this section). It is also not clear how the uncertainty in thermodynamic data is accounted for with the modifications in the chemical transition times. For example, the solubility of plutonium does not appreciably vary between three assumed chemical

conditions. However, DOE indicates that the uncertainty in thermodynamic data can result in two orders of magnitude variation in solubility (SRNL-STI-2012-00404, Rev. 0). Accordingly, varying the chemical transition times would not account for the uncertainty in thermodynamic data related to the solubility of plutonium.

For all four elements (technetium, uranium, neptunium, and plutonium), two solubility limits are shown in Section 5.6.3.3 of the HTF PA (SRR-2010-00128, Rev. 1) with equal probabilities. The 0.5 probability values assigned to each solubility limit are said to be "based on observations in the literature and expected thermodynamic stability," with a citation of SRNL-STI-2012-00404, Rev. 0. The latter report, however, does not contain information supporting these probabilities, which appear to have no technical basis. For uranium, neptunium, and plutonium, the higher of the two values corresponds to the base case solubility limits (based on modeling of pure phase solubility), while the lower value is based on the iron co-precipitation model. In other words, the probabilistic uncertainty analyses for these elements do not consider values higher than the base case. Such an analysis does not appear useful for observing the effects of underestimating the solubility limit. These comments concerning the two values adopted for the probabilistic analysis do not directly apply to technetium under oxidizing conditions, because the higher technetium probabilistic value is not from the base case and corresponds to the case with no solubility limit. Nevertheless, it is not clear that the very low solubility limits used in all technetium probabilistic distributions are appropriate.

Furthermore, the uncertainty/sensitivity analysis does not take advantage of the information gathered in SRNL-STI-2012-00404, Rev. 0 on solubility limit uncertainties. To help account for the uncertainty in relying on thermodynamic data for solubility values, DOE utilized the uncertainty information provided by the NEA with the NEA database (SRNL-STI-2012-00404, Rev. 0). The range of solubility values, as calculated in SRNL-STI-2012-00404, Rev. 0, exceeds the base case solubility values assumed in the probabilistic model for plutonium, neptunium, and uranium. The final paragraph of Section 5.6.3.3 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1) states, "Uncertainty in the controlling phase is presented above, while uncertainty in thermodynamic quantities is addressed by implementing four bounding conditions discussed in detail in Section 4.2.2." The four bounding conditions may address conceptual model uncertainty, but they are not related to thermodynamic data uncertainties. In summary, the probabilistic uncertainty/sensitivity analyses do not appear entirely useful for understanding system behavior in response to plutonium, neptunium, and uranium solubility limits. The NRC staff discusses in RAI-NF-9 (Mohseni, 2013a [ML13196A135]) that they disagree with DOE's assertions that (1) solubility controlling phases for plutonium, neptunium, technetium, and uranium are conservatively selected and (2) varying the chemical transition times adequately accounts for the uncertainty in thermodynamic data. Based on the lack of direct evidence supporting the assumed solubility controlling phases and the thermodynamic modeling reported in SRNL-STI-2012-00404, Rev. 0, which suggests higher solubility values are possible, the NRC staff is concerned that the assumed probability distributions in the HTF PA for plutonium, neptunium, technetium, and uranium are optimistic.

The HTF PA provides only limited consideration of the effects of higher solubility limits for plutonium, neptunium, technetium, and uranium (SRR-CWDA-2010-00128, Rev. 1). In Section 5.6.7.1 of the HTF PA, DOE evaluates higher solubility limits in a deterministic "synergistic" sensitivity case using Case C, which has (1) a fast pathway through the grout with no conditioning of the CZ by the overlying grout and (2) early steel liner failure times. This

synergistic case does not show a significant increase in dose; however, the case includes solubility values and basemat performance that the NRC staff considers to be optimistic, based on DOE's limited model support. In Section 5.6.7.3, DOE conducts a sensitivity analysis around the base case using the deterministic PORFLOW™ model, except with solubilities set to more conservative values (SRR-CWDA-2010-00128, Rev. 1). The results of "Study 1" in Table 5.6-46 of the HTF PA show no elevated doses before 10,000 years (Fig 5.6-70; SRR-CWDA_2010-00128, Rev. 1). Nevertheless, an appropriate probabilistic analysis would have provided more information on the system behavior of plutonium, neptunium, technetium, and uranium. As mentioned earlier in this report, the NRC staff does not consider this one-off sensitivity analysis to be definitive with respect to solubility uncertainty, because many other processes and system components contribute to the dose results.

In response to the NRC staff's comments on this topic in RAI-NF-9, DOE did not explain how the original probability distributions were determined nor how its approach incorporated such uncertainties as those associated with thermodynamic data (SRR-CWDA-2013-00106, Rev. 1). DOE's response reports new probabilistic analyses—for five water chemistry conditions—that include, for each element, a "pessimistic" solubility limit that is much higher than the base case, except for technetium under oxidizing conditions. DOE assigns 25 percent probabilities to each of four values, corresponding to (1) the modeled pure phase solubility, (2) the pessimistic case, (3) the normal iron co-precipitation case, and (4) the pessimistic iron co-precipitation case. The basis for the 25 percent assignment is not provided; effectively, the very low-solubility iron co-precipitation cases still comprise 50 percent probability. The resulting probabilistic base case GoldSim™ calculation was very similar to the result reported in the HTF PA (Figure 5.6-31; SRR-CWDA-2010-00128, Rev. 1), with a peak mean dose of 11 mrem/yr (Table RAI-NF-9.6; SRR-CWDA-2013-00106, Rev. 1). The new probabilistic calculation is informative about system behavior, and it is recommended that DOE in the future incorporate in probability distributions "pessimistic" values that exceed base case solubility limits. The NRC staff has not changed its conclusion that the values used in the probabilistic analysis, as well as their probability assignments, need stronger, more explicit technical bases.

In summary, the NRC staff continues to recommend that the SRS tank closure programs perform 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. The NRC staff also recommends experiments to study the pH and E_h evolution of grouts proposed for SRS tank facility closure. In combination with data gathering, DOE should continue to refine the technical bases for model and parameter selections and associated uncertainty analyses. These efforts will provide more defensible bases for solubility limit abstractions supporting assessments demonstrating compliance with performance objectives. DOE and the NRC staff have been discussing these topics in the context of FTF closure monitoring and expect tocontinue these discussions during HTF interactions. For example, the NRC staff is aware of several relevant ongoing and planned studies that have not yet been incorporated into the HTF PA, including waste release experiments on Tank 18 residual solids (SRNL-RP-2013-00203, Rev. 0).

NRC Evaluation of Annulus and Sand Pad Release

As discussed in Section 4.2.8.3 of this TER, in addition to the tank inventories contained within the primary liners, there is relatively soluble contamination located outside of the primary liners in the Type I and Type II tanks. In RAI-NF-12 and RAI-NF-13, the NRC staff questioned whether the risk from these radionuclide inventories is adequately addressed within the HTF PA (Mohseni, 2013a [ML13196A135]). For Tanks 9 and 10, the annular waste is loaded within the bottom of the annular reducing grout in DOE's PORFLOW™ model (see Figure 1 of RAI-NF-12; Mohseni, 2013a [ML13196A135]). Loading of the annular waste into a thin layer at the bottom of the reducing grout in the model assigns the waste K_d values associated with reducing grout. For redox-sensitive radionuclides, this appears to significantly limit mobility even though this waste is assumed to be highly mobile. In addition, the preferential pathway represented in the deterministic Cases B-E does not intersect the annular waste in Tanks 9 and 10 (see Section 4.2.9.4). For Tanks 14 and 16, the potential release of radionuclides from the sand pads may be limited by the amount of water that contacts the sand pad inventory and/or by diffusion of the radionuclides out of the sand pads (see Section 4.2.9.4). In the HTF PA, DOE assumes that the steel liners in between the sand pads in these tanks are not barriers to flow. However, the preferential pathway is modeled as occurring above the sand pads in the contaminated zone (see Figure 2 of RAI-NF-13; Mohseni, 2013a [ML13196A135]).

In response to RAI-NF-12 and RAI-NF-13 in SRR-CWDA-2013-00106, Rev. 1, DOE discusses that it does not expect the waste inventories associated with the annuli or sand pads to contribute significantly to risk. DOE acknowledges that model simplification results in unintended consequences, such as diffusion of contaminants from the annulus and sand pads into the waste tank grout. DOE believes that these modeling assumptions will maximize overall HTF peak doses as the contamination outside of the primary liners is essentially added to the in-tank inventories (SRR-CWDA-2013-00106, Rev. 1). DOE provides additional analyses using the GoldSim™ model in deterministic mode to evaluate the risk from the annular inventory (Response to RAI-NF-12) and the sand pad inventory (Response to RAI-NF-13). These analyses use modeling assumptions considered by DOE to be pessimistic and identified as "Flow Run 65, No Holdup", which includes tank liner failure at time zero, a fast flow path through the tank system, fast hydraulic degradation of cementitious materials, and no closure cap. Based on the results of these analyses, DOE concludes that the annulus and sand pad inventories are not risk significant.

DOE also provides additional information to clarify the as-modeled release from the annulus and sand pad inventories (SRR-CWDA-2013-00144, Rev. 0). In review of DOE's model data, the annulus and sand pad radionuclide inventories appear to be primarily migrating into the basemat rather than through a preferential pathway. Based on historical evidence of waste release from Tank 16 into the environment (DP-1358) and operational observations of groundwater in-leakage over a relatively short timeframe (SRR-ESH-2013-00078), the NRC staff is concerned that radionuclides could migrate out of the vaults through a fast pathway early in the performance period. Grouting of the tanks and annuli will help limit the presence of preferential pathways and the hydraulic head associated with the fully and partially submerged tanks; however, grouting will not necessarily eliminate flow. Because of the activity of radionuclides in the annulus and sand pads (e.g., Cs-137, Sr-90) and the high solubility of this

waste (Section 3.4.2.2; SRR-CWDA-2010-00128, Rev. 1) even minimal flow through the grouted annuli and sand pads could result in a significant dose.

The NRC staff is concerned that risk from the radionuclide inventories outside of the primary liners is not adequately accounted for in DOE's analyses. The NRC staff recommends that DOE conduct a more comprehensive analysis of the potential release of radionuclides from the annuli and sand pads in the Type I and Type II tanks. Implementation of this recommendation is necessary for the NRC staff to have reasonable assurance that the performance objectives in 10 CFR Part 61, Subpart C can be met. Dose projections from the potential release of the radionuclides in the annuli and sand pads are likely to be very sensitive to several key assumptions, which should be well supported. These assumptions include, but are not limited to (1) the assumed release scenario; (2) the chemical composition of the infiltrating water; (3) the volumetric flow rate through grouted tanks, including shrinkage gaps and cracks; and (4) the solubility of the annulus and sand pad waste. If the possibility of rise and fall of the water table in the vicinity of the Type I and II tanks cannot be excluded, DOE should evaluate a scenario where water drains from any gaps in the annulus and sand pad regions.

4.2.9.4 NRC Evaluation of Near-Field Flow and Transport Modeling

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

For the fully and partially submerged tanks, DOE assumes that the water contacting the CZ consists of 90 percent unconditioned groundwater and 10 percent grout conditioned water for Conditions B, C, and D. The assumption of 10 percent mixing of grout conditioned water provides a significant chemical barrier to radionuclide release. However, the presence of engineered barriers at the HTF calls into question the conservatism of this assumption. The closure cap and the grouted tanks both decrease the likelihood that 10 percent of the water contacting the contamination zone will be conditioned by the overlying grout. The presence of the closure cap is assumed to limit water migrating into the partially submerged tanks early in the performance period. In addition, the low hydraulic conductivity of the grout relative to the high potential hydraulic conductivity of preferential pathways also calls into question the assumption that 10 percent of the water that contacts the contamination zone will be conditioned by the overlying grout. In response to RAI-NF-4 (SRR-CWDA-2013-00106, Rev. 1), DOE provides several lines of reasoning to support the assumption that some of the water interacting with the CZ will be conditioned.

Included in the arguments that water interacting with the CZ will be conditioned is the premise that the engineered barriers (i.e., closure cap, tank vault, grout) would not significantly disturb the natural system ratio of 90:10 and the CZ will be conditioned by the overlying grout. DOE also reiterates HTF PA analyses of the impact of varying the grout transition time and the number of pore volumes required for chemical transition. These analyses show only minor sensitivity of the HTF PA results to these parameter changes when applied to the base case, whereas the results indicate an exceedance of the performance objectives within 10,000 years when applied to alternative Case C. The NRC staff is concerned that DOE has developed an insufficient technical basis for the expected variation in the ratio of unconditioned to conditioned groundwater and the impact of those changes on the chemical transition times. Accordingly, the NRC staff remains concerned that the presence of engineered barriers and preferential pathways may reduce the conditioning of infiltrating water by the overlying grout. The NRC staff

recommends that DOE provide more support for the assumption that the engineered system will not interfere with the ability of the overlying grout to sufficiently condition the infiltrating water for the fully and partially submerged tanks.

In addition to the assumed 10 percent mixing of grout conditioned water for the fully and partially submerged tanks, DOE relies on a low assumed dissolved oxygen concentration in the groundwater to extend the transition time from reducing to oxidizing chemical conditions in the CZ. As stated in SRNL-STI-2012-00404, Rev. 0, "[t]he reason for the longer transition time compared to the non-submerged tanks is the low dissolved oxygen concentration (3.8E-5 molar) in the background well nearest the H-Tank Farm." The assumed low dissolved oxygen in groundwater prolongs the reducing conditions for the fully and partially submerged tanks relative to the non-submerged tanks. DOE relies on data from well P27D, because it is currently the only well in HTF area with measured dissolved oxygen values. However, well P27D is anomalously low relative to the other SRS water table wells. DOE states that the low dissolved oxygen is due to local geology and that the values are likely reasonable for Type I tanks, however, DOE expects the dissolved oxygen values to be higher for Type II tanks (Shaffner, 2013b [ML13126A127]). The NRC staff is concerned that several factors could have resulted in well P27D dissolved oxygen values being lower than groundwater conditions within the HTF, including screening depth and locally impacted groundwater conditions. DOE provided a base case sensitivity analysis in response to RAI-NF-5 evaluating the impact of the ground water dissolved oxygen concentration using dissolved oxygen values from other SRS water table wells (SRR-CWDA-2013-00106, Rev. 1). DOE relies on this analysis to conclude that the lower dissolved oxygen concentrations are reasonable; however, the NRC staff has multiple concerns with the conceptualization and parameterization of the base case, which limits the information provided by the sensitivity analysis. The NRC staff recommends that DOE include dissolved oxygen concentrations in their modeling that are consistent with measurements of unimpacted groundwater across SRS or collect additional dissolved oxygen measurements within the HTF at locations and elevations that are in closer proximity to the tanks.

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

Natural and geo-engineered material properties assumed for near-field modeling are based on data obtained from E-Area (WSRC-STI-2006-00198, Rev. 0). Site-specific vadose zone field data from the HTF should be acquired and analyzed to support material property estimates for the HTF vadose zone. The vadose zones at other GSA locations (e.g., F- and E-Areas) are characterized by sandier soils than that at H-Area; at H-Area, these sandy soils are currently located below the water table in the UTRA-UZ. Instead, limited available evidence suggests that the H-Area vadose zone consists of finer-grained sediments that have only minor representation at other GSA areas. The NRC staff will evaluate the results of any additional confirmatory work performed by DOE during the monitoring period.

DOE's use of gravel characteristic curves to model fast flow (Cases B–E) through intact grout may not adequately reflect the physics of crack flow. As the NRC staff discusses in the FTF TER, flow through porous media is fundamentally different than crack flow and applying material properties of a continuous porous material (e.g., gravel) to a shrinkage gap or crack may under-represent flow through a grout monolith (Camper, 2011 [ML112371751]). Representation of flow through shrinkage gaps and cracks is a challenge due to 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 crack flow, model support (e.g., crack flow experiments coupled with field-scale observations) may be necessary. Furthermore, DOE has not provided a technical basis for their assumption that for Cases B and D, which have fast flow paths, infiltrating water will flow through the grout matrix in such a manner that allows the full volume of the tank grout and its reducing capacity to influence water chemistry prior to the water reaching the contaminated zone.

In addition, DOE assumes an MCC for the sand pad layers within the Type II tanks that is based on a naturally-occurring GSA sand. This characteristic curve does not appear to be representative of the procured sand. DOE's current model assumes that these sand pads are saturated, and therefore, the MCCs do not influence flow (Shaffner, 2013a [ML13106A338]). However, if future models consider unsaturated conditions in the sand pads, then DOE may need additional support for the assumed relative permeability of the sand layers under unsaturated conditions.

As discussed in Section 4.2.8.3 of this TER, in addition to the tank inventories contained within the primary liners, there is relatively soluble contamination located outside of the primary liners in the Type I and Type II tanks. In RAI-NF-12 and RAI-NF-13, the NRC staff questioned whether the risk from these radionuclide inventories is adequately addressed within the HTF PA (Mohseni, 2013a [ML13196A135]).

Despite the relatively short transport pathway in the concrete basemat, retardation of neptunium and plutonium may have a significant effect on dose. In the FTF PA and the supplemental analyses, the basemats were shown to be an important barrier in attenuating the releases of key radionuclides (e.g., plutonium, neptunium). Although solubility control may be more important for plutonium and neptunium in DOE's HTF PA, compared to the FTF PA, the NRC staff also expects the HTF basemats to be important. PORFLOW™ model output from DOE's analysis of annulus and sand pad contamination indicates that the basemats significantly attenuate the release of Cs-137, Sr-90, and potentially Pu-239, even in the cases with preferential pathways. In the barrier analysis section (Section 4.218.5 of this TER), the NRC staff discusses the importance of the assumed basemat performance in attenuating the release of key radionuclides. Cementitious material K_d values for these elements are, therefore, important.

In addition, cementitious K_ds may also affect tank release if radionuclides diffuse upward through the grout in the annulus or the grout overlying the CZ. The adopted, very high K_d values for neptunium and plutonium in cementitious materials (Table 4.2-29; SRR-CWDA-2010-00128, Rev. 1) are subject to uncertainty related to the possibility of solubility control during sorption experiments (FTF RAI-NF-7;SRR-CWDA-2011-00054, Rev. 1; SRNL-STI-2009-00473, Rev. 0). DOE responses to FTF RAIs have not yet established that the high K_d values they have adopted are reasonable for representing sorption rather than solubility; furthermore, literature syntheses DOE uses to justify its selected values (e.g., Tables 17 and 18; SRNL-STI-2009-00473, Rev. 0) reveal a diversity of experimental results that include K_d values significantly lower than the adopted values.

These factors demonstrate that cementitious material K_ds for neptunium and plutonium 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 (FTF RAI-NF-12; SRR-CWDA-2011-00054, Rev. 1). DOE needs to provide more support for the assumptions that the basemat will remain intact and will provide long-term hydraulic and chemical performance at HTF. The NRC staff also discusses this need under Monitoring Factor 3.5 for FTF (Camper, 2013a [ML12345A322]).

In the HTF PA, DOE assigns neptunium K_ds for cementitious materials that are much higher than are used for the FTF PA: 10,000 mL/g for Reduced Region II, 10,000 mL/g for Oxidized Region II, and 5,000 mL/g for Oxidized Region III (Table 4.2-29; SRR-CWDA-2010-00128, Rev 1). The corresponding values for the FTF PA are 3,000, 1,600, and 250 mL/g (Table 4.2-33; SRS-REG-2007-00002, Rev. 1). The cited reference for the HTF PA values is SRNL-STI-2009-00473, Rev. 0 and Tables 17 and 18 in that report contain the rationales for these values. The technical bases provided in those tables do not appear to support the establishment of these new values for modeling neptunium transport in cementitious materials.

For middle-age reducing conditions (i.e., Reduced Region II), the discussion in SRNL-STI-2009-00473, Rev. 0, Table 18, does not clearly support the use of 10,000 mL/g. Several cited measurements are well below 10,000 mL/g and higher values are attributed to solubility control during the experiments. In response to a question from the NRC staff on the basis for this value, DOE provides, in SRR-CWDA-2013-00086, Rev. 0, the text from Table 17 of SRNL-STI-2009-00473, Rev. 0, which was presumably written to support K_ds for oxidizing conditions. Like Table 18 from SRNL-STI-2009-00473, Rev. 0, the Table 17 text includes discussion of some values substantially below 10,000 mL/g without discounting their usefulness. Based on the DOE response to a question on oxidized neptunium K_ds in SRR-CWDA-2013-00086, Rev. 0, the apparently correct reference for all the neptunium K_d values is SRNL-STI-2009-00636. That report does recommend 5,000 mL/g for old-age oxidizing conditions (i.e., Oxidized Region III), and states that 10,000 mL/g should be used for the other two conditions (Reduced Region II and Oxidized Region II) when the concentration is below the "apparent" solubility limit of 10^{-13} to 10^{-12} M. However, SRNL-STI-2009-00636 argues that the measured neptunium aqueous concentrations were controlled by solubility, not sorption, and does not provide clear support for the adopted 5,000 and 10,000 mL/g values. A K_d that is based on experiments in which solubility was controlling the aqueous neptunium concentration is not appropriate for modeling sorption.

Similarly, the arguments for plutonium K_ds in SRNL-STI-2009-00473, Rev. 0 mention that very high K_ds can be attributed to solubility control. Experimental work reported in SRNL-STI-2009-00636 concludes that its high apparent K_ds resulted from solubility control of plutonium, rather than sorption. The same arguments above questioning the use of very high measured neptunium K_ds in transport modeling apply also to plutonium. In addition, the HTF PA (Table 4.2-29; SRR-CWDA-2010-00128, Rev 1) lists K_ds for Pu, Pu_4, and Pu_5. The HTF PA (SRR-CWDA-2010-00128, Rev. 1), however, does not make clear how the different oxidation state values are used. The NRC staff has previously questioned DOE's practice of linearly combining plutonium K_ds for the natural system for different oxidation states to calculate a combined value. The PA report does not state whether this approach is also taken for cementitious materials in the HTF PA. The NRC staff will continue to monitor activities related the determination and use of neptunium and plutonium K_ds for cementitious materials.

With Ra-226 being a significant dose contributor, the cementitious materials radium K_ds (Table 4.2-29; SRR-CWDA-2010-00128, Rev 1) are potentially important. Very few experimental studies applicable to SRS cementitious materials are available, and the discussion supporting the HTF values in SRNL-STI-2009-00473, Rev. 0 does not provide clear evidence. (For example, one literature reference is incorrect and another is not available to the NRC staff.) The NRC staff believes the topic of radium sorption to cementitious materials merits continued attention.

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 FTF CC-NF-8 (SRR-CWDA-2011-00054, Rev. 1) that it is reasonable to assume that the variability in sorption characteristics of a relatively homogenous material like concrete or grout would be lower than variability in natural sedimentary rocks. Therefore, DOE's application of uncertainty terms for natural system K_ds to cementitious material K_ds likely overestimates uncertainty in the cementitious materials. The NRC staff concludes DOE should further consider the appropriateness of using the sediment statistics to define cementitious material K_ds uncertainty (i.e., as more data become available). In addition, as discussed later in this section, the NRC staff thinks more analyses are needed to refine the statistical approach to constraining K_d uncertainty.

The NRC staff evaluated the information on natural system K_d values presented in the HTF PA (Sections 4.2.2.2.2 and 4.2.2.2.7, Table 4.2-25; SRR-CWDA-2010-00128, Rev. 1) and supporting documents. The NRC staff notes that the correct reference for the radium K_ds is likely SRNL-STI-2011-00011, Rev. 0, rather than the report noted on Table 4.2-25. 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 staff's evaluation relied, in part, on comparisons to an NRC-sponsored compilation of site-specific K_d measurements for SRS (Prikryl and Pickett, 2007 [ML073510127]). The overall DOE approach to establishing natural system K_ds is appropriate—employing site-specific data when possible and relying on literature data when site-specific data are lacking or sparse. The NRC staff finds, in general, that the values used for the deterministic analysis are reasonable, but still has concerns about the treatment of neptunium and plutonium.

For each sediment type, DOE defines plutonium K_ds of 5,950 mL/g for backfill/clayey soil and 650 mL/g for vadose zone/sandy soil (Table 4.2-25; SRR-CWDA-2010-00128, Rev. 1). The clayey soil value is unchanged from what is used for the FTF PA, that is, a linear combination of values was determined for Pu(III/IV) and Pu(V/VI). While the NRC staff has concerns regarding the combining of oxidation state K_ds, in this case the effect is not major because the two values are 5,000 and 6,000 mL/g. The Pu(V/VI) clayey sediment value of 5,000 mL/g in SRNL-STI-2009-00473, Rev. 0, however, appears too high. Prikryl and Pickett (2007 [ML073510127]) refer to a number of studies that yielded significantly lower values. The Pu(III/IV) K_d of 6,000 mL/g for clayey sediments is consistent with the Prikryl and Pickett (2007 [ML073510127]) evaluation. The question, therefore, becomes whether 5,950 mL/g is an appropriate value for bulk plutonium in clayey sediment. The argument in SRNL-STI-2009-00473, Rev. 0 that, given a relatively short amount of time, virtually all plutonium in SRS soils will convert to Pu(IV) is compelling. Even short-term experiments with clayey sediments summarized in SRNL-STI-2009-00473, Rev. 0 yielded K_ds on the order of 10^3 mL/g. A summary of plutonium sorption

data on SRS sediments gives mean values greater than 5,950 mL/g overall and for pH above and below 7 (SRNL-STI-2011-00672), but the standard deviation is very large.

Under Monitoring Factor 4.1 in the FTF Monitoring Plan (see Appendix E; Camper, 2013a [ML12345A322]), the NRC staff discusses DOE's assumed plutonium K_d for sandy sediment. The sandy sediment K_d for plutonium of 650 mL/g is derived from SRNL-STI-2011-00672. This study bases the recommended value on (1) information from a modeling analysis (Demirkanli et al., 2007) of long-term lysimeter studies (Kaplan et al., 2006) indicating that the K_d should be 1,800 mL/g, and (2) the site-wide statistical analysis showing that the 290 mL/g value used for the FTF PA is in the lower quantiles. The sediment in the lysimeter appears to have had more clay in it than typically found at the FTF location, therefore, the 1,800 value was lowered to 650 mL/g. The NRC staff does not find the argument for the 650 mL/g to be well supported, and thinks it does not account for different oxidation states as discussed in the following paragraph.

The NRC staff acknowledges, as was written in the FTF TER (Camper, 2011 [ML112371751]), that experimental evidence summarized in SRNL-STI-2009-00473, Rev. 0 suggests that longer-term sorption studies show conversion of large amounts of initially oxidized plutonium into Pu(IV) in sediment-water systems. Nevertheless, because even small quantities of mobile plutonium could have dose impacts, the NRC staff still has concerns about the use of single hybrid K_ds for plutonium in SRS soils and sediments. The NRC staff still thinks a more accurate representation of the transport of multivalent plutonium would be to treat the two species separately, assuming the oxidation state distribution could be reasonably quantified. The NRC staff will follow this technical issue and DOE's efforts to address this issue during the monitoring period.

DOE's use of a cementitious leachate impact factor to adjust vadose zone K_ds (Table 4.2-25; SRR-CWDA-2010-00128, Rev. 1) is established in SRNL-STI-2009-00473, Rev. 0 and is a good first step to addressing the chemical effects on transport of tank grouts. DOE calculates cementitious leachate impact factors using Hanford site data. Applying the Hanford site data to calculate the factors for SRS (SRNL-STI-2009-00473, Rev. 0) is not, however, justified without an element-by-element analysis of the chemical processes affecting sorption. The Hanford site geological and geochemical environment contrasts sharply with SRS. Such element-specific considerations are applied in SRNL-STI-2009-00473, Rev. 0 (Table 2) to some, but apparently not all, elements. While DOE acknowledges that this approach is for "early guidance until appropriate measurements can be made," (Page 46; SRNL-STI-2009-00473, Rev. 0), the NRC staff has concerns about some aspects of the set of values. For example, Hanford site factors were adopted for the potentially important elements radium and neptunium apparently without a more detailed analysis. Some factors apparent from the values in the HTF PA (Table 4.2-25; SRR-CWDA-2010-00128, Rev. 1), do not follow the guidance in SRNL-STI-2009-00473, Rev. 0. The deviations from SRNL-STI-2009-00473, Rev. 0 (e.g., uranium and plutonium) do tend to vary in the conservative low-K_d direction. Nevertheless, DOE should explain in future documents the rationales for specific deviations from the originally recommended factors.

The application of cementitious leachate impact factors to the actinides uranium, neptunium, and plutonium is worthy of more detailed comment. For each element, except for plutonium in sandy sediment, the factor is greater than 1. A factor of 0.9 is used for plutonium in sandy sediment, while 2 is used for clayey sediment. In fact, as explained in the DOE response to CC-NF-3 (SRR-CWDA-2013-00106, Rev. 1), the Hanford site data suggests a factor of 0.25 for

plutonium and the SRS investigators elected to use higher values. In justifying this choice, DOE's response argues that the preponderance of evidence shows that plutonium solubility decreases with pH and that plutonium will precipitate out of solution. A similar argument is made for uranium in SRNL-STI-2009-00473, Rev. 0.

Two points can be made with respect to this argument. First, while precipitation due to the solubility of a solid phase can be considered one of many processes under the general term "sorption," the use of a K_d implements reversible adsorption to surfaces in a model and does not depend on solubility. Selection of a K_d must be independent of solubility considerations; in fact, an experiment that observes precipitation is invalid for establishing a K_d. Therefore, the argument that solubility decreases with increasing pH has no bearing on how a K_d is selected.

Second, the sorption behavior of these actinides at high pH in unsaturated conditions (i.e., in communication with air) may strongly depend on the aqueous activities of carbonate species. Sorption coefficients may, in fact, decrease at elevated pH if high pCO_2 promotes the stability of actinide carbonate or hydroxy-carbonate species (Turner et al., 2002). The determination of cement leachate impact factors needs to explicitly consider the potential for decreased K_d as pH increases, as a function of aqueous speciation. The NRC staff will continue to interact with DOE on the topic of cement leachate sorption impact as DOE information and models evolve.

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

The NRC staff questioned DOE's conceptual model for the nature of flow through HTF tanks (i.e., slow bulk matrix flow as in the base case (Case A) versus preferential fast flow (Cases B–E), discussed in Section 4.2.18 of this TER. DOE assigns a probability of 75 percent to the base case (Table 5.6-5; SRR-CWDA-2010-00128, Rev. 1), assuming that preferential pathways are not likely to occur throughout the entire period of performance. In response to RAI-NF-10, DOE discusses that the modeling approach (e.g., alternative fast flow modeling configurations, reasonably conservative modeling assumptions) is appropriate for understanding the risk from the HTF (SRR-CWDA-2013-00106, Rev. 1). However, the NRC staff is concerned that the probability of water migrating into the tanks through preferential pathways in the concrete vaults and contacting the waste is greater than assumed in the HTF PA based on (1) historical evidence of waste release from Tank 16 into the environment (DP-1358), (2) operational observations of groundwater in-leakage over a relatively short timeframe (SRR-ESH-2013-00078), and (3) the presence of transfer lines, which DOE does not plan to fill with grout and may, therefore, act as fast pathways into the tanks (CC-NF-4; Mohseni, 2013a [ML13196A135]). Grouting of the tanks and annuli will help limit the presence of preferential pathways; however, grout shrinkage and degradation are likely to result in preferential pathways. The NRC staff agrees with the authors of LA-UR-12-00079, who state that preferential flow through cracks is a much more likely scenario than matrix flow. Based on the risk significance of the conceptual model and waste release uncertainty in DOE's PA, the 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 (see Monitoring Factors 2.2, 3.2, and 3.3; Camper, 2013a [ML12345A322]). The NRC staff will continue to evaluate these uncertainties during monitoring in its assessment of compliance of the HTF with the performance objectives in 10 CFR Part 61, Subpart C as tank farm closure progresses.

In addition to the tank inventories contained within the CZ, the NRC staff raises concerns about the risk from the inventories located within the annuli and sand pads of the Type I and II tanks (Mohseni, 2013a [ML13196A135]). The preferential pathway represented in the deterministic Cases B-E does not intersect the annulus waste in Tanks 9 and 10 (see Figure 1 of RAI-NF-12; Mohseni, 2013a [ML13196A135]). Also, it is not clear in the HTF PA the extent to which the preferential pathway in Cases B-E interacts with the contamination located in the sand pads of the Type II tanks. The preferential pathway is modeled as occurring above the sand pads in the contaminated zone (see Figure 2 of RAI-NF-12; Mohseni, 2013a [ML13196A135]). As part of the response to RAI-NF-12 and RAI-NF-13, DOE provides additional analysis using modeling assumptions identified as "Flow Run 65, No Holdup" (SRR-CWDA-2013-00106, Rev. 1), which includes tank liner failure at time zero, a fast flow path through the tank system, fast hydraulic degradation of cementitious materials, and no closure cap. Based on the results of these analyses, DOE concludes that the annulus and sand pad inventories are not risk significant. The NRC staff's review of DOE's model data provided in SRR-CWDA-2013-00144, Rev. 0 indicates that the annulus and sand pad inventories are primarily migrating into the basemat rather than through a preferential pathway. Based on the historical and operational evidence cited in the preceding paragraph, the NRC staff is concerned that radionuclides could migrate through a fast pathway early in the performance period. Grouting of the tanks and annuli will help limit the presence of preferential pathways and the hydraulic head associated with the fully and partially submerged tanks; however, grouting will not necessarily eliminate flow. Because of the activity of key short-lived (e.g., Cs-137, Sr-90) and long-lived radionuclides (e.g., Pu-239) in the annulus and sand pads and the high solubility of this waste (Section 3.4.2.2; SRR-CWDA-2010-00128, Rev. 1) even minimal flow through the grouted annuli and sand pads could result in a significant dose. The NRC staff is concerned that risk from the radionuclide inventories outside of the primary liners is not adequately accounted for in DOE's analyses. As discussed in more detail in Section 4.2.9.3, the NRC staff recommends that DOE conduct a more comprehensive analysis of the potential release of radionuclides from the annuli and sand pads in the Type I and Type II tanks.

With respect to transport and the GoldSim™ stochastic analysis, lognormal uncertainty distributions for sorption coefficients (Section 5.6.3.4; SRR-CWDA-2010-00128, 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 were measured for radioisotopes of Cd, Ce, Co, Cs, Hg, Sr, and Y in 27 samples that represent the upper vadose, lower vadose, and aquifer zones. DOE recommends an upper vadose zone range for clayey soils and a range for sandy soils that was an average of those for the E-Area lower vadose zone and aquifer (in H-Area, the "lower vadose zone" material occurs below the water table in the UTRA-UZ (SRNL-STI-2009-00150, Rev. 1; Shaffner, 2013b [ML13126A127]). DOE's HTF PA (Section 5.6.3.4; SRR-CWDA-2010-00128, Rev. 1) adopts minimum and maximum multipliers based on material type for the lognormal distributions and also calculates standard deviations based on multipliers of the geometric mean (Table 5.6-13; SRR-CWDA-2010-00128, Rev. 1).

The 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, 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 all of the risk-significant species in DOE's HTF PA results. Neptunium is 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) represent the radionuclide-specific distributions. The complex and potentially variable behavior of the key elements such as plutonium and technetium 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 were obtained from a single borehole in E-Area (WSRC-STI-2008-00285) and an argument is not made in later reports that the analysis is applicable to the transport pathways in the HTF PA (SRNL-STI-2009-00150, Rev. 1; Section 5.6.3.4 of SRR-CWDA-2010-00128, 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. DOE noted during the FTF PA review (CC-NF-8 response SRR-CWDA-2011-00054, Rev. 1) that Section 8.2 in the FTF PA (SRS-REG-2007-00002, Rev. 1) states that DOE will continue to refine their representations of uncertainty in future stochastic analyses. The 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 HTF PA.

4.2.9.5 Near-Field Review Results and Recommendations

The NRC staff performed a comprehensive review of DOE's HTF PA documentation and supporting references related to near-field modeling. Primary review results and recommendations are presented in this section. The 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 the NRC staff notes large uncertainties associated with modeling degradation processes over the long time periods relied on for performance in DOE's HTF PA.

- Major corrosion processes appear to be considered in the steel liner degradation modeling; however, DOE's assumptions regarding the performance of cementitious materials that are relied on as a barrier to steel liner corrosion may be overstated and the importance of aggressive corrosion processes understated in DOE's HTF 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 could be 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 erroneous 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's assumptions regarding solubility limiting phases, solubility limits, and chemical transition times are risk-significant and have not been confirmed through waste characterization and experimentation.

- DOE's approach to modeling waste release from the Type I and Type II tanks does not adequately assess the risk from the contamination located within the annuli and sand pads.

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

The NRC staff concludes that DOE's near-field model presents an acceptable framework to facilitate decision-making regarding HTF closure. However, characterization of DOE waste tank residuals and observation of shorter- and longer-term performance of key components critical to DOE's compliance demonstration (i.e., tank liners) is lacking. The NRC staff recognizes the difficulty in obtaining additional information to reduce technical uncertainties identified in the NRC staff's review. However, the NRC staff has identified technical areas where additional information could 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.

1. As a primary recommendation, the NRC staff recommends DOE conduct waste release experiments to:

    a. Increase experimental support for key modeling assumptions related to the behavior of tank fill grout over time, including the evolution of pH and E_h (High Risk Significance, Short-to-Intermediate Term);

    b. 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);

    c. Determine constant concentrations of elements of concern under conditions of exposure to local groundwater and grout leachate via static tests (High Risk Significance, Short and Intermediate Term); and

    d. Distinguish between releases from high solubility compounds and low solubility compounds via semi-dynamic leach tests (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 the formation of new compounds that may alter the leachability of radionuclides.

2. DOE should conduct a more comprehensive analysis of contaminant release from the annuli and sand pads in the Type I and II tanks. This analysis should include well-supported assumptions for (a) the assumed release scenario; (b) the chemical composition of the infiltrating water; (c) the volumetric flow rate through grouted tanks, including shrinkage gaps and cracks; and (d) the solubility of the annulus and sand pad waste (Medium-to-High Risk Significance, Short and Intermediate Term).

3. DOE should continue to evaluate the appropriateness of selected transport parameters (e.g., cementitious material and soil K_ds) during the monitoring period (Medium Risk Significance, Intermediate Term).

4.2.10 Hydrology and Far-Field Transport

The natural, hydrogeological system at H-Area is a significant barrier limiting the impact of releases from the HTF and is, therefore, a major component of the HTF PA[12]. The HTF far-field model is used to simulate groundwater flow and contaminant transport of releases of radionuclides from HTF tanks and ancillary equipment into the natural environment. DOE uses the far-field model to calculate radionuclide concentrations in groundwater at various points of compliance to demonstrate compliance with the performance objectives in 10 CFR Part 61, Subpart C.

DOE uses the 1995 GSA database of hydrostratigraphic and hydrogeologic data to develop three-dimensional groundwater flow and contaminant transport models for the GSA, including the HTF saturated zone or far-field model. This database includes elevations for tops of hydrostratigraphic units, sediment core descriptions, water levels in the UTRA-UZ and UTRA-LZ, water levels in the Gordon Aquifer, and permeability data from laboratory tests, multiple and single well pump tests, and slug tests. However, the GSA database does not incorporate any new data collected at H-Area since 1995—an evaluation of recent data is the subject of SRNL-STI-2010-00128, Rev. 1.[13] The location of the HTF on the GSA (and nearby surface water locations) is depicted in Figure 4-10.

In the area of the HTF, the hydrological system consists of three aquifers of interest: (1) the water table or UTRA, which is split into upper and lower zones, (2) the Gordon Aquifer, and (3) the Crouch Branch Aquifer. Radionuclide releases from HTF are expected to impact the uppermost aquifers, which are the Upper Three Runs and Gordon aquifers. Because an upward flow gradient exists between the Crouch Branch and Gordon Aquifer near HTF (SRNL-STI-2010-00148, Rev. 0), releases are not expected to affect the deeper Crouch Branch Aquifer. Groundwater flow in the UTRA is driven by recharge. Hydraulic heads within the

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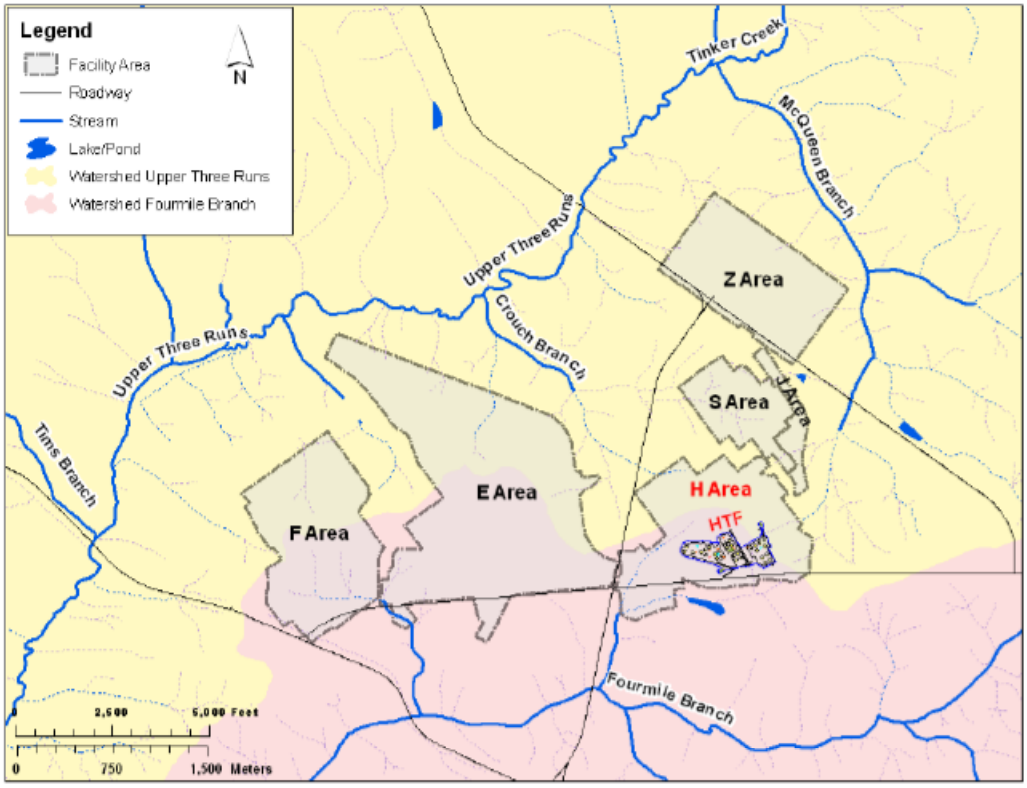
[12] See Section 3.1.5 “Hydrogeology,” Section 4.2.2.1.3 “Saturated Zone,” Section 4.2.2.2.7 “Saturated Zone Hydraulic Properties,” Section 4.3.1.2, “PORFLOW,” Section 4.4.4.1 “PORFLOW™ Modeling Process,” and Section 5.2.1, “Groundwater Concentrations at 100 m” in SRR-CWDA-2010-00128, Rev. 1 for more information on the far-field model used by DOE.

[13] More recent data is reviewed by DOE in SRNL-STI-2010-00148, Rev. 0, to evaluate the adequacy of the GSA/PORFLOW™ model.

Figure 4-10 Location of HTF Within General Separations Area



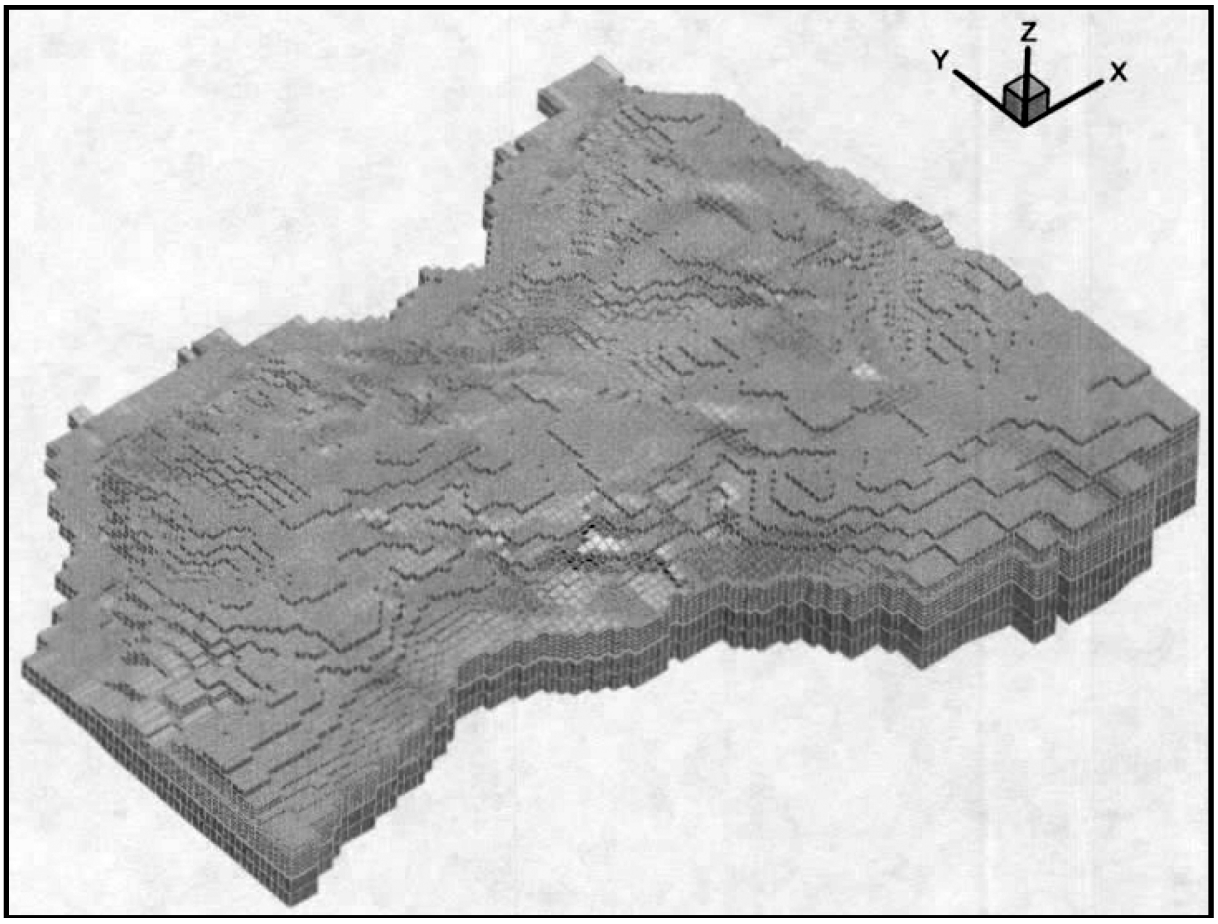
Adapted from Figure 3.1-4 in SRR-CWDA-2010-00128, Rev. 1.

underlying Gordon Aquifer are strongly influenced by discharge into the Upper Three Runs Creek. The Gordon Aquifer is recharged by (1) downward leakage from the UTRA-LZ above and (2) upward leakage from the Crouch Branch aquifer below. Conceptually, upwelling recharge of the Gordon Aquifer by the Crouch Branch Aquifer is very small in comparison to downward recharge from the UTRA-LZ, therefore, this upwelling recharge is neglected in modeling. See Figure 1-7 for an illustration of the GSA hydrogeological conceptual model.

4.2.10.1 Model Construction

DOE previously constructed a regional GSA groundwater flow model using the FACT computer code to provide a common framework to perform various hydrogeological investigations for the GSA, as documented in WSRC-TR-96-0399, Vols. 1 and 2, Rev. 1. The regional GSA flow model was migrated to PORFLOW™ version 5.95.0 in 2004 for use in various GSA PAs as documented in WSRC-TR-2004-00106, Rev. 0. The PORFLOW™ version of the GSA regional flow model is referred to in this TER as the GSA/PORFLOW™ model. Figure 4-11 shows a perspective view of the regional GSA/PORFLOW™ model. The GSA/PORFLOW™ model is used to simulate flow and contaminant transport of constituents released from HTF to surface water. Seepline concentrations are extracted from the GSA/PORFLOW™ model for use in calculating surface water pathway doses in the HTF PA.

Figure 4-11 Perspective View of Regional GSA/PORFLOW™ Flow and Transport Model Domain



Adapted from Figure 2-3 in WSRC-TR-2004-00106, Rev. 0.

The interior areal resolution of the GSA/PORFLOW™ model is 61 m×61 m (200 ft×200 ft); 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 UTRA-UZ is represented by up to 10 cells in the vertical direction; the vadose zone is considered to be part of the UTRA-UZ. The TCCZ is represented by two cells in the vertical direction and is assumed to be laterally continuous. The UTRA-LZ is represented by five cells in the vertical direction. The underlying GCU and Gordon Aquifer are each represented by two 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.

The hydrostratigraphy of the GSA/FACT and GSA/PORFLOW™ models are similar with some notable exceptions (WSRC-TR-2004-00106, Rev. 0). Adjustments to the GSA/FACT model mesh were necessary to overcome limitations of the PORFLOW™ model in accurately representing the velocity field for highly distorted elements present in the FACT model or to

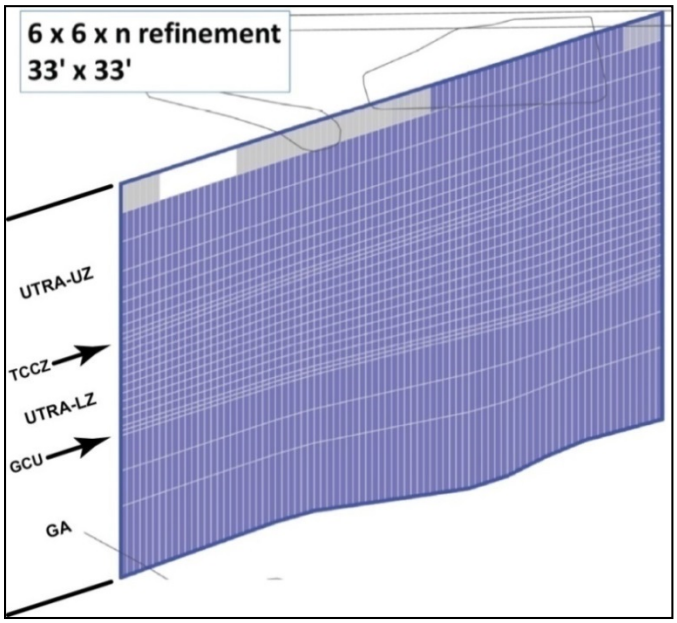
accommodate specific project needs. Most notably, GSA/PORFLOW™ model layers are assumed to be truncated by the ground surface to prevent excessively thin layers that are present in the GSA/FACT model. Additionally, GSA/PORFLOW™ model layers above the TCCZ are non-uniformly distributed compared to their counterparts in the GSA/FACT model. Thinner layers just above the TCCZ were desirable for E-Area modeling where the water table is located just above the TCCZ. The layering below the TCCZ is essentially the same between the two models.

The primary focus of the local HTF/PORFLOW™ saturated zone flow and transport model (Figure 4-12) is contaminant concentrations in wells at 1- and 100-m (3- and 330-ft) compliance points for calculating groundwater pathway doses to the intruder (10 CFR 61.42) and members of the public (10 CFR 61.41), respectively. The areal resolution of the local HTF/PORFLOW™ model is 10.2 m×10.2 m (33.3 ft×33.3 ft), which is attained on the horizontal plane by evenly dividing each GSA/PORFLOW™ model grid cell six ways in both coordinate directions into 36 transport model grid cells (Figure 4-13). The longitudinal horizontal numerical dispersion for this mesh size is approximately 5.1 m (16.7 ft) (WSRC-TR-2004-00106, Rev. 0). In the vertical direction, each GSA/PORFLOW™ model grid layer in the UTRA-LZ was divided into two layers; the upper layer of the Gordon Aquifer was also divided into two layers while the vertical resolution of the remaining layers was preserved (Figure 4-13).

Figure 4-12. HTF/PORFLOW™ Saturated Zone Model Domain Showing Discrete and Distributed Sources, Stream Traces from Waste Tank Sources, and 1-m (3-ft) and 100-m (330-ft) Compliance Evaluation Sectors

Adapted from Figure 5.2-5 in SRR-CWDA-2010-00128, Rev. 1.
Large red polygons represent areal extent of transfer lines.

Figure 4-13 Vertical Slice through HTF/PORFLOW™ Saturated Zone Model



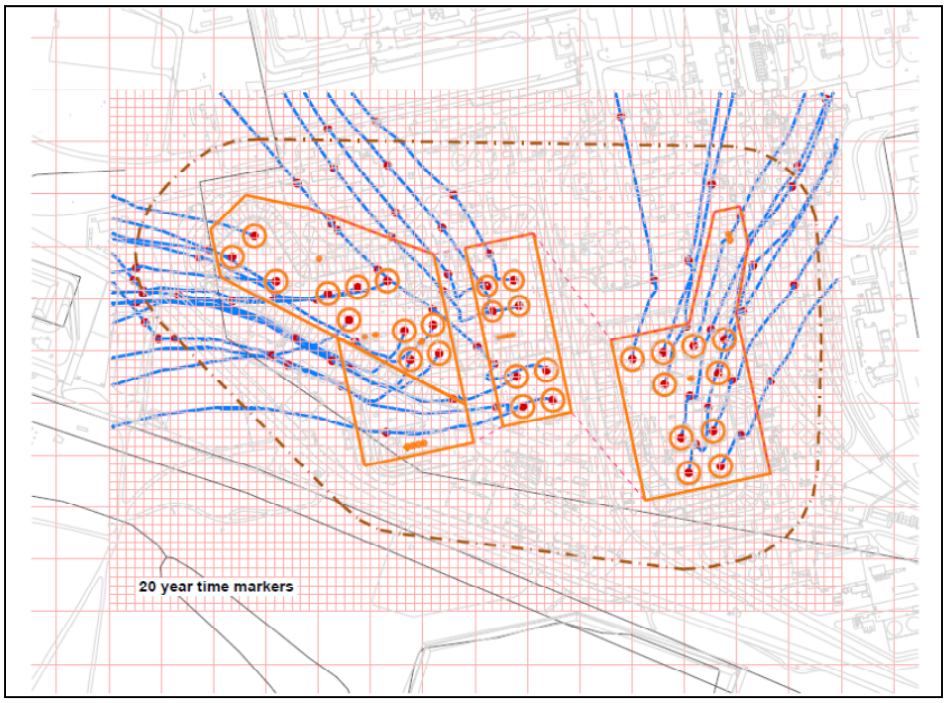
Adapted from Figure 2(b) in SRNL-STI-2012-00465.

The velocity field for the local HTF/PORFLOW™ model is generated with a mass-conserving linear interpolation scheme directly from the regional GSA/PORFLOW™ velocity model; thus, the local HTF/PORFLOW™ model does not necessitate a separate flow model with its own boundary conditions and material properties. Within the lateral confines of the local HTF/PORFLOW™ model, the velocity field includes the complete vertical extent of the regional GSA/PORFLOW™ model. Figure 4-14 displays HTF stream traces with 20-year time markers.

4.2.10.2 Boundary Conditions

The GSA/PORFLOW™ saturated zone model domain encompasses the GSA and its surface water discharge points. The location of H-Area on the GSA and bordering streams are depicted in Figure 4-15. Streams define three lateral domain boundaries: Upper Three Runs Creek forms the northern boundary; McQueen Branch forms the eastern boundary; and Four Mile Branch forms the southern boundary. McQueen Branch and Four Mile Branch provide natural, no-flow boundary conditions for the UTRA. This aquifer unit is absent at the northern model boundary due to Upper Three Runs 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 the Upper Three Runs Creek, so hydraulic head values from a contour map are prescribed over the west, south, and east faces of the model, but the Upper Three Runs 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.

Figure 4-14 HTF Stream Traces and 20-Year Time Markers

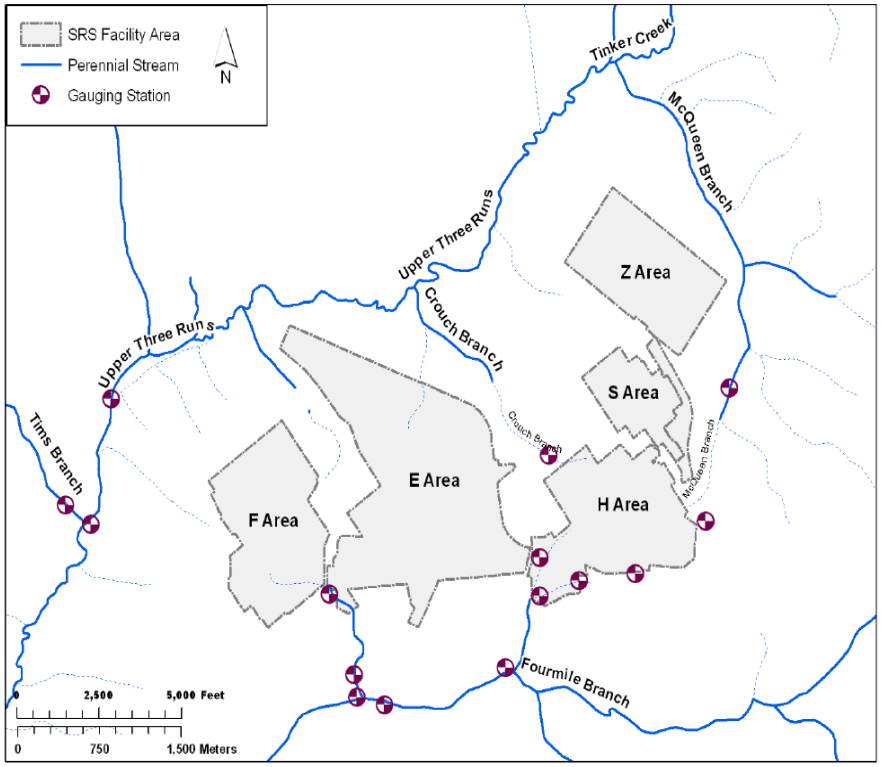


Adapted from Figure 4.4-14 in SRR-CWDA-2010-000128, Rev. 1.

DOE applies a specified flux boundary condition to the upper surface of the GSA/PORFLOW™ saturated zone model (Figure 2-4; WSRC-TR-2004-00106, Rev. 0). A recharge rate of 48.3 cm/yr (19 in/yr) is applied over most of the model domain, although the average recharge rate of 37.3 cm/yr (14.7 in/yr) is lower as it includes drainage, as well as recharge (Section 3.2; WSRC-TR-2004-00106, Rev. 0). Although the closure cap is assumed to reduce infiltration through the disposal system for thousands of years, the local infiltration-reducing effect of the closure cap is not considered in the far-field model. The potential impacts of the closure cap on the HTF flow field is evaluated in separate documentation (Portage, 2008). During a technical exchange, DOE posited that the closure cap could cause preferential flow of contaminants downward instead of laterally away from the site because closure cap runoff water is anticipated to infiltrate at the cap edge and partially move toward the center of the HTF footprint. However, the cap should also locally flatten the groundwater divide beneath the tank farm, thus somewhat reducing the vertical hydraulic head gradient (Shaffner, 2013c [ML13154A327]).

Contaminant fluxes that exit the individual near-field models (i.e., one each per waste tank, distributed transfer line source region, pump tank, and evaporator pot) are applied to the local HTF/PORFLOW™ saturated zone transport model. The radionuclide contaminant fluxes are assigned to the HTF/PORFLOW™ model mesh by (1) uniform distribution to water table grid cells with centroids lying within the footprint of the waste source, or (2) to lower elevation cells representative of the basemat location for submerged Type I and II tanks (Shaffner, 2013c [ML13154A327]). Figures RAI-NF-14.1 and RAI-NF-14.2 (SRR-CWDA-2013-00106, Rev. 1) illustrate horizontal and vertical source loading locations. Based on Figure 4-12, the fewest and largest number of grid cells representing the footprint of an individual waste tank is one and four, respectively, with a small majority of them represented by two cells. However, the NRC

Figure 4-15 Map View Context Image of the General Separations Area and H-Area With Emphasis on Creeks That Form Model Domain Boundaries



Adapted from Figure 1 in WSRC-TR-96-0399, Vol. 2, Rev. 1.

staff plotted the source cells obtained from HTF/PORFLOW™ input files provided by DOE and it appears that the number of source cells is significantly greater, ranging from 4 to 6 as discussed in Section 4.2.11.[14]

4.2.10.3 Material Properties and Parameters

The regional GSA/PORFLOW™ model used for the HTF PA generally inherits its parameter set from that of the predecessor GSA/FACT model. The GSA/FACT parameter set was derived from the GSA database of hydrologic data, which includes hydraulic conductivity data from laboratory tests, pump tests (multiple and single well tests), and slug tests. The same approach used in the GSA/FACT modeling to assign initial hydraulic conductivity fields was also used during GSA/PORFLOW™ model construction.

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

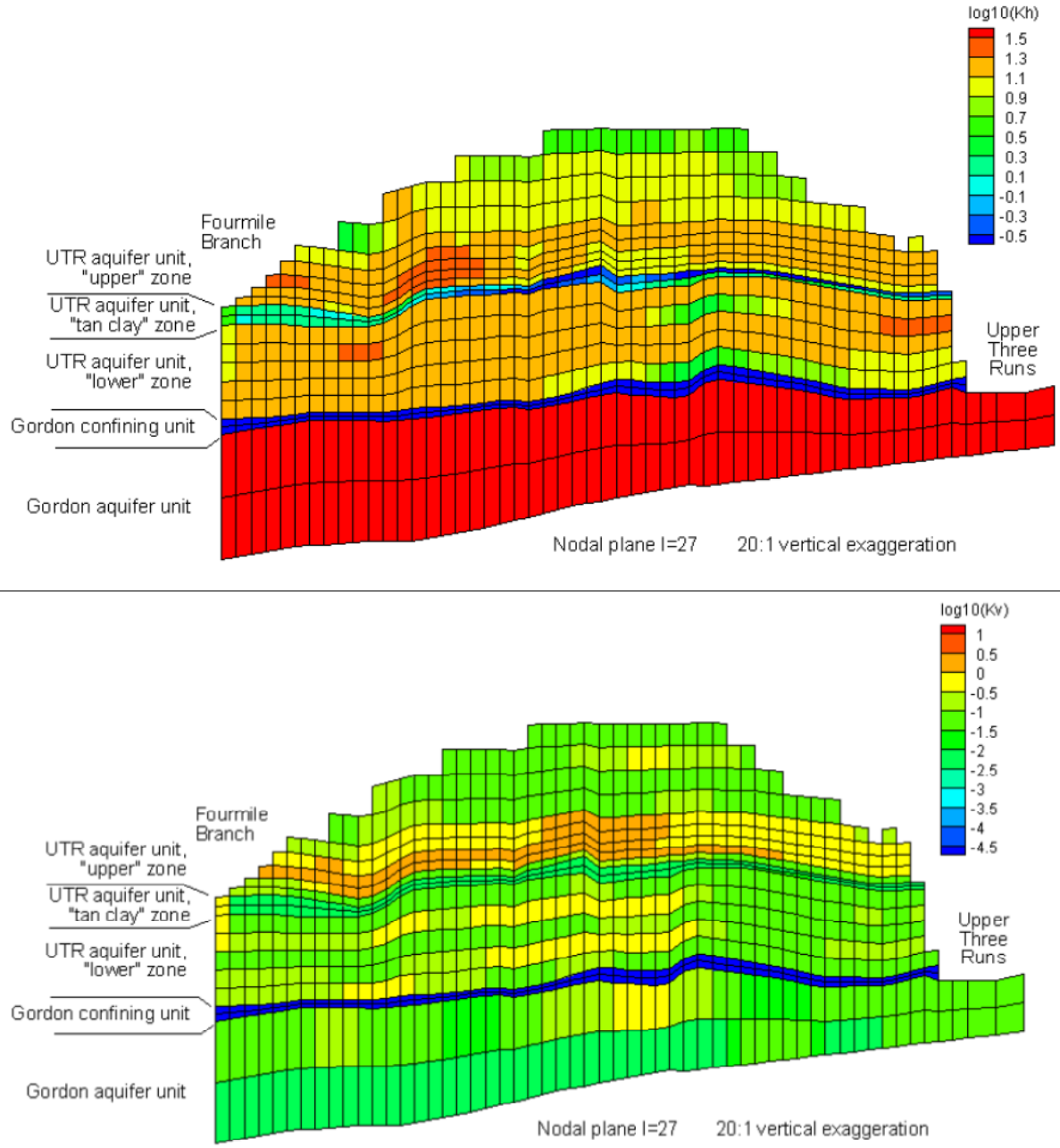
[14] The DOE response to RAI-NF-14 confirms that 4 to 6 source cells were modeled in PORFLOW (SRR-CWDA-2013-00106, Rev. 1).

and siliciclastic), where mud is the summation of silt and clay fractions (see corrected Figures CC-FF-5.1 and -5.2;SRR-SWDA-2011-00054). Known core lithologies (e.g., mud fraction) from discrete intervals were correlated with known field- or laboratory-measured hydraulic conductivities from the same intervals (Shaffner, 2013c [ML13154A327]). There were slight differences in the initial definition of GSA/FACT and GSA/PORFLOW™ model hydraulic conductivity fields due to differences in the model meshes and manner by which property assignments are made (i.e., at cell nodes in FACT versus on cell faces in PORFLOW), (Section 2.3; WSRC-TR-2004-00106, Rev. 0). The initial GSA/PORFLOW™ hydraulic conductivity field was subsequently modified during model calibration. Modifications made to the hydraulic conductivity field during GSA/PORFLOW™ model calibration were different than those made during GSA/FACT model calibration, leading to differences between the two models in terms of the final calibrated hydraulic conductivities. Figure 4-16 illustrates the hydraulic conductivity distribution in feet per day along a vertical slice through the GSA.

The vadose zone is included in the regional GSA/PORFLOW™ model mesh; therefore, soil characteristic curves are required. Pseudo-soil water retention characteristic curves, which exhibit greater linearity than real soil water curves, are used to simulate unsaturated flow from the land surface to the water table under steady-state conditions (Figure 2-7; WSRC-TR-2004-00106, Rev. 0). Horizontal and vertical saturated hydraulic conductivities of vadose zone cells with saturation less than 90 percent are set to 3.5×10^{-5} cm/s (0.1 ft/d) to minimize lateral flow and ensure that modeled water movement in the vadose zone is vertically downward (Section 2.3; WSRC-TR-2004-00106, Rev. 0).

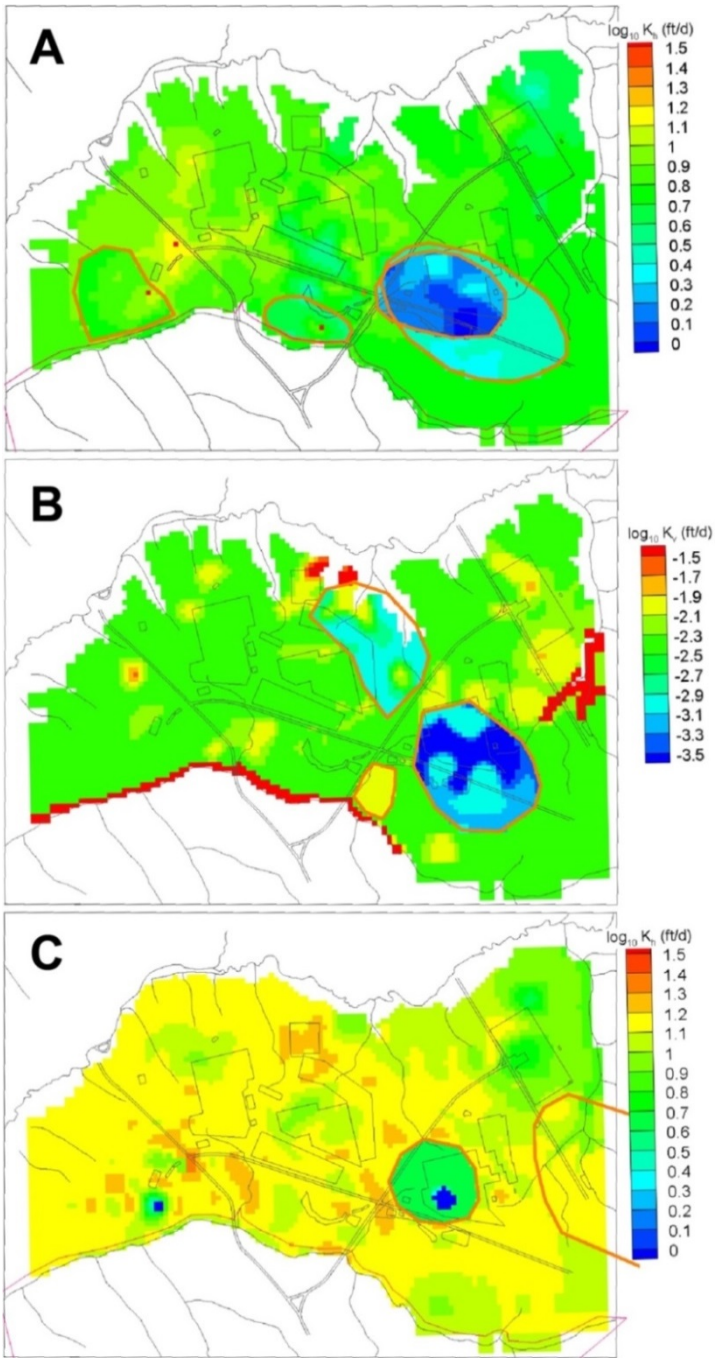
Because DOE thinks that the relatively high water levels observed at H-Area may have resulted from a relatively low-permeability confining zone, the TCCZ is locally assigned relatively low, vertical hydraulic conductivities spanning the range $1.1–2.8\times10^{-7}$ cm/s ($3.1\times10^{-04}–8\times10^{-04}$ ft/d) during calibration (Figure RAI-FF-3.6; SRR-CWDA-2013-00106, Rev. 1). The lower vertical hydraulic conductivity assigned to the TCCZ locally at HTF makes the zone a more competent aquitard in the model compared to other areas of the GSA (see Figure 4-17). The GCU is assigned a uniform calibrated vertical conductivity of 3.5×10^{-8} cm/s (1×10^{-04} ft/d) and represents a significant clayey aquitard across the entire GSA (Page 22; WSRC-TR-96-0399, Vol. 2, Rev. 1). The calibrated Gordon Aquifer unit, for which there is limited field data, is assigned a uniform horizontal hydraulic conductivity of 1.3×0^{-2} cm/s (37 ft/d) because the initial heterogeneous conductivity field was inconsistent with measured hydraulic heads (Page 22; WSRC-TR-96-0399, Vol. 2, Rev. 1). The selected value is said to be consistent with multiple well pumping test data and prior modeling (Page 22; WSRC-TR-96-0399, Vol. 2, Rev. 1). The uniform calibrated hydraulic conductivity values for the GCU and Gordon Aquifer (p. 22; WSRC-TR-96-0399, Vol. 2, Rev. 1) were not modified during the migration from GSA/FACT to GSA/PORFLOW™ (WSRC-TR-2004-00106, Rev. 0).

Figure 4-16 Example Calibrated Horizontal (Top) and Vertical (Bottom) Hydraulic Conductivity Assignments to the Regional GSA/PORFLOW™ Model



Adapted from Figure 4.2-18 in SRR-CWDA-2010-00128, Rev. 1.
Four Mile Branch Is South and the Upper Three Runs Stream Is North of the HTF.

Figure 4-17 GSA/PORFLOW™ Hydraulic Conductivity Adjustment Zones: (A) Vertically Averaged Horizontal Conductivity* in the UTRA-UZ. (B) Vertically Averaged Vertical Conductivity* in the TCCZ. (C) Vertically Averaged Horizontal Conductivity in the UTRA-LZ



Adapted from Figures RAI-FF-3.5—3.7 in SRR-CWDA-2013-00106, Rev. 1.
*To Convert from ft/d to m/d, Multiply by 0.3.

Grid cells in the regional GSA/PORFLOW™ model that have calibrated hydraulic conductivity greater than 1.0×10^{-7} cm/s (3×10^{-04} ft/d) are defined as sandy, while those that have calibrated hydraulic conductivity less than this value are defined as clayey (SRR-CWDA-2010-00128, Rev. 1). K_ds 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 (see Section 4.2.8). The entire UTRA (i.e., the UTRA-UZ, UTRA-LZ, and TCCZ) is assigned sandy K_ds (FTF CC-FF-4 response; SRR-SWDA-2011-00054). K_ds 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_ds are only assigned to the GCU (FTF CC-FF-4 response; SRR-SWDA-2011-00054). K_d values are listed in Table 4.2-25 of the HTF PA (SRR-CWDA-2010-000128, Rev. 1).

With respect to transport properties, effective diffusion coefficients (Table 4-6) 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 (Table 5-14; WSRC-STI-2006-00198, Rev. 0). GSA/PORFLOW™ transport modeling assumes an effective porosity value (Table 4-6) for all aquifers and aquitards to account for dead-end pores that do not participate in radionuclide transport (WSRC-TR-96-0399, Vol. 2, Rev. 1; WSRC-STI-2006-00198, Rev. 0). Because the UTRA and Gordon Aquifer material properties are similar to undisturbed, lower vadose zone material properties measured in E-Area, DOE converts average values for E-Area lower vadose zone bulk density [101 lb/ft^3 (1.62 g/cm^3)] and particle density [166 lb/ft^3 (2.66 g/cm^3)] to effective values for saturated zone bulk density [64.9 lb/ft^3 (1.04 g/cm^3)] and particle density [86.7 lb/ft^3 (1.39 g/cm^3)] using the average value (39 percent) and effective value (25 percent) of E-Area lower vadose zone porosity (Table 5-14; WSRC-STI-2006-00198, Rev. 0). DOE also assumes these same effective material property values for the TCCZ and GCU.

Table 4-6 Saturated Zone Material Properties Used in PORFLOW™ Modeling

Saturated Sediments	Saturated Effective Diffusion Coefficient (cm^2/s)*	Effective Porosity (%)	Effective Dry Bulk Density (g/cm^3)†	Effective Particle Density (g/cm^3)†
Sandy	5.3×10^{-6}	25	1.04	1.39
Clayey	4.0×10^{-6}			

* To convert to in^2/s, multiply by 0.16
† To convert to lb/ft^3, multiply by 62.4

Hydrodynamic dispersion in the local HTF/PORFLOW™ transport model is represented by the following longitudinal and transverse, horizontal and vertical dispersivities:

- α_{LH} = 3.16 m (10.4 ft)
- α_{TH} = 0.316 m (1.04 ft)
- α_{LV} = 0.316 m (1.04 ft)
- α_{TV} = 0.0316 m (0.104 ft),

which are 3.16, 0.316, 0.316, and 0.0316 percent of the nominal 100 m (330 ft) plume travel distance to compliance wells (SRNL-STI-2012-00465). Gelhar et al. (1992) noted that vertical transverse dispersivities are typically two orders of magnitude lower than longitudinal dispersivities, consistent with the approach of SRNL-STI-2012-00465. In comparison, however, dispersivities even lower than those selected by SRNL-STI-2012-00465 [α_L = 1.50 m (5 ft), α_{TH} =

0.10 m (0.33 ft), α_{TV} = 0.01 m (0.03 ft)] were used for a tritium model-optimization study at E-Area Old Burial Ground (WSRC-TR-96-0037) as indicated in the FTF TER (Camper, 2011 [ML112371751]). Nonetheless, DOE significantly lowered dispersivities in the HTF PA relative to those used in the FTF PA (SRS-REG-2007-00002, Rev. 1), which limits the amount of physical dispersion or spreading of the contaminant plume, therefore, the NRC staff finds DOE's dispersivities reasonable for use at HTF.

4.2.10.4 Model Verification, Calibration and Validation

Software Quality Assurance

Acceptance testing of the finite-difference PORFLOW™ version 5.95.0 code, which was used to construct the GSA/PORFLOW™ model, confirmed that the code conserves mass and satisfies Darcy's Law (Section 4.0; WSRC-TR-2004-00106, Rev. 0). Software quality assurance, testing, and verification for PORFLOW™ version 6.30.2 used for local HTF/PORFLOW™ calculations are documented in WSRC-SQP-A-00028, Rev. 0; WSRC-STI-2007-00150, Rev. 0; G-STP-A-00009; G-SQP-A-00012; SRNS-TR-2008-00003, Rev. 0; and SRNL-TR-2010-00213, Rev. 0. In the interest of traceability, DOE's testing and verification of other PORFLOW™ code versions (those used at SRS after GSA/PORFLOW™ model development but before performing HTF/PORFLOW™ calculations) were documented in software testing and verification reports (G-TR-G-00002, Rev. 0; SRNL-TR-2010-00023, Rev. 0; SRNL-TR-2010-00195, Rev. 0.)

Model Calibration

The initial hydraulic conductivity fields of the GSA/PORFLOW™ saturated zone model were similar to those of the GSA/FACT model because they were based upon the same laboratory and field data from the 1995 GSA database. Initial values were subsequently modified to the final calibrated parameter set by locally changing the hydraulic conductivity fields, as necessary, to achieve an acceptable match between model results and measured hydraulic heads and stream baseflow estimates from USGS gauge stations on the Upper Three Runs, Four Mile Branch, McQueen Branch and Crouch Branch streams (Page 8 and Table 1; WSRC-TR-96-0399, Vol. 2, Rev. 1). The average calibrated horizontal conductivities of the GSA/PORFLOW model are given in Table 4-7 (SRR-CWDA-2010-00128, Rev. 1; WSRC-TR-2004-00106, Rev. 0).

The HTF/PORFLOW™ model uses the flow field output from the GSA/PORFLOW™ model to simulate contaminant fate and transport in order to make dose predictions in the HTF PA. During its technical review[15], the NRC staff expressed concern with GSA/PORFLOW™ model calibration. The concern is that if the HTF/PORFLOW™ model is not well-calibrated, the dose predictions may underestimate the dose for some sources and radionuclides. In its HTF RAIs (Mohseni, 2013a [ML13196A135]), the NRC staff documents its concerns with the final calibrated model parameters given information on the sub-optimal calibration of the GSA/PORFLOW™ model, particularly for hydraulic head targets near HTF.

[15] The NRC staff also expressed concern with model calibration at H-Area in the F Tank Farm TER (Camper, 2011 [ML112371751]).

Table 4-7 Calibrated Hydraulic Conductivity Summary

Model Unit	Parameter	GSA/PORFLOW™ Average Calibrated Hydraulic Conductivities (cm/s)*	Reduced H-Area Hydraulic Conductivities (cm/s)*
UTRA-UZ†	K_H	3.5×10^{-03}	$<1.8\times10^{-03}$ or $<7\times10^{-04}$
TCCZ	K_V	2.1×10^{-06}	1.8×10^{-08} – 1.8×10^{-07}
UTRA-LZ	K_H	4.6×10^{-03}	$<2.3\times10^{-03}$
GCU	K_V	3.5×10^{-09}	—
Gordon	K_H	1.3×10^{-02}	—

(SRR-CWDA-2010-00128, Rev. 1; SRR-CWDA-2013-00106, Rev. 1)
* To convert to ft/d, multiply by 2,835.
† Two hydraulic conductivity zones are specified for the UTRA at H-Area. Therefore, two values are provided in column 4 (see SRR-CWDA-2013-00106, Rev. 1, Figure RAI-FF-3.5).

The GSA/FACT and GSA/PORFLOW™ models were calibrated to what was considered long-term average water levels at the time of modeling. However, operational sources and sinks at HTF may have influenced the water level measurements used to develop calibration targets. Calibration targets may also be biased high or low in comparison to long-term values given the relatively short averaging interval of water level measurements. The NRC staff requested that DOE provide additional information regarding calibration statistics local to the area of interest at HTF. Calibration statistics provide an indication of how well the model matches measured field parameters. The NRC staff also requested that DOE re-evaluate hydraulic heads used as calibration targets.

In technical discussions between the NRC and DOE staff, DOE indicated that the GSA/PORLOW™ model was calibrated to hydraulic head targets that were, on average, higher than indicated by more recent well data. For example, during a May 9, 2013, technical exchange DOE indicated that the GSA database is thought to have overestimated hydraulic heads particularly near Tanks 48, 49, 50, and 51 (Shaffner, 2013c [ML13154A327]). Environmental Restoration Data Management System well data suggest that water levels are lower in this part of H-Area than the calibration targets and that the vadose zone is thicker (Shaffner, 2013c [ML13154A327]). In its RAI responses (Response to RAI-FF-1; SRR-CWDA-2013-00106, Rev. 1), DOE continues to evaluate head targets at HTF using more recent 2002 and 2003 data taken from WSRC-TR-2003-00250, Rev. 0. After developing a new set of calibration targets, DOE evaluated the sub-set of (52) wells that had residuals greater than 6 ft (2 m), double the assumed level of natural variability in water level from year to year. Environmental Restoration Data Management System data were also used to evaluate these problematic wells. No effort was made to evaluate wells with relatively low residuals [less than 6 ft (2 m)], although some of the low residual wells might also be suspect, potentially biasing the calibration statistics. Additionally, no wells in the UTRA-LZ at HTF were evaluated. In all, 21 of the 52 problematic wells DOE evaluated were designated as “not credible”, due to issues such as errors in the screened aquifer, suspect or clearly erroneous data based on comparisons to nearby wells, and/or sparse data.

Based on the updated study, DOE identifies two areas that could benefit from additional model calibration: (1) an area north of HTF (near H-Canyon) and (2) an area southwest of HTF (see Figure 4-18). Both areas had modeled heads that were under-predicted by 6 ft (2 m) or more from the calibration targets. Additionally, the GSA/PORFLOW™ model tended to over-predict head by more than 10 ft (3 m) at HTF (see Figure 4-18). In other words, modeled head is generally too high at HTF and too low along groundwater flow pathways away from HTF. Therefore, the horizontal hydraulic gradient in the UTRA-UZ appears to be significantly less than modeled based on more recent data and adjustments made to hydraulic conductivities during model calibration (1) may not have been needed and (2) may not be appropriate.

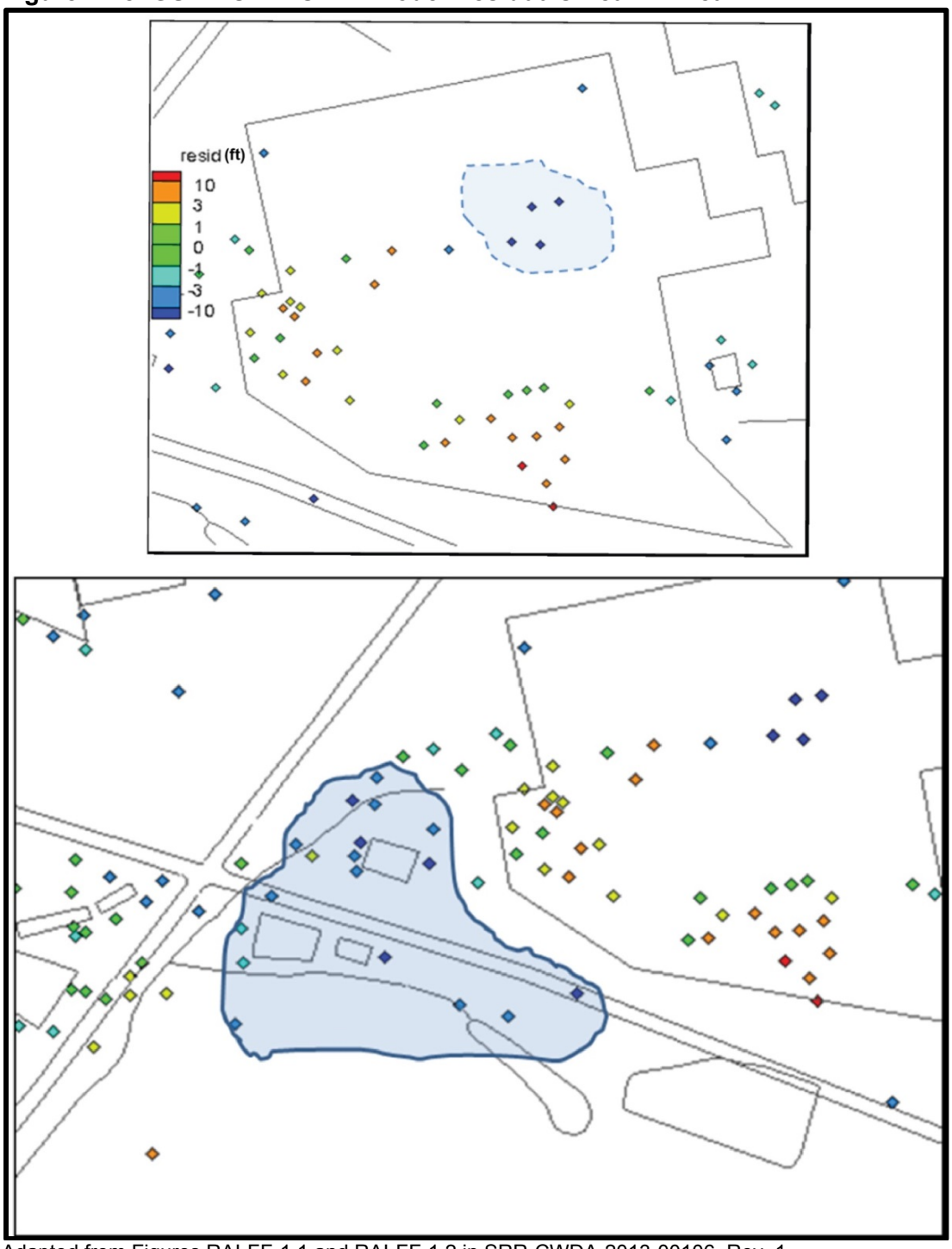
In its HTF RAIs, the NRC staff expressed concern with the lack of corroborating evidence for assignment of lower hydraulic conductivities at HTF during the model calibration process. GSA/PORFLOW™ model documentation (WSRC-TR-2004-00106, Rev. 0) indicates that during model calibration, hydraulic conductivity was lowered in lieu of the use of artificial recharge sources at HTF—a method used in predecessor GSA/FACT model calibration. The GSA/PORFLOW™ model documentation indicates that the elevated hydraulic heads in H-Area suggested by the GSA database may have resulted from a low vertical hydraulic conductivity confining zone (TCCZ), as well as from possible low horizontal hydraulic conductivity aquifer zones in the UTRA. For this reason, during calibration of the GSA/PORFLOW™ model, H-Area hydraulic conductivities were lowered in various elliptical zones consisting of multiple UTRA units (Table 4-7; Figure 4-18) to improve the local match between GSA database information and modeled hydraulic heads (Response to RAI-FF-3; SRR-CWDA-2013-00106, Rev. 1). In technical discussions between the NRC and DOE staff, DOE indicated that there may be evidence of low permeability zones and perched water at HTF but later, and upon further investigation, DOE stated that there appears to be a lack of corroborating evidence for perched zones at HTF (Shaffner, 2013b [ML13126A127]; Shaffner, 2013c [ML13154A327]). No additional information to support the selection of lower hydraulic conductivities at HTF was provided in the response to HTF RAIs.

Model Validation

DOE compared GSA/PORFLOW™-modeled streamlines with E-, F-, and H-Area contaminant plume distribution data to provide confidence that the regional GSA/PORFLOW™ saturated zone model reproduces known plume trajectories in map view (SRR-CWDA-2010-00128, Rev. 1). Other important points of comparison, such as travel time and concentration, were not addressed.

DOE also performed a qualitative comparison of survey data on seep lines to modeled seepage. Field observations and model results are generally consistent (Figure 3-6; WSRC-TR-2004-00106, Rev. 0). DOE also indicates that the GSA/PORFLOW™ model is validated with respect to hydraulic conductivity because 83 percent of the calibrated hydraulic conductivity field agrees or is neutral with respect to measured hydraulic conductivity data (Page 23; WSRC-TR-2004-00106, Rev. 0).

Figure 4-18 GSA/PORFLOW™ Model Residuals Near H-Area



Adapted from Figures RAI-FF-1.1 and RAI-FF-1.2 in SRR-CWDA-2013-00106, Rev. 1.
Encircled Zones Encompass Wells Within and Near H-Area Yielding Model Residuals >1.8 m (>6 ft).
H-Canyon Wells (Top) and Wells Southwest of H-Area (Bottom).
To Convert from ft to m, Multiply by 0.3.

Although no specific calibration criteria related to baseflow are provided, WSRC-TR-96-0399, Vol. 2, Rev. 1, suggests that measured and modeled baseflow are in general agreement (i.e., differences are within the uncertainty range of the data of ±20 to 50 percent). GSA/PORFLOW™ simulated baseflow is in good agreement with Upper Three Runs and Crouch Branch estimates developed from field observations, given the level of uncertainty in the data (Table 4-8). Differences in observed and modeled baseflow for Four Mile Branch and McQueen Branch are a little more significant. However, total observed baseflow is similar to (20 percent higher) the GSA/PORFLOW™ modeled baseflow.

Table 4-8 Comparison of Estimated and GSA/PORFLOW™-Modeled Baseflows

Stream	Estimated Baseflow m^3/s (ft^3/s)	Modeled Baseflow m^3/s (ft^3/s)	Percent Difference (%)
Upper Three Runs & Tributaries	0.52 (18.2)	0.32 (11.4)	−38.5
Four Mile Branch	0.07 (2.6)	0.11 (3.8)	+57.1
McQueen Branch	0.04 (1.5)	0.07 (2.4)	+75.0
Crouch Branch	0.05 (1.8)	0.05 (1.7)	−5.6
Total	0.7 (24.1)	0.5 (19.3)	−0.20
Adapted from Table 3-2 in WSRC-TR-96-0399, Vol. 2, Rev. 1. Uncertainty Range of Data: ± 20 to 50 Percent (WSRC-TR-96-0399, Vol. 2, Rev. 1)			

Particle tracking simulations were performed to compare groundwater travel times for the GSA/FACT and GSA/PORFLOW™ models, including two particle tracks seeded at H-Area (Figure 4-6; WSRC-TR-2004-00106, Rev. 0). Initial simulations showed that GSA/PORFLOW™ modeled travel times were generally longer than GSA/FACT modeled travel times, so the maximum recharge rate was increased from 45.7 cm/yr (18 in/yr) in GSA/FACT to 48.3 cm/yr (19 in/yr) in GSA/PORFLOW™ (Page 7; WSRC-TR-2004-00106, Rev. 0).

GSA/PORFLOW™ tritium transport simulations for E-Area were also benchmarked against the results of GSA/FACT simulations reported in WSRC-TR-2003-00432, Rev 0. Peak concentrations between the simulations were similar but individual runs varied on the order of ±25 percent (p. 25; WSRC-TR-2004-00106, Rev. 0). The report WSRC-TR-2004-00106, Rev. 0, concludes that particle tracking and transport comparisons indicate the velocity fields of the two GSA models are similar.

4.2.10.5 Compliance Point

The NRC's guidance found in NUREG-1854 indicates that after the end of the institutional control period, the dose receptor evaluated to demonstrate compliance with the 10 CFR 61.41 performance objective is assumed to be at the point of maximum exposure located outside of the tank farm area and buffer zone. DOE's selected HTF buffer zone extends approximately 100 m beyond an imaginary boundary drawn around HTF transfer lines and other ancillary equipment (Figure 4-12). DOE indicated that their selection of the compliance boundary for

HTF was predicated on including both waste tanks and ancillary equipment as inventory sources[16] (Shaffner, 2013b [ML13126A127]). In their selection of the compliance boundary, DOE did not preemptively exclude potential inventory sources based on the relatively low contributions of certain sources (Shaffner, 2013b [ML13126A127]). DOE also indicated that compliance points within the 1-m boundary could be used to assess the impact of moving the 1-m compliance boundary closer to more significant HTF sources (Shaffner, 2013c [ML13154A327]).

DOE assumes the groundwater concentrations at the 100-m (330-ft) boundary (Figure 4-12) to be the highest concentrations, and that concentrations decrease beyond the 100-m (330-ft) boundary. DOE indicates that this assumption is supported by Figures 5.2-3 and 5.2-4 in DOE's HTF PA (SRR-CWDA-2010-00128, Rev. 1), which present 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 HTF/PORFLOW™ simulations. Therefore, DOE concludes that hypothetical monitoring wells assumed to be located at the 100-m (330-ft) boundary are adequate to capture the peak concentrations that could occur at or beyond the 100-m (330-ft) boundary (SRR-CWDA-2010-00128, Rev. 1). Actual travel distances from HTF tanks to the 100-m (330-ft) boundary are much longer for some tanks. Table 5.2-2 of DOE's PA shows the approximate travel distances for constituents released from the HTF to DOE's 100-m (330-ft) boundary (SRR-CWDA-2010-00128, Rev. 1); actual travel distances from HTF waste tanks to the 100-m (330-ft) compliance boundary range from 113 to 574 m.

DOE calculates 100-m (330-ft) concentrations for six sectors (Sectors A – F) in the local HTF/PORFLOW™ model, as shown in Figure 5.2-5 of DOE's HTF PA (SRR-CWDA-2010-00128, Rev. 1) reproduced in Figure 4-12. The peak concentration values for the 100-m (330-ft) results are recorded for the three aquifers of concern (i.e., UTRA-UZ, UTRA-LZ, and Gordon Aquifer). The overall peak concentration for each sector independent of the aquifer is also recorded. The vertical thicknesses of the computational mesh are less than 3 m (10 ft) in the UTRA-UZ and less than 4.6 m (15 ft) in the UTRA-LZ. Well screen averaging was not used to determine the concentrations for dose calculations because the typical well screen length of 6.1 m (20 ft) is similar to the thickness of the computational mesh but conservative with respect to being lower in vertical thickness than the average well screen length. 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 HTF. The six sectors are analyzed for each radionuclide and chemical to find the maximum groundwater concentrations 100 m (330 ft) from the HTF. DOE also calculates 1-m (3-ft) concentrations for six sectors (Sectors A – F), as shown in Figure 5.2-5 of DOE's PA (SRR-CWDA-2010-00128, Rev. 1) and Figure 4-12. Using the sectors to determine the highest groundwater concentrations causes the calculated peak

[16] See Section 3.1 for additional information on development of inventories for ancillary equipment.

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 HTF 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 key area of concern for the NRC staff is the coupling and linkages between the various PORFLOW™ models that comprise the waste tank base case (Case A) that DOE heavily relies upon 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 HTF sources that may not mimic the actual behavior of potential releases from the HTF facility.

Another source of uncertainty in DOE's far-field model is the treatment of the calcareous soft zones that are located in the UTRA-LZ. While the geotechnical impact of calcareous zones on the stability of various facilities has been extensively studied at SRS, considerably less attention has been placed on the properties of these zones that might impact contaminant flow and transport. Due to the strong vertical gradient near the groundwater divide, HTF plumes are expected to traverse the lower zone of the UTRA. Discussions with DOE during FTF PA review revealed that mapping of surface water seeps along Upper Three Runs and Four Mile Branch have not focused on surface seeps or other features associated with these zones, although future work in this area could be conducted. DOE has also indicated that cave-like features have been observed in at least one down dip location.

4.2.11 NRC Evaluation of Hydrology and Far-Field Transport

4.2.11.1 NRC Evaluation of Model Construction Including Boundary Conditions

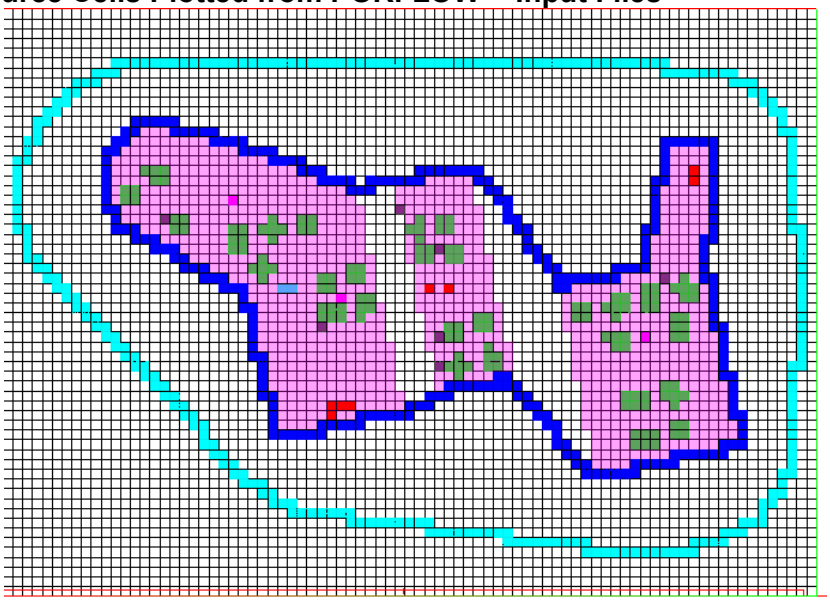
DOE expended a great deal of time and effort to develop a GSA hydraulic conductivity parameter set from field and laboratory characterization data (WSRC-TR-96-0399, Vol. 2, Rev. 1), rather than using a simpler approach of using a uniform parameter set and adding variation as needed to match measured hydraulic heads and estimated stream baseflows. However, as discussed in Section 4.2.11.3, calibration of DOE's model could be improved. While data from pumping tests in the area of interest may be lacking, other data not available in the GSA database are now available through other sources. For example, the GSA database does not incorporate relatively dense H-Area site data acquired since 1995 (e.g., cone penetrometer testing picks for the top and thickness of the TCCZ; SRNL-STI-2010-00148, Rev. 0) that could be used to improve the model. Furthermore, DOE's review of more recent water level data to support the HTF PA indicates significant variability in water levels over time and potential issues with targets used in the GSA/PORFLOW™ model calibration, as discussed in technical discussions with DOE staff (Shaffner, 2013b [ML13126A127]) and the HTF RAI responses (SRR-CWDA-2013-00106, Rev. 1). Inclusion of a longer period of record of water level data for HTF wells could improve the development of steady-state water levels and provide insight regarding the impact of natural variability on the flow field. More recent data, if

incorporated, could improve model calibration (Section 4.2.11.3) and resulting matches between modeled and measured hydraulic heads.

With regard to model integration, DOE uses two separate models to calculate radionuclide concentrations at various points of compliance where the member of the public and inadvertent intruder doses are calculated—a near-field model near HTF sources in the unsaturated zone and a far-field model that encompasses the saturated zone and discharge locations away from HTF sources. DOE applies the mass flux of each radionuclide exiting the near-field model (or exiting submerged tank basemats) to far-field model source elements at or below the water table. DOE distributes the contaminant fluxes to water table grid cells with centroids lying within the footprint of the waste source (orange squares in Figure 4-12 or green squares in Figure 4-19) or to cells at the depth of submerged Type I and II tank basemats (Shaffner, 2013c [ML13154A327]).

Upon comparing source cells that are illustrated in Figure 4-12, which is reproduced from the HTF PA, to source cells that are listed in HTF/PORFLOW™ input files and plotted by the NRC staff in Figure 4-19, there are significant differences in the number of source cells to which the flux is applied. The NRC staff's review of DOE's RAI-NF-14 response (SRR-CWDA-2013-00106, Rev. 1), confirms that HTF PA Figure 5.2-5 from which Figure 4-12 is adapted is in error and the HTF/PORFLOW™ input files plotted by the NRC staff reflect the actual modeled source cells. In any event, the number of source cells used in the modeling influences the starting concentration of key radionuclides in the aquifer. Additionally, given the strong vertical gradient at HTF and the apparent differences in flow direction in the upper and lower zones of the UTRA, the elevation of source loading may also significantly influence the degree of vertical spreading and the modeled level of dispersion and dilution in the aquifer.

Figure 4-19 Source Cells Plotted from PORFLOW™ Input Files



Tank sources depicted with green squares.
Includes ancillary equipment (e.g., pump pits are depicted with red squares).

While DOE indicates that results are not sensitive to source loading assumptions (SRR-CWDA-2010-00128, Rev. 1), DOE appears to have evaluated the sensitivity of the results to a range of source cells different from what was actually used in the PORFLOW™ modeling. Additionally, DOE attributes differences in path lines presented in Figures 5.2-6 (correct) and 4.4-13 (incorrect) in the HTF PA (SRR-CWDA-2010-00128, Rev. 1) to slight differences in the assignment of velocity vectors in plotting software, further suggesting the potential sensitivity of the results to slight changes in the flow field or source loading locations. Finally, the grid cells in the upper zone of the UTRA have a coarser vertical discretization compared to the lower zone of the UTRA and it is not clear whether eastern source cells are loaded above the water table, based on Figure RAI-NF-14.2 in SRR-CWDA-2013-00106, Rev. 1.

Sensitivity of the results to the elevation or location of source loading, as well as the number of source cells should be further evaluated. In future PA analyses, the manner in which contaminant fluxes are loaded in the far-field model should be evaluated by DOE to ensure that the dose estimates are not significantly underestimated.

During the FTF review and in response to FTF RAI-FF-3 (SRR-SWDA-2011-00054), DOE presented information that indicated additional mesh refinement may be necessary to reduce numerical dispersion in cases of very low to no 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 three to four times higher with a mesh refined by a factor of two in each dimension (or a factor of 8 times more cells). Thus, numerical dispersion is sometimes used to represent physical dispersion if little to no physical dispersion is observed. Mesh discretization of the local HTF/PORFLOW™ model was improved considerably over that of the local model used in the FTF PA, with grid refinement 6 x 6 in the x- and y-directions in the local HTF/PORFLOW™ model as described above. The NRC staff's review of DOE's representation of physical dispersion in DOE's HTF/PORFLOW™ model is discussed in the next section.

4.2.11.2 NRC Evaluation of Material Properties and Parameters

In HTF aquifer transport modeling, hydrodynamic dispersion is represented by a stratified dispersion model (SRNL-STI-2012-00465) defined by longitudinal horizontal, longitudinal vertical, transverse horizontal, and transverse vertical dispersivities of 3.16 m (0.104 ft), 0.316 m (0.104 ft), 0.316 m (0.104 ft), and 0.0316 m (0.104 ft), respectively, which are 3.16, 0.316, 0.316, and 0.0316 percent of a nominal 100-m (330-ft) plume travel distance. The NRC staff thinks that (1) the improved capability to simulate dispersion in the updated version of PORFLOW™ (version 6.30.2) used to develop the HTF/PORFLOW™ model (compared to a previous version of PORFLOW™ that used to develop the FTF/PORFLOW™ model), and (2) reductions in the assumed level of dispersion in the HTF/PORFLOW™ model (compared to the dispersivities used in the FTF/PORFLOW™ model) have increased the technical defensibility of the HTF/PORFLOW™ model with respect to the modeled level of dispersion.

Hydraulic conductivity values at H-Area were arbitrarily lowered over large polygonal areas in the UTRA-UZ, UTRA-LZ, and TCCZ to better match calibration targets (SRR-CWDA-2013-00106, Rev. 1). There is no known geological basis for the locally low hydraulic conductivities assigned to the model layers representing H-Area and its vicinity. A potential consequence of the arbitrarily assigned low hydraulic conductivities is that modeled hydraulic heads beneath the

HTF are artificially high, whereas modeled hydraulic heads in outlying portions of H-Area are artificially low (Figure 4-18). The overall consequences to dose estimates due to inadequate flow and transport representation in the vicinity of H-Area are difficult to assess.

With regard to K_ds, DOE's response to FTF RAI-FF-4 (SRR-SWDA-2011-00054) indicates that it will only pursue follow-up work related to soft zone sorption (e.g., plutonium sorption to soft zone material in the presence of potentially high carbonate concentrations) if soft zones are determined to be a risk-driver. Among the NRC staff's concerns are the potential effects of pH and aqueous carbonate concentrations on sorption properties, particularly for actinides. A key point in DOE's response is 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" SRR-SWDA-2011-00054. The 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 the potential impacts of these zones on contaminant transport. DOE progress in this area will be evaluated during the monitoring phase. DOE performed deterministic sensitivity analysis in HTF/PORFLOW™ on porosity and bulk density to study the potential impact of calcareous zones on contaminant flow and transport at HTF (Page 709; SRR-CWDA-2010-00128, Rev. 1). The results of the sensitivity analysis indicate the potential risk significance of calcareous zone impacts on HTF dose projections at the 100-m (330-ft) boundary (see Sections 4.2.18 and 4.2.19).

DOE separately evaluates the impact of sorption on dose projections (Page 710; SRR-CWDA-2010-00128, Rev. 1) considering a soil K_d that is ½ and ¼ the nominal values. The sensitivity analysis on the soil $K_d$sshows a significant impact on peak dose over longer simulation periods (20,000 year) due to the impact of sorption on Ra-226 peak dose. The NRC staff more fully evaluates DOE's selection of base case K_ds in Section 4.2.9.4.

4.2.11.3 NRC Evaluation of Model Calibration and Validation

The HTF/PORFLOW™ model uses the flow field output from the GSA/PORFLOW™ model to simulate contaminant fate and transport for the purpose of making dose predictions in the HTF PA. During its technical review, the NRC staff expressed concern with the GSA/PORFLOW™ model calibration. The concern is that if the HTF/PORFLOW™ model is not well-calibrated, the dose predictions may underestimate dose for some sources and radionuclides. In its HTF RAIs (Mohseni, 2013a [ML13196A135]), the NRC staff documents its concerns with the final calibrated model parameters given information on the sub-optimal calibration of the GSA/PORFLOW model against hydraulic head targets near HTF. For example, GSA/PORFLOW™ and GSA/FACT[17] documentation suggest that the GSA model is not well calibrated in the vicinity of HTF.

[17]The GSA/FACT model is the predecessor to the GSA/PORFLOW™ model. Similar data sets were used to construct both models. A similar conceptual model of the GSA is implemented in the models.

Specifically, WSRC-TR-96-00399, Rev. 1, Vol. 2 indicates that (1) there are unexpected high residuals east of HTF (page 23); (2) relatively larger residuals are found in and east of HTF (page 24); (3) additional work is needed to better define the artificial recharge and hydraulic conductivity field at HTF, and that artificial recharge may be excessive suggesting the hydraulic conductivity field may require additional adjustment (page 25); and (4) additional work is needed to better define uncertainty in model predictions (page 25). Further, WSRC-TR-2004-00106, Rev. 0 indicates (page 23) that GSA/PORFLOW™ head residuals are generally relatively large compared to GSA/FACT. The report also states that the artificial recharge zone in the GSA/FACT model was more effective at reducing head residuals at HTF but was considered less realistic and that more extensive model calibration would improve the GSA/PORFLOW™ model (page 24).

The NRC staff agrees with WSRC-TR-2004-00106, Rev. 0, that more extensive model calibration could significantly improve the GSA/PORFLOW™ model (page 24), particularly in the H-Area. In its RAI responses (RAI-FF-2; SRR-CWDA-2013-00106, Rev. 1), DOE provides revised calibration statistics for the entire GSA as well as for the local area near HTF (see Table 4-9.) It is important to note that the updated study is focused on the water table aquifer or the UTRA-UZ at HTF. Therefore, no UTRA-LZ well data are evaluated in the updated study. However, the vertical gradient at HTF is rather strong and high UTRA-LZ residuals are observed in the GSA/PORFLOW™ model.

Table 4-9 Summary Calibration Statistics for the GSA/PORFLOW™ Model Using 2003 Well Targets Across (1) the Entire General Separations Area, and (2) Local to H-Area

Aquifer Zone	Number of Wells	Residual Median	Residual Average	Residual RMS	Residual Minimum	Residual Maximum
GSA						
Gordon	—	—	—	—	—	—
UTRA-LZ	52	+2.7	+1.6	4.3	−9.8	+9.0
UTR-UZ	406	+0.4	−0.0	3.4	−15.2	+7.5
H-Area						
Gordon	—	—	—	—	—	—
UTRA-LZ	—	—	—	—	—	—
UTR-UZ	85	0.8	−2.0	5.4	−15.2	+5.9

Although no calibration metrics were provided by DOE, the NRC staff performed a qualitative evaluation against calibration goals cited in technical literature. In general, the magnitude of the error of the total head loss in the area of interest should be small (Anderson and Woessner, 1992; ASTM, 2002). Rumbaugh (1998) has suggested that the minimum acceptable criteria is 5 percent of the mean error (ideally 2 percent), and 10 percent for the root mean squared (RMS) error (and ideally 5 percent). Review of the 2003 water table surface in SRR-CWDA-2010-00128, Rev. 1 (Page 286) reveals a slight hydraulic gradient at HTF [see Figure 4-20(a)]. Along the margins of H-Area, the gradient increases. A head loss of 45 ft (14 m) is assumed for comparison against DOE's revised RMS error for HTF. An ideal RMS error would be 2.3 ft (0.68 m), while an acceptable RMS error would be 4.5 ft (1.4 m). However, the DOE calculated RMS error reported in Table 4-9 above is 5.4 ft (1.6 m) or 12 percent of the head loss across the HTF. The RMS error may be biased low because only high residuals are re-evaluated and

removed in the revised study, and no UTRA-LZ data are provided, although high residuals in the UTRA-LZ are reported in the GSA/PORFLOW model documentation (WSRC-TR-2004-00106, Rev. 0) [see Figure 4-21(a)]. Finally, data are clustered [see Figure 4-20(b)] leading to inequitable spatial weighting of calibration targets with some locations dominating the results. If observed versus modeled water levels are plotted, the plot would show that the errors are not random but are spatially biased with high calibration target areas too high and low calibration targets areas too low in the UTRA-UZ, leading to significant error in the hydraulic gradient at HTF.

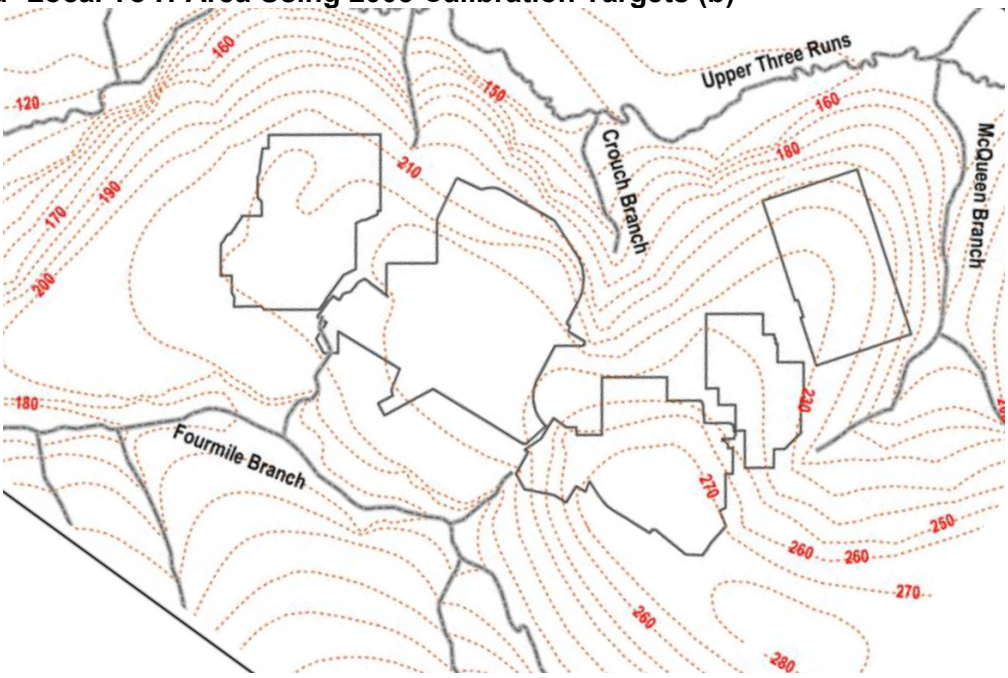
Given (1) the significant errors in hydraulic head residuals in H-Area and (2) the unsupported, low hydraulic conductivities assigned to elliptical regions in the UTRA in order to achieve the existing calibration, more extensive calibration is desirable to assure realistic flow and transport model performance. DOE should consider using parameter estimation and inverse modeling techniques to improve the existing model calibration. Model adjustments should be supported by data, and therefore, if pumping tests in the vicinity of H-Area are needed to better define hydraulic conductivity fields at H-Area, then these studies should be conducted.

The GSA/PORFLOW™ model applies a uniform 48 cm/yr (19 in/yr) recharge rate over much of the model domain (WSRC-TR-2004-00106, Rev. 0), although several water balance studies conducted at SRS support a lower recharge rate of approximately 38 cm/yr (15 in/yr) (Table 9; WSRC-STI-2007-00184, Rev. 2). Additionally, the long-term recharge rate does not consider the effects of an infiltration-reducing closure cap that is assumed to be effective in the first 2,000 years following facility closure.

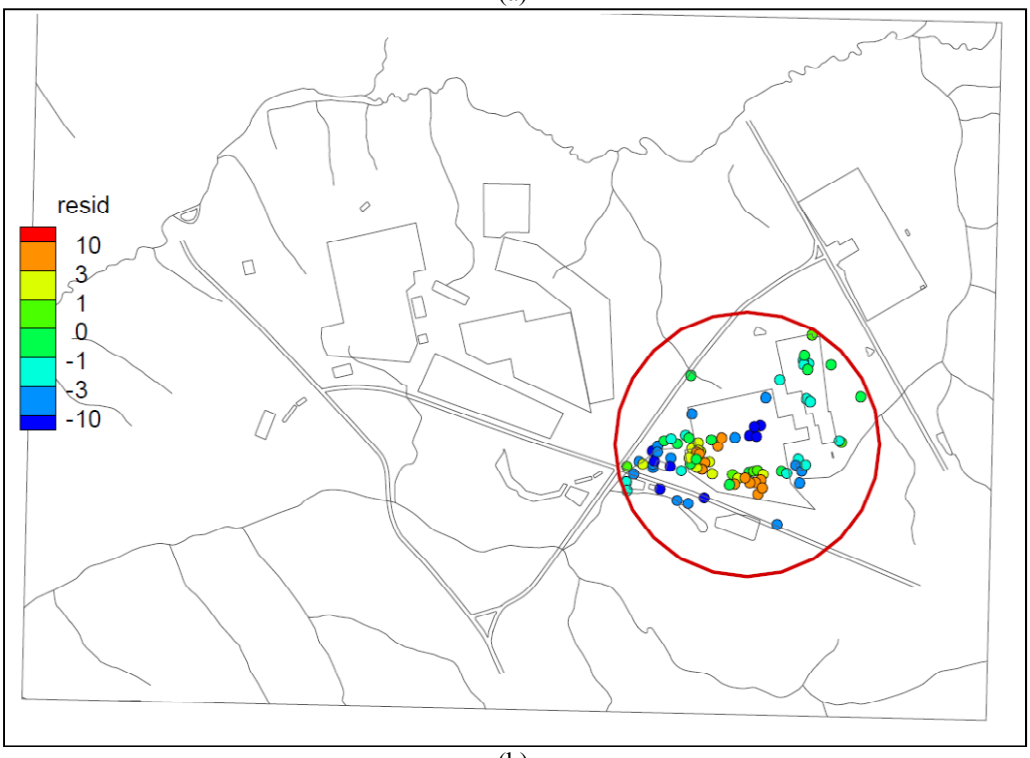
To address the risk significance of a sub-optimal calibration of the GSA/PORFLOW™ model in the area of HTF on the HTF flow field, DOE performed a sensitivity study presenting a “fast” and “slow” flow case, in addition to the nominal case. In this study DOE varies (1) recharge, (2) leakage through the GCU, and (3) effective porosity. These parameters are adjusted to their +/- 95 percent confidence level to produce bounding “fast” and “slow” travel times. The UTRA hydraulic conductivity field is also adjusted to maintain an adequate calibration to hydraulic head targets. DOE provides particle track results for HTF tank sources for the nominal, fast, and slow flow cases in SRR-CWDA-2013-00106, Rev. 1 (pages 148-150). Significant differences in flow directions and rates are observed in these simulations. Plumes tend to converge in the “slow” case, while significantly faster travel times to the compliance boundary are realized in the “fast” case (couple to a few tens of years in the fast case compared to few tens to more than 100 years in the slow case).

DOE also performed additional deterministic sensitivity analysis on Darcy velocity using the HTF GoldSim™ model. A “fast” and “slow” Darcy velocity case were run using the same modeled path lines from the base case (change in flow direction were not studied). The “fast” Darcy velocity case is not significantly different than the nominal, or base case, Darcy velocity case. The results show very modest impacts on the peak dose. For example, the doses at earlier times are slightly higher for the slow velocity case due to less dilution of long-lived and mobile Tc-99. Later doses are slightly higher for the fast velocity case due to faster travel times for moderately long-lived but relatively less mobile Ra-226, because of less decay of Ra-226 prior to transport to the 100-m (330-ft) boundary. Ra-226 peak dose is higher in the fast velocity case, despite the fact that higher velocity also leads to additional dilution and mixing. This sensitivity analysis differs from the sensitivity analysis presented in the HTF PA in that (1) the

Figure 4-20 2003 Water Table Surface* (a) and GSA/PORFLOW Model Residuals Reported† Local To H-Area Using 2003 Calibration Targets (b)



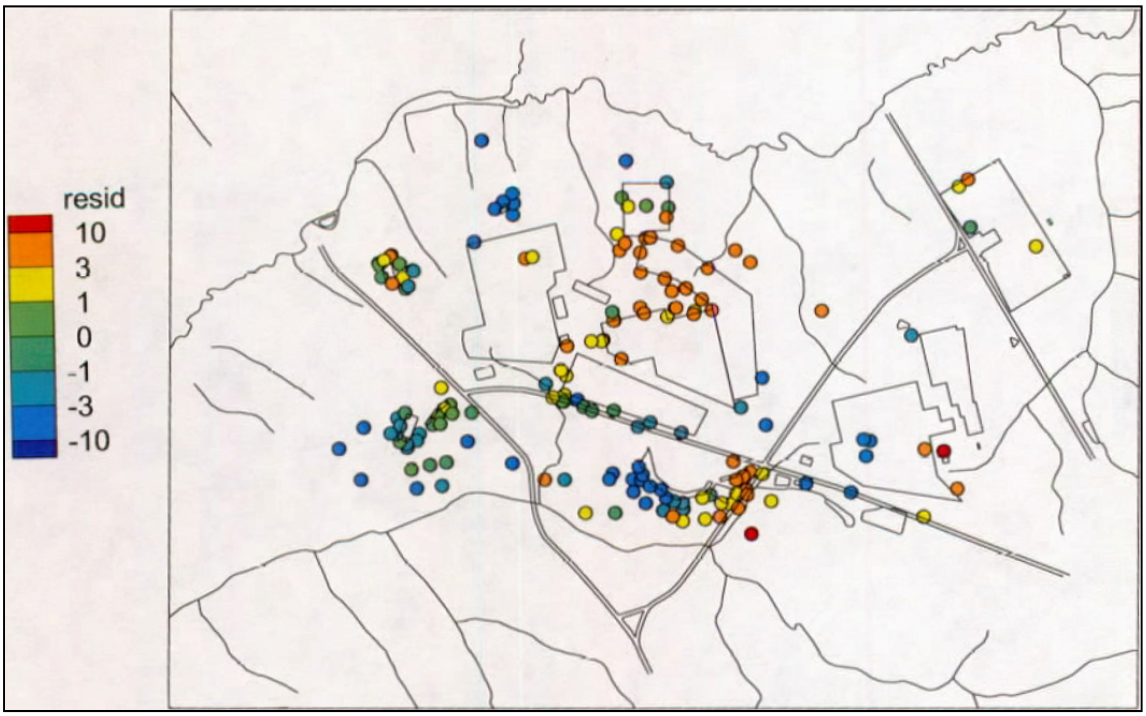
(a)



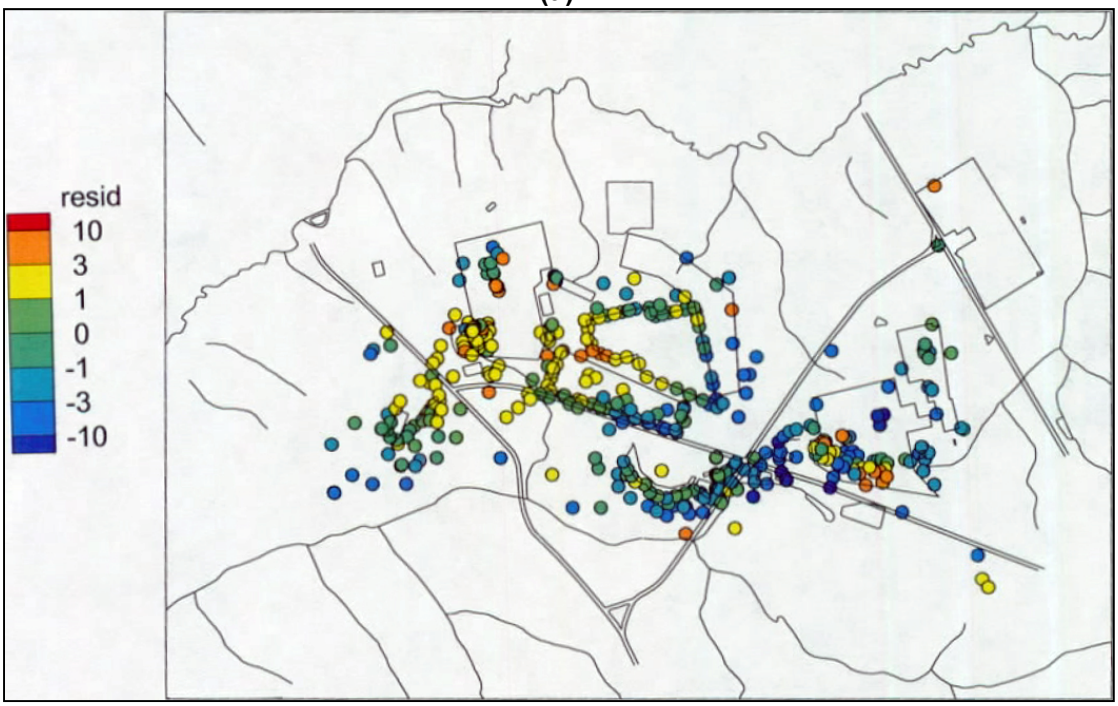
(b)

*SRNL-STI-2010-00148, Rev. 0
†SRR-CWDA-2013-00106, Rev. 1

Figure 4-21 GSA/PORFLOW Residuals in Feet for the (a) UTRA-LZ and (b) UTRA-UZ



(a)



(b)

WSRC-TR-2004-00106, Rev. 0
* To convert to m, multiply by 0.3.

impact of flow direction is not evaluated (contributions from multiple tanks is not considered), and (2) all of the Darcy velocities are changed in the "slow" and "fast" case, rather than assuming the same Darcy velocity for all sources. The more bounding sensitivity cases presented in the HTF PA (page 726; SRR-CWDA-2010-00128, Rev. 1) show differences in base case peak doses of up to a factor of four higher over a 20,000-year compliance period. Additional discussion regarding the HTF PA sensitivity case is found in Section 4.2.18.

Finally, DOE presents arguments in its RAI-FF-4 response (SRR-CWDA-2013-00106, Rev. 1) regarding the lack of sensitivity of potentiometric surfaces to natural variability, including climatic variations. DOE presents water levels from the HTF PA for the entire GSA for three time periods (1995, 1998, and 2003), as evidence of the lack of change in the water table surface. DOE's arguments focus on the larger GSA rather than local impacts at the engineered HTF facility that are important to the NRC staff's review (e.g., local 1-m [3-ft] and 100-m [330-ft] boundary impacts from the engineered HTF system). DOE indicates that the UTRA-LZ potentiometric surface at HTF is similar to the water table because the TCCZ separating the UTRA-UZ from the UTRA-LZ is generally not considered to be a major impedance to the vertical movement of groundwater. DOE states that the HTF/PORFLOW™ modeling and associated flow field reflect a strong correlation between the two aquifer zones. This is despite the fact that the TCCZ at HTF supports a large gradient between the UTRA-UZ and UTRA-LZ, and that DOE lowered the K_v of the TCCZ at HTF during the calibration process, thereby, making it a more effective barrier. In HTF technical discussions, DOE also implicated vertical transport through the TCCZ with a change in flow direction in the UTRA-LZ (e.g., see kinks in path lines in Figures 4-12 and 4-14 that reflect a change in flow direction after particles travel through the TCCZ into the UTRA-LZ). The hydrogeological conceptual model at HTF remains ambiguous and it is clear that additional data collection is needed to better understand natural system performance.

To study the sensitivity of the results to recharge, DOE presents sensitivity analysis results for a low-recharge case (100 times lower) over the HTF that shows significant impacts on the HTF flow field (SRR-CWDA-2013-00106, Rev. 1). The low-recharge case leads to a more prominent vertical trajectory and converging flow paths to the north of HTF. Portage simulations of closure cap impacts also show the sensitivity of the HTF flow field on reduced recharge over the footprint of the HTF and increased infiltration along the perimeter (PORTAGE-08-022, Rev. 0). The Portage simulations do not consider focused flow between the West and East Hills and only allowed one third of the diverted infiltration to drain along the outer perimeter. Nonetheless, important insights are gained from review of the simulations results. In all cases, a strong vertical gradient is observed through the Type I and II tanks, which were the only HTF tanks simulated. The intact closure cap case results in reduced velocity vectors and changes to flow trajectories (some stream traces went to Fourmile Creek versus Upper Three Runs Creek). Finally, the intact closure cap case shows Type II tank bottoms above the water table, while the degraded closure cap case shows Type II tank bottoms in slight contact with the water table. Additional information regarding the effects of the infiltration-reducing cap is found in Sections 4.2.4 and 4.2.5.

Given the limited nature of DOE sensitivity analysis to study far-field model uncertainty (i.e., only a small suite of radionuclides are considered in Case A simulations and generally only one or two parameters are varied in sensitivity analyses), the impact of improved model calibration in the area of interest on HTF PA dose predictions is still not clear. Plausible ranges of recharge and hydraulic conductivity, and the resulting hydraulic gradient, appear to significantly affect

modeled plume overlap and flow rates based on DOE sensitivity analyses. In conclusion, changes to recharge, hydraulic conductivities, and hydraulic gradients could lead to greater cumulative impacts and/or less aquifer dilution than assumed in the HTF PA. The level of uncertainty in the dose projections based on the HTF hydrogeological model uncertainty alone is expected to be less than an order of magnitude for relatively mobile, long-lived radionuclides. However, the impact of uncertainty could be higher for other constituents (e.g., orders of magnitude for relatively short-lived and moderately to strongly sorbing constituents). Improved calibration in the area of interest would serve to provide additional support for the assumed level of dilution/dispersion of HTF releases.

4.2.11.4 NRC Evaluation of Compliance Boundaries

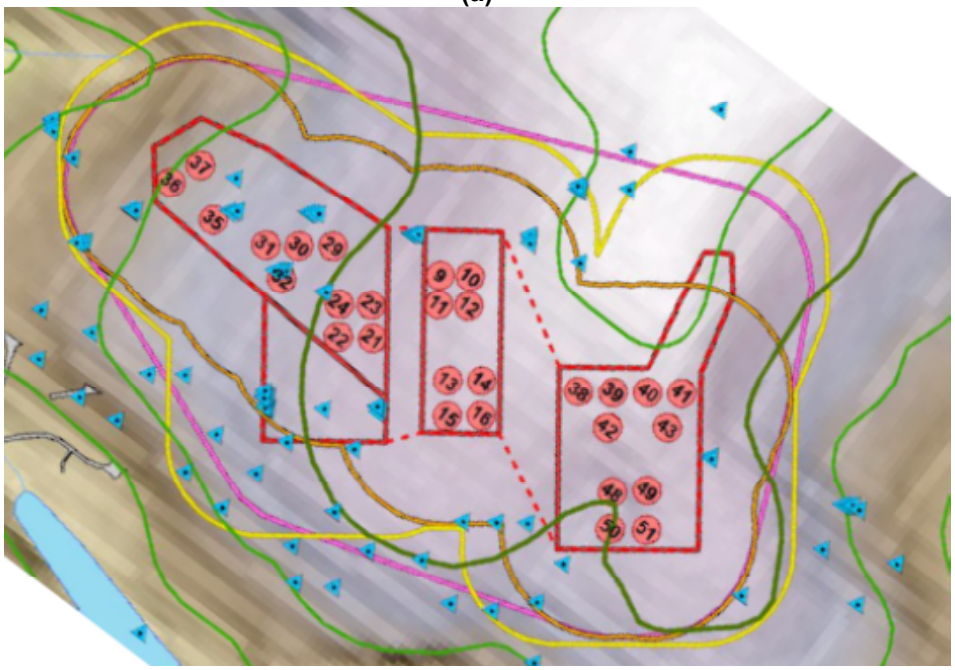
In determining the facility boundary from which the 100-m (330-ft) buffer zone is calculated, DOE includes several HTF sources that are considered much less risk significant than other HTF tank sources, based on HTF PA modeling results. Most notably, relatively insignificant HTF sources such as the pump pits to the north, east, and south central portions of the HTF are included in the facility boundary from which the 100-m (330-ft) boundary was calculated (see Figure 4-22[a]). The most significant impact of inclusion of these less risk-significant sources is an extension of the distance that source plumes must travel to reach the 1-m (3-ft) boundary (e.g., note the distances Type II tanks 13-16, Type IV tanks 21-24 located in the central portion of the HTF, and several Type IIIA tanks located on the East Hill must travel to the 1-m (3-ft) boundary in Figure 4-22[a]). With respect to the 100-m (330-ft) boundary, Type IIIA tanks on the East Hill and some Type II tanks on the West Hill also experience extended travel distances to the 100-m (330-ft) boundary based on DOE's selection of the facility boundary. The orange boundary displayed on Figure 4-22(b) shows an alternative, more conservatively drawn boundary representing a 100-m (330-ft) buffer zone around just the HTF tanks, as suggested in NUREG-1854. Source pathlines (illustrated in Figure 4-22[a]) would intersect the orange boundary in Figure 4-22(b) much closer to the source compared to DOE's pink, 100-m (330-ft) boundary (also illustrated on Figure 4-22[b]). The risk significance of the extended boundary is expected to be greater for relatively short-lived radionuclides, as well as moderately to strongly sorbing constituents[18], while the impact of the extended boundary is expected to be low to moderate for long-lived, relatively mobile radionuclides such as Tc-99.

DOE indicates that upgradient sources located far from the 100-m (330-ft) boundary (such as Tanks 9–14) will cross the TCCZ inside the boundary, especially for eastern sources (SRR-CWDA-2013-00106, Rev. 1). However, sources located near the 100-m (330-ft) boundary (such as Tanks 36 and 37), may or may not cross the TCCZ inside the boundary. This suggests that maximum concentrations for sources closer to the 100-m (330-ft) boundary may

[18] Increased sorption primarily affects the timing of peak dose but can also affect the magnitude of peak dose. For example, the magnitude of peak dose can be significantly lower for relatively short-lived radionuclides due to decay during extended transport to a compliance well (if the half-life of the radionuclide is comparable to or less than the travel time to the well). The magnitude of peak dose can also be significantly lower if the travel time to the well is significantly longer than the time period over which significant source release occurs (i.e., peak dose is a function of the K_d and source release profile).

Figure 4-22 Pathlines from HTF Sources and 1-m and 100-m Boundaries (a) and Alternatives for the 100-m Compliance Boundary: (1) DOE Selected Boundary (Pink), (2) 100-m Buffer Around HTF Tanks (Orange), and (3) Buffer Around Inner Red 1-m Boundary (Yellow) (b)

(a)



(b)

occur in the UTRA-UZ and that maximum concentrations in the UTRA-LZ and in the Gordon Aquifer may not be fully realized at the 100-m (330-ft) well location extracted from the HTF/PORFLOW™ model. The location of the maximum concentration in each aquifer at or beyond the 100-m (330-ft) boundary is needed to estimate the Gordon Aquifer concentration in the probabilistic model because transport in the Gordon Aquifer is not simulated but rather estimated from UTRA transport simulations.

DOE assigns a relatively high probability that a well will be drilled into the Gordon Aquifer in the probabilistic analysis (see Table 4-10). DOE assumes the Gordon Aquifer concentration is a factor of 100 lower than the UTRA concentration in the GoldSim™ model based on observations from the PORFLOW™ model (page 635; SRR-CWDA-2010-00128, Rev. 1). Sensitivity analysis indicates well completion (aquifer) is one of the single-most important parameters 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 UTRA-UZ Concentration	Weighted Concentration
UTRA-UZ	0.04	1	0.04
UTRA-LZ	0.52	1	0.52
Gordon	0.44	0.01	0.0044
Total	1.0	N/A	0.56

As indicated in the FTF TER (Camper, 2011 [ML112371751]), Gordon Aquifer concentrations should not be used to demonstrate compliance with the performance objectives if higher concentrations are observed in another aquifer that can support groundwater dependent pathways. While information on the attenuation of HTF releases due to transport through the GCU into the Gordon Aquifer is informative and provides additional risk information, dose at the point of maximum exposure in the UTRA is more appropriate for comparison against the performance objectives. The NRC staff presents dose statistics for just those realizations with aquifer completion in the UTRA in Section 4.2.19.

4.2.11.5 NRC Evaluation of Data and Model Uncertainty

Finally, the NRC staff continues to be concerned with the potential impact of calcareous zones on contaminant flow and transport at the tank farm facilities. In the FTF TER (Camper, 2011 [ML112371751]), the NRC staff expresses concern with the potential for preferential flow and reduced attenuation of key radionuclides due to calcareous zone dissolution. The report SRNL-TR-2012-00160, Rev. 0 reviews calcareous zone investigations at the SRS and suggests that calcareous zones are less of a factor in H-Area than in areas to the southeast and that the hydraulic properties for any zones within HTF are encompassed by the range of physical properties included in the GSA hydrogeological model and other dependent models (page 7). Because HTF/PORFLOW™ model results suggest that many HTF plumes are likely to be transported in the lower Upper Three Runs Aquifer prior to reaching the 100-m (330-ft) compliance boundary (e.g., CC-FF-1 response; SRR-CWDA-2013-00106, Rev. 1), this issue is of relevance to HTF.

No formal mapping to identify calcareous zone seeps along stream valleys has been conducted. Nonetheless, DOE indicated in an FTF RAI resolution meeting (Shaffner, 2011b [ML112580225]) that a field mapping activity such as this could be incorporated in the future. The NRC staff continues to support such an activity to evaluate the impact of potentially highly porous and conductive soft zones in the UTRA-LZ on the HTF hydrogeologic system. The NRC staff will evaluate DOE's progress in this area during monitoring at HTF. Tracer studies in the HTF UTRA-LZ with innocuous tracers that are commonly used to understand preferential flow and transport also 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 the NRC staff during the monitoring period.

As discussed in the FTF TER and FTF Monitoring Plan (Camper, 2011 [ML112371751]; Camper, 2013a [ML12345A322]), the NRC staff has technical concerns with the K_d averaging approach employed in the FTF PA (SRS-REG-2007-00002, Rev. 1) for plutonium that tends to delay the timing of peak dose for more mobile forms of plutonium that may exist in SRS environments based on site-specific data and modeling. In the HTF PA, DOE revises plutonium K_ds upwards using information from a modeling analysis of long-term lysimeter studies and the results of a statistical analysis of 65 K_d values taken from many areas and materials around the SRS (SRNL-STI-2011-00672). The analysis documented in SRNL-STI-2011-00672 considers only a limited set of chemical conditions (i.e., pH) and does not consider redox states for plutonium (see discussion in Section 4.2.9).

Long-term lysimeter experiments (as referenced in Kaplan et al., 2006) indicate that although most plutonium is in the (IV) oxidation state, there is a small component that at times is much more mobile. Additional detail is provided in Appendix E of the FTF Monitoring Plan (Camper, 2013a [ML12345A322]). Recognizing that plutonium chemistry is especially complex and disproportionation presents a difficult problem, the NRC staff continues to evaluate the appropriateness of plutonium K_d values used in the HTF PA, as well as the K_d values for other key radionuclides.

4.2.11.6 Far-field Review Results and Recommendations

The NRC staff concludes that DOE's far-field model presents an acceptable framework to facilitate decision-making regarding HTF closure. However, the NRC staff notes a number of areas where DOE could make improvements to its far-field model to reduce uncertainty in dose modeling predictions. Changes in key parameters (e.g., hydraulic conductivities and time-variant recharge rates) and improved model calibration in the area of interest could lead to (1) changes in flow direction and (2) changes in Darcy velocities that could result in significantly different levels of modeled dilution and dispersion, depending on source location and radionuclide. The overall impact of a more accurate HTF flow model on the projected dose is unclear. DOE sensitivity analyses (SRR-CWDA-2010-000128, Rev. 1 and SRR-CWDA-2013-00106, Rev. 1) show modest impacts from changes to the Darcy velocity and groundwater flow directions on the results. However, the impacts are not insignificant and the cumulative impact from changes to multiple parameters has only been evaluated in a limited sense. The NRC staff concludes that far-field model uncertainty can lead to less than an order of magnitude changes on peak dose for relatively long-lived, mobile constituents, while far-field model uncertainty can lead to an order of magnitude or more changes in the peak dose for relatively short-lived, more sorbing constituents. Other areas of uncertainty in the far-field model include (1) the impact of

selection of the compliance boundary and source loading assumptions on projected doses, (2) the extent to which calcareous zones will affect contaminant flow and transport at HTF, and (3) the appropriateness of DOE's selection of K_ds for key radionuclides that are expected to have a significant impact on the projected doses.

The NRC staff concludes that in future PA updates, DOE could improve the current far-field model to reduce uncertainty in dose modeling predictions. For example:

1. Model calibration could be improved particularly local to H-Area to provide confidence that the modeled level of dilution and dispersion in the HTF PA is not overstated. The success of this activity may be dependent on the collection of additional data (e.g., pumping test data) in the area of interest to constrain parameter values; DOE should also continue to study uncertainty in calibration targets. These activities are necessary to improve the fidelity of contaminant flow and transport simulations in the HTF PA. [Medium Risk Significance, Intermediate Term]

2. Selection of the compliance boundary and loading of the contaminant source cells (i.e., tank cells in the far-field model) could be evaluated to ensure that the dose estimates are not significantly underestimated. [Medium-to-Low Risk Significance, Long Term]

3. Additional information could be collected during the monitoring period to support DOE's modeling treatment of the calcareous zones in the UTRA-LZ. DOE could 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 UTRA-LZ. Site-specific K_ds may also need to be developed for the UTRA-LZ. [Medium Risk Significance, Long Term]

4. DOE should continue to evaluate the appropriateness of selected transport parameters (e.g., soil K_ds) and the selection of sorption models during the monitoring period. [Medium Risk Significance, Intermediate Term]

As a result of the uncertainty in the far-field model, the NRC staff will monitor these items when the HTF PA is revised as part of DOE's PA maintenance program. DOE can address this monitoring area by making appropriate revisions during future HTF PA updates.

4.2.12 Dose Methodology

The dose methodology used by DOE in the HTF PA process is 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 are described in the HTF PA (SRS-CWDA-2010-00128, 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 HTF to a human receptor is expected to be leaching to the groundwater

(after degradation of the grouted tanks and vaults) followed by human consumption and use of well water for domestic purposes. The exposure pathways considered in the HTF PA involving contaminated well water include direct ingestion, ingestion of milk and meat from dairy and beef cattle, as well as meat and eggs from poultry, consuming contaminated well water, and ingestion of plant and animal products that are grown and raised in areas irrigated with contaminated well water.

In the analysis performed by DOE for the HTF PA, parameters for the exposure pathways include bioaccumulation factors which quantify radionuclide transfer between environmental media, duration of time that crops are exposed to radioactivity, consumption rates of contaminated water and foodstuffs, durations of activities in which exposures to radioactivity could occur, and physical parameters of the biosphere environment. DOE documents the development of transfer factors in SRNL-STI-2010-00447, Rev. 0. DOE documents the parameters values for biosphere factors in Section 4.6 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1).

In a teleconference between DOE and the NRC staff on July 3, 2013 (Appendix A of the Enclosure; Mohseni, 2013a [ML13196A135]), DOE responded to a request for clarification from the NRC staff on changes to biosphere modeling between the FTF PA (SRS-REG-2007-0002, Rev. 1) and HTF PA (SRR-CWDA-2010-00128, Rev. 1). DOE identified the following changes from the FTF PA: (1) new pathways are included for chicken and egg ingestion; (2) the leachate factor is based on water introduction from precipitation and irrigation; (3) the garden crop yield is increased from 0.7 to 2.2; (4) stochastic sampling of the transfer factor for beef ingestion is included in the probabilistic analyses; (5) transfer coefficients are aligned with IAEA-472 (IAEA-2010) report; (6) the 15-cm exposure depth external dose conversion factor for Ra is updated by 2 to 3 orders of magnitude; (7) holdup times are ignored; (8) the annual time spent swimming is increased 75 percent; and (9) the annual time spent boating is increased 5 percent.

DOE implements radionuclide dose conversion factors, which convert concentrations of radionuclides in environmental media into radiological doses incurred by a receptor, from the Federal Guidance Reports (FGR) developed by the EPA and International Commission on Radiological Protection (ICRP). 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 are included in Section 4.7.1 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1).

4.2.13 NRC Evaluation of Dose Methodology

In general, the dose methodology implemented in the HTF PA is well supported and suited for the purpose. Many of NRC's guidance documents provide recommendations on the approach and use of the specific dose conversion factors used in the HTF PA process. These include NUREG-1573, which provides guidance on the use of pathway dose conversion factors for calculating doses via the potential exposure pathways, and NUREG-1757 (Appendix I, Vol. 2), 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; EPA, 1993), as well as those developed by the ICRP and published in ICRP-72 (ICRP, 1995).

In Section 4.2.17 of the FTF TER (Camper, 2011 [ML112371751]), the NRC staff identifies several concerns related to DOE's FTF biosphere parameters, specifically: (1) the appropriateness of the use of root vegetable transfer factors for plant transfer factors in the SRS regions, (2) the treatment of uncertainty in plant transfer factors, and (3) the use of a drinking water consumption rate less than 2 L/day. In the HTF PA, DOE addresses the use of root vegetable transfer factors by weighting transfer factors that are available in IAEA-472 (IAEA, 2010) by the estimated percentages of the major plant groups commercially produced in the SRS Area (Section 3.1.2; SRNL-STI-2010-00447, Rev. 0). For the remainder of the NRC staff's previous concerns, the NRC staff continues to recommend, as it did in the FTF TER, that future updates to PAs provide a more technically defensible rationale for these issues.

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 HTF PA to 0.25 mSv (25 mrem) TEDE. The HTF 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 HTF 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 HTF 100-m (330-ft) buffer zone. The all-pathways dose estimated in the HTF PA combines water-dependent pathways originating from contaminated groundwater and air pathways originating from the release of volatile radionuclides from the wasteforms. DOE also performed probabilistic analyses to assess the impacts of variability and uncertainty on a demonstration of compliance with 10 CFR 61.41. Sections 4.2.18 and 4.2.19 of this TER summarize the DOE probabilistic analyses and detail the NRC staff's evaluation of DOE's probabilistic analyses, respectively.

The peak all-pathways TEDE to a member of the public was estimated in the deterministic analysis (from Case A) to be 0.04 mSv/yr (4.0 mrem/yr) occurring at year 8,790 using the highest 100 m (330 ft) groundwater pathway dose results during the 10,000-yr 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 Tc-99. The second ranked primary pathway was vegetable ingestion with 35 percent of the peak dose and the same principal radionuclide. DOE also evaluated time periods out to 100,000 years to assess longer-term impacts to assess uncertainty and sensitivity in the HTF PA outcomes. Results from these evaluations are presented in Table 4-3.

Using the probabilistic model, DOE estimates a peak of the mean all-pathways dose of 0.13 mSv/yr (13 mrem/yr) to a member of the public within 10,000 yrs for Case A. Including the alternative cases, DOE estimates a peak of the mean all-pathways dose of 0.15 mSv/yr (15 mrem/yr) to a member of the public within 10,000 years.

4.2.15 NRC Evaluation of Protection of the Public

The 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 to be acceptable for incidental WDs. DOE's PA indicates that compliance with 10 CFR 61.41 for protection of the public can be demonstrated within 10,000 years. Results from the HTF PA indicate that doses

would exceed 0.25 mSv (25 mrem) TEDE at some point in the future from Ra-226. Given the delayed ingrowth of Ra-226 from its parent radionuclides (Pu-238, U-234, and Th-230), it seems likely that the peak dose from Ra-226 would occur after 10,000 years. On the other hand, the uncertainty and sensitivity analyses show the potential for other radionuclides to dominate the peak dose earlier in time and at levels potentially above the performance objectives, should the assumptions and approaches relied on for Case A prove to be invalid.

Thus, the NRC staff provides several recommendations in this TER 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.

4.2.16 Protection of Intruders

To demonstrate compliance with the 10 CFR 61.42, "Protection of Individuals from Inadvertent Intrusion" performance objective, DOE qualitatively and quantitatively evaluated several potential intruder scenarios. Given the depth and accessibility of the waste, DOE considers the bounding scenarios to be an acute intruder-drilling scenario and a chronic intruder-agricultural scenario. DOE evaluates these scenarios deterministically[19] at the end of the 100-year institutional control period, or later, as described below.

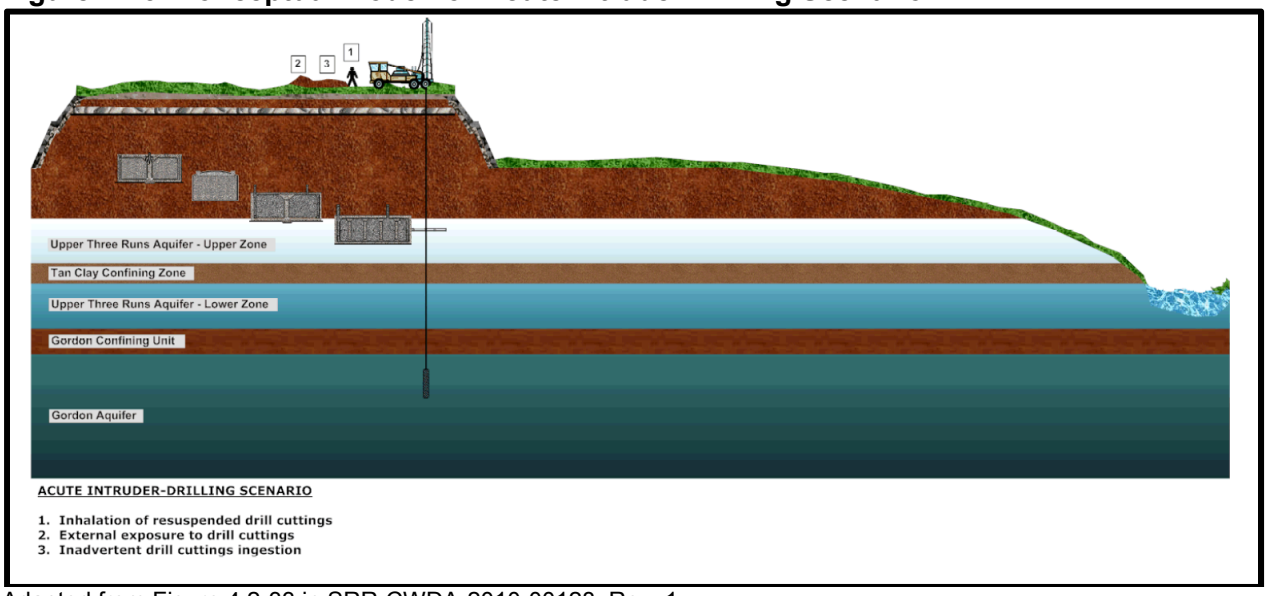
The acute intruder drilling scenario assumes that an intruder drills a well into a 3-in (8-cm) transfer line at the end of the 100-yr institutional control period. Pathways of exposure include external radiation exposure and inhalation and inadvertent ingestion of drill cuttings (see Figure 4-23).

The peak dose for the acute intruder drilling scenario is 0.014 mSv (1.4 mrem) at year 100 and is primarily due to direct exposure to drill cuttings (95 percent). The direct exposure dose is from the principal radionuclides Cs-137/Ba-137m (96 percent) and Sr-90/-Y-90 (3 percent). The radionuclides Sr-90/Y-90, Pu-238, and Am-241 also contribute to the peak dose from ingestion and inhalation, although these pathways cumulatively contribute only 5 percent to the peak dose.

The chronic intruder agricultural scenario assumes that at the end of institutional controls (i.e., 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 contaminated well water. The intruder in the agricultural scenario is also exposed by ingestion and inhalation of well water and garden soil, ingestion of food, and by direct exposure from the contaminated garden soil (see Figure 4-24).

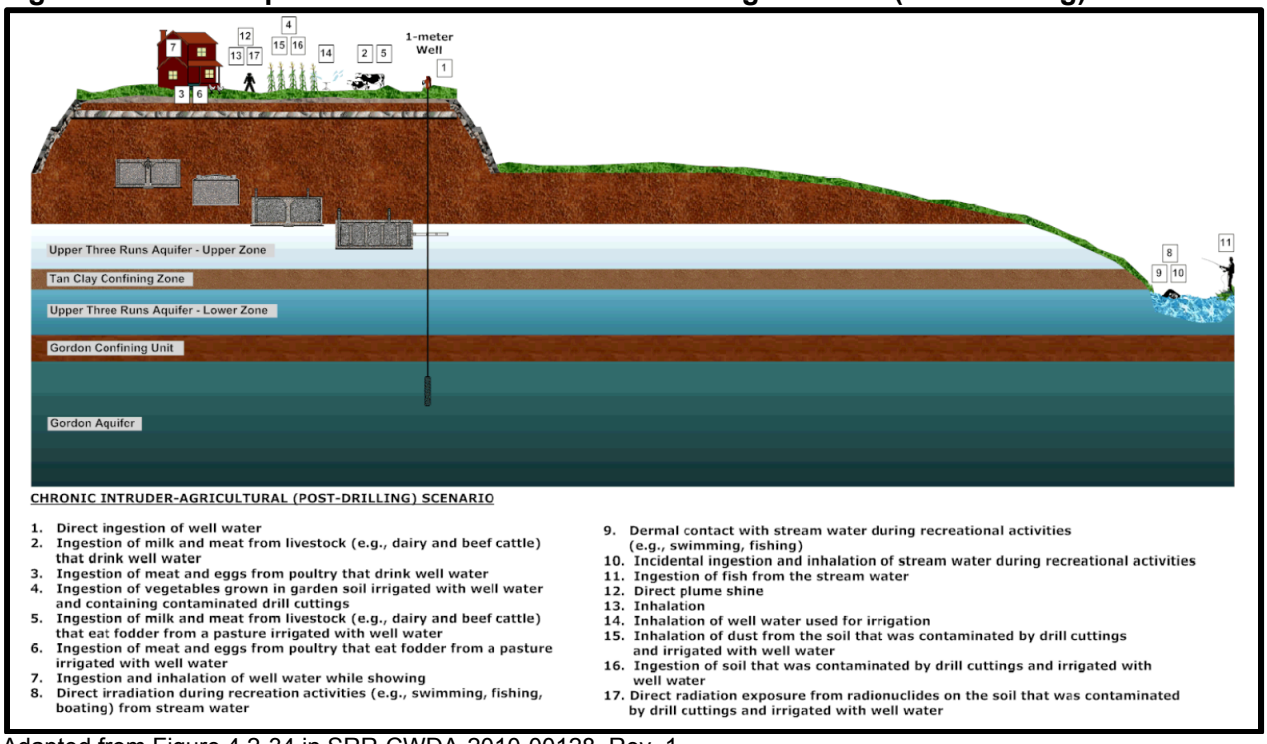
[19] Additional deterministic sensitivity and probabilistic analyses are discussed later in this section.

Figure 4-23 Conceptual Model for Acute Intruder Drilling Scenario



Adapted from Figure 4.2-33 in SRR-CWDA-2010-00128, Rev. 1.

Figure 4-24 Conceptual Model for Chronic Intruder Agricultural (Post-Drilling) Scenario

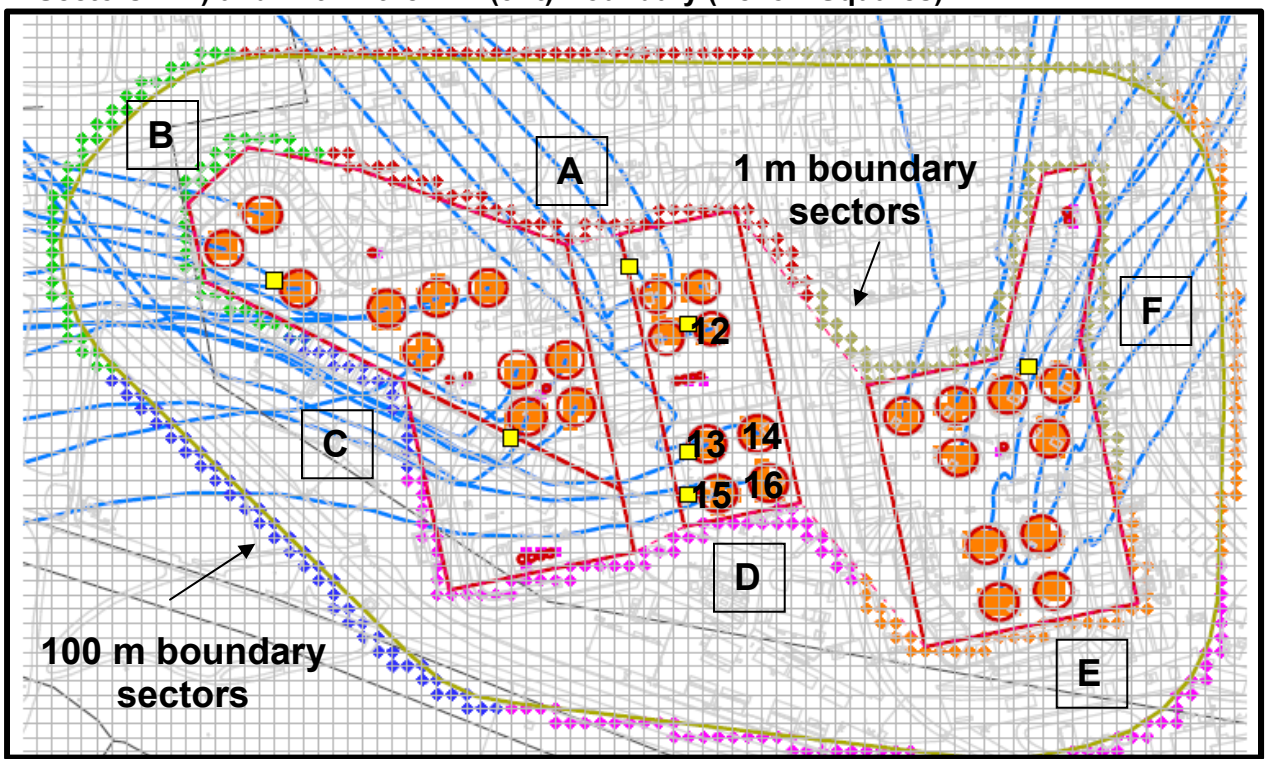


Adapted from Figure 4.2-34 in SRR-CWDA-2010-00128, Rev. 1.

The intruder in the agricultural scenario is also assumed to participate in several recreational activities that incur radiological exposure. Boating, swimming, and fishing are also assumed to occur on site streams (Upper Three Runs and Four Mile Creek) leading to inadvertent intruder doses associated with direct radiation, dermal contact, and incidental ingestion and inhalation of stream water. Ingestion of fish that bio-accumulate radioactivity in stream water is also considered.

The water well in the chronic intruder agricultural scenario is assumed to be located 1 m (3 ft) from the boundary of the HTF. Additional wells are also evaluated 1 m (3 ft) from individual waste tanks in uncertainty and sensitivity analyses. Figure 4-25 illustrates the 1-m (3-ft) boundary wells (grouped in Sectors A-F) and the onsite intruder wells (yellow squares) near individual waste tanks. Peak radionuclide decay chain concentrations in the lower and upper zones of the UTRA and the Gordon Aquifer are evaluated to determine groundwater-dependent doses to an inadvertent intruder. The 1-m (3-ft) boundary points are divided into Sectors A-F to facilitate data storage (time series data for the minimum, maximum, and average concentration of each radionuclide and sector are saved rather than saving data for all points in every sector). Seep-line concentrations are also recorded at site streams to calculate intruder dose from recreational activities.

Figure 4-25 Intruder Well Locations at the 1-m (3-ft) Boundary (Inner Boundary Grouped in Sectors A-F) and Within the 1-m (3-ft) Boundary (Yellow Squares)



Adapted from Figure 6.5-5 in SRR-CWDA-2010-00128, Rev. 1.
Notes: To convert meters to feet, multiply by 0.3.
Also illustrated are member of the public well locations at the 100-m (330-ft) boundary (outer boundary grouped in Sectors A-F) and particle tracks (blue lines).

Compliance points are important because they determine the points at which groundwater concentrations are evaluated. Wells located closer to HTF sources have relatively higher concentrations of leached radionuclides compared to wells located further downgradient from HTF sources. A discussion regarding DOE's selection of the 1-m (3-ft) compliance boundary is found in Section 4.2.11.

The peak dose within 10,000 years to the chronic intruder (agricultural scenario) occurs at 10,000 years in Sector D at 0.51 mSv/yr (51 mrem/yr), with the most important pathway being water ingestion (74 percent of peak dose). The principal radionuclides contributing to drinking water dose are Ra-226 (31 percent), U-234 (29 percent), and U-233 (11 percent). Vegetable ingestion contributes about 23 percent of the peak dose with the same radionuclides contributing. The peak dose within 20,000 years is greater, specifically 2.6 mSv/yr (260 mrem/yr) also in Sector D. No information is reported in the HTF PA on primary pathways or radionuclides related to the 20,000 year dose. Interestingly, the sector with the greatest intruder dose is Sector D for the chronic intruder scenario. This is in contrast to the member of the public[20] dose evaluated at the 100-m (330 ft) boundary (see Figure 4-25) for which the Sector D dose is an order of magnitude lower than the peak dose in Sectors A, B, and C. The difference in the location of the peak dose at the 1-m versus 100-m (330-ft) boundary, where the intruder and member of the public doses are respectively evaluated, is likely related to DOE's selection of the compliance boundary. Although the Sector D wells along the 1-m (3-ft) boundary are perpendicular to the primary flow direction from HTF Tanks 13-16, due to the proximity of Sector D wells to these tanks, the well concentrations are actually higher in Sector D wells compared to wells located directly downgradient but further away from Tanks 13-16 in Sectors A, B, and C. Figure 4-25 illustrates the relative distances between Tanks 13-16 and crossgradient wells in Sector D versus downgradient wells in Sectors A, B, and C.

At 100 years, coinciding with the end of the institutional control period, DOE reports a peak dose to the chronic intruder (agricultural scenario) of around 0.4 mSv/yr (40 mrem/yr) (Figure 6.4-2; SRR-CWDA-2010-00128, Rev. 1). This dose is stated to be primarily due to Sr-90/Y-90 from drill cuttings (Page 769; SRR-CWDA-2010-00128, Rev. 1). No information is provided in the HTF PA regarding the primary pathway for the dose observed year 100. Interestingly, the peak dose in the acute intruder well drilling scenario at 100 years is primarily from Cs-137/Ba-137m, and not Sr-90/Y-90. The HTF PA (SRR-CWDA-2010-00128, Rev. 1) indicates that the dose conversion factor (dose per unit concentration) for Cs-137/Ba-137m is two orders of magnitude higher than that of Sr-90/Y-90 under the acute intruder well-drilling scenario discussion. The dominance of Sr-90/Y-90 in the chronic intruder agricultural scenario is due to the dominance of ingestion pathways in the chronic intruder scenario compared to the dominance of the direct radiation exposure pathway, and Cs-137/Ba-137m, in the acute intruder well-drilling scenario.

[20] Member of the public dose refers to the dose calculated from the evaluation of the 10 CFR 61.41 performance objective. The member of the public dose is evaluated at 100 m (330 ft) from the HTF boundary. This is in contrast to the evaluation of the intruder dose at the 1-m (3-ft) boundary.

DOE considers a number of additional intruder scenarios but finds them (1) to not be applicable or (2) to be bounded by the two scenarios that were evaluated. The intruder-construction, intruder-discovery, and bio-intrusion scenarios are not considered to be applicable because of the depth and form of the waste. Waste residuals will be at least 3 and up to 12 m (10 to 40 ft) below the HTF closure cap and will be contained in protective barriers (e.g., concrete vaults, steel liners, and tank grout) that are assumed to deter intrusion and help to clearly distinguish the waste from the surrounding soil. The chronic intruder-resident scenario and the chronic intruder-recreational hunting/fishing scenario are considered by DOE to be bounded by the intruder-agriculture scenario. The pathways associated with the intruder constructing a house over the containments are considered insignificant by DOE because of the shielding provided by the disposal depth of the contaminants, as well as the waste tanks and ancillary equipment containment.

DOE also performed a probabilistic analysis to better understand uncertainty in intruder dose projections given the uncertainty in several parameter values that influence the intruder dose. Additional deterministic sensitivity analyses[21] were also conducted to better understand uncertainty in the dose results given the uncertainty in key parameters. The deterministic sensitivity analysis results are discussed first.

DOE evaluates two additional sources in additional sensitivity analyses: (1) acute and chronic exposure from drilling into a 10-cm (4-in) transfer line, and (2) acute and chronic exposure from drilling into a tank (Tank 13 is the only tank evaluated). The former source is considered unlikely (only 0.24 percent of the transfer lines are 10-cm [4-in] lines) and the latter source is not considered credible due to the presence of engineered barriers whose long-term strength and durability would preclude intrusion, as well as provide a recognizable wasteform. The sensitivity results indicate doses of 0.03 mSv/yr (3 mrem/yr) for the acute scenario and 0.7 mSv/yr (70 mrem/yr) for the chronic scenario for drilling into a large (10-cm [4-in] versus 8-cm [3-in]) transfer line at 100 years. These doses are approximately 2 times higher than doses from the 8-cm (3-in) transfer line scenario. Peak doses over longer periods of performance when groundwater-dependent doses dominate are not sensitive to assumptions regarding the transfer line diameter. This is because the peak groundwater-dependent dose is dominated by contributions of the waste tanks to radiological contamination at the intruder wells and not due to contributions of drill cuttings or groundwater contamination from the transfer lines. A potential intruder could get a dose of 0.14 mSv/yr (14 mrem/yr) from acute exposure at 500 years and a dose of 1 mSv/yr (100 mrem/yr) from chronic exposure from drilling into Tank 13 at 500 years. Doses associated with chronic exposures to a potential intruder are increasing beyond 1 mSv/yr (100 mrem/yr) at 10,000 years.

In addition to considering alternative scenarios, DOE also quantitatively considers different receptor locations (i.e., intruder wells located within the 1-m [3-ft] boundary). Seven locations are evaluated as illustrated with yellow squares in Figure 4-25. The location of the intruder well has a very significant impact on the chronic intruder dose projections. The highest dose occurs from contributions from the well nearest Tank 12 (peak dose of 3 mSv/yr [300 mrem/yr]) versus

[21] Deterministic sensitivity analyses vary one parameter at a time.

the dose at the 1-m (3-ft) boundary along Sector D that is located adjacent to Tanks 13-16 (0.5 mSv/yr [50 mrem/yr]). Figure 4-26 illustrates the variability in peak dose based on intruder well location, as well as major radionuclides contributing to the peak chronic intruder dose at the intruder well near Tank 12.

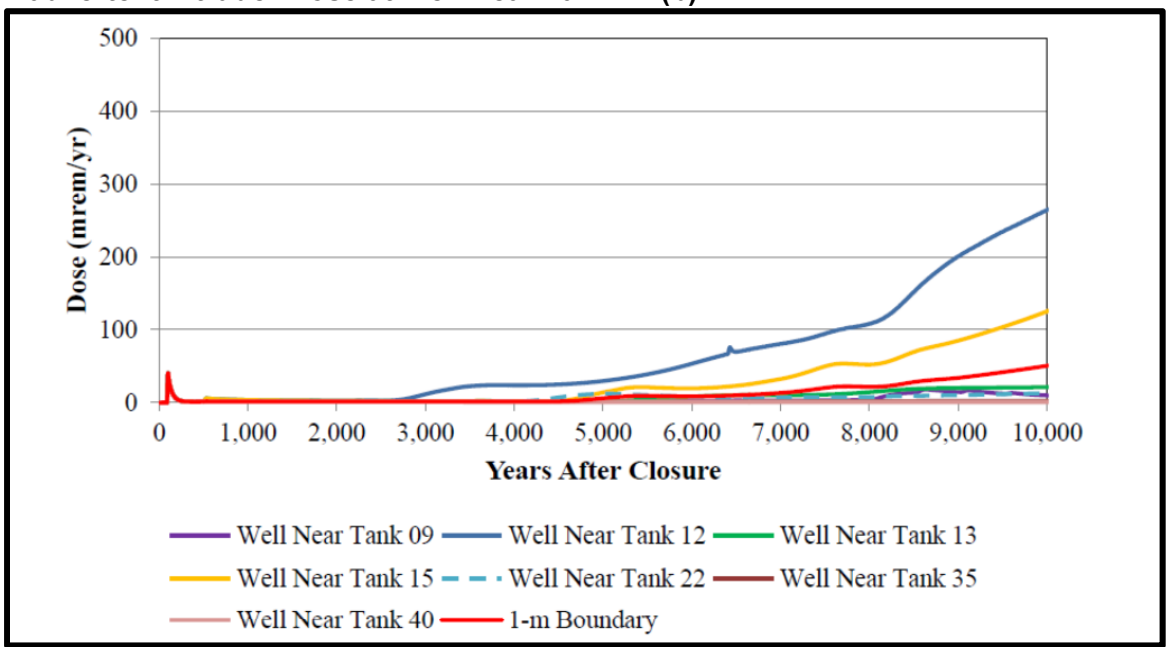
Results of the probabilistic intruder analysis indicated peak of the mean doses of 7.62 mSv/yr (762 mrem/yr) for the chronic intruder [and median doses of 5 mSv/yr (500 mrem/yr)] associated with groundwater-dependent pathways and waste release from the tanks within a 10,000-year performance period. DOE indicates that projected doses are most sensitive to well-completion stratum and parameters related to the release of Tc-99 (e.g., Tc-99 inventory multiplier), which is similar to results of the member of the public sensitivity analysis. The peak of the mean of the probabilistic assessment is greater than the performance objective of 5 mSv/yr (500 mrem/yr). DOE indicates that the results of the probabilistic analysis are higher than the deterministic analysis due to differences in modeled flow paths (i.e., pathways through the annulus in PORFLOW™, which is used for the deterministic analyses, are longer due to horizontal flow). Furthermore, DOE indicates that a few realizations sampled at the tail of the parameter distributions could be biasing the mean dose high.

4.2.17 NRC Evaluation of Protection of Intruders

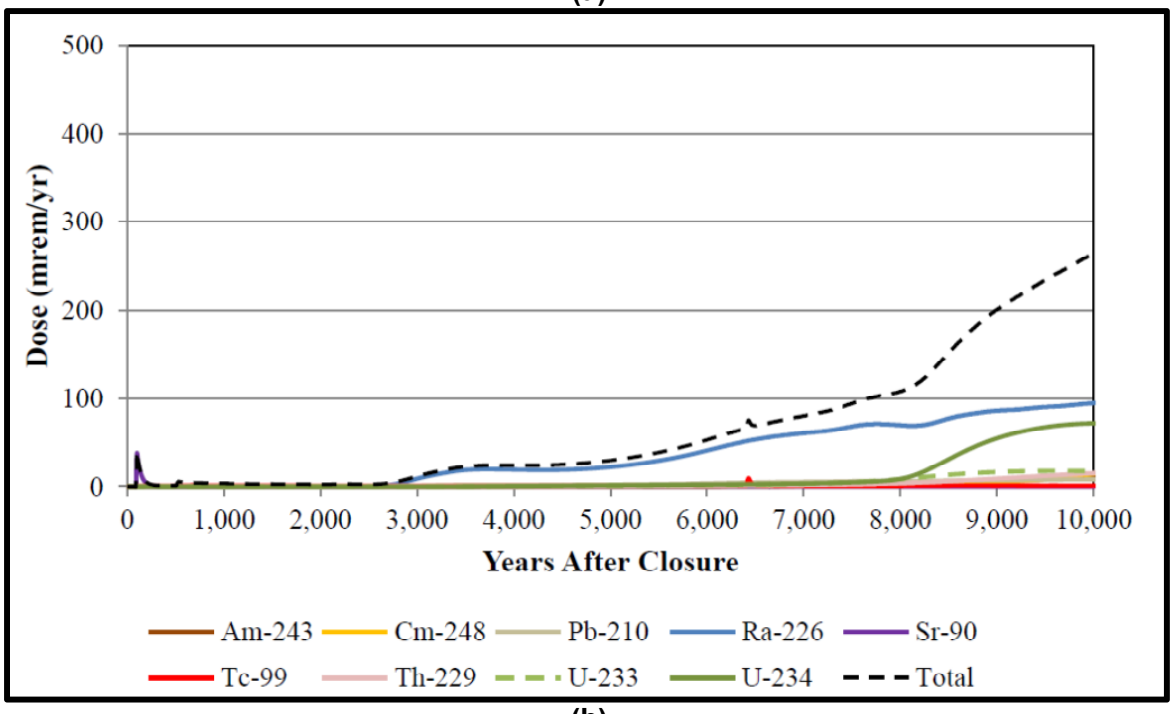
No dose limit is provided in 10 CFR 61.42 for protection of the inadvertent intruder; however, a dose of 5 mSv/yr (500 mrem/yr) guided the development of waste classification requirements designed to protect the inadvertent intruder as described in the Draft Environmental Impact Statement (NUREG-0782) for the 10 CFR Part 61 rulemaking. Consistent with the review procedures in NUREG-1854 (Chapter 5) the NRC staff uses a dose limit of 5 mSv/yr (500 mrem/yr) when evaluating DOE's inadvertent intruder analysis results.

DOE performs inadvertent intruder analyses both deterministically and probabilistically with differing results. While DOE's deterministic modeling results indicate that the 10 CFR 61.42 performance objective can be achieved (i.e., all intruder doses are calculated to be less than 5 mSv/yr (500 mrem/yr) in deterministic analyses), DOE's probabilistic modeling results show that doses could be greater than 5 mSv/yr (500 mrem/yr). DOE attributes the differences in deterministic and probabilistic results to differences in modeled flow paths between the two-dimensional PORFLOW™ and one-dimensional GoldSim™ models used to perform the deterministic and probabilistic analyses, respectively. However, differences between the GoldSim™ and PORFLOW™ models were shown to primarily affect the timing and not necessarily the magnitude of the peak dose in supporting documentation. Based on information provided in SRR-CWDA-2010-00093, Rev. 2, that shows similar peak dose output from GoldSim™ and PORFLOW™ for a comparable case (i.e., an intruder well located near Tank 12), the NRC staff expects the difference in the probabilistic and deterministic results to be less pronounced than indicated in the HTF PA (SRR-CWDA-2010-00128, Rev. 1).

Figure 4-26 Chronic Intruder Doses Evaluated Within the 1-m (3-ft) Boundary At Various Locations (a) and Radionuclides Contributing More Than 0.05 mSv/yr (5 mrem/yr) to Inadvertent Intruder Dose at Well Near Tank 12 (b)



(a)



(b)

Adapted from Figures 6.5-6 and 6.5-7 in SRR-CWDA-2010-00128, Rev 1.
Note: To convert mrem/yr to mSv/yr, multiply by 0.01.

Considering uncertainty in the timing of peak dose in the deterministic PORFLOW™ analyses[22], the complicated flow paths observed in PORFLOW™ that may delay the timing of peak dose require additional review. The NRC staff also observes differences in the level of attenuation between the two models when evaluating intermediate results between the source and exposure point locations. DOE notes, in the HTF PA, that differences between the two models are more prominent closer to the source and that additional aquifer mixing leads to more comparable results at the 100-m (330-ft) boundary. Adjustments made during benchmarking to align GoldSim™ and PORFLOW™ results may have muted the significant difference in breakthrough times observed in the inadvertent intruder analysis performed for Tank 12. Due to significant differences between the GoldSim™ and PORFLOW™ modeling results, the NRC staff plans to continue to evaluate the PORFLOW™ modeling assumptions and results for the compliance case (Case A) during the monitoring period to provide confidence that the timing of peak dose is not artificially delayed.

DOE develops reasonable inadvertent intruder scenarios to demonstrate compliance with 10 CFR 61.42. However, DOE did not originally evaluate alternative cases in its HTF PA that could hasten the time to risk-significant releases and increase the magnitude of the peak dose. In response to CC-INT-1 and CC-INT-2, DOE provided additional results for alternative cases (SRR-CWDA-2013-00106, Rev. 1). These results indicate that if more pessimistic assumptions are made with respect to engineered barrier performance, the inadvertent intruder doses could be greater than 5 mSv/yr (500 mrem/yr).

For example, alternative Case E results indicate a peak dose of 10 mSv/yr (1,000 mrem/yr) at around 2,000 years. The peak Case E dose is attributed to Np-237. Because Np-237 has a relatively high solubility under oxidizing conditions and high mobility in the natural environment, only the tank basemats provide a significant barrier to Np-237 release when the system becomes oxidized. Therefore, when the attenuating properties of the basemat are degraded in Case E, the Np-237 dose is significantly greater than in the compliance case, Case A.

It is important to note that the peak dose reported in DOE's RAI response for Case E does not consider uncertainty in the selection of the compliance boundary that has also been shown to have a significant impact on the inadvertent intruder doses. For example, intruder wells located next to tank sources result in doses a factor of 5 times higher than peak doses at other well locations (e.g., the Tank 12 intruder well location is shown to have a peak dose five times higher than the highest dose at the 1-m [3-ft] boundary-well location in the HTF PA). Figure 4-26 illustrates the variability in chronic intruder dose based on intruder well location and demonstrates that the doses are still increasing at 10,000 years.

[22]Peak groundwater dose continues to increase between 10,000 and 20,000 years in the chronic intruder analysis (see Figure 6.4-4; SRR-CWDA-2010-00128, Rev. 1) and peak dose is still increasing at Tank 12 between 10,000 and 20,000 years (see Figure 7.1-15 in SRR-CWDA-2010-00093, Rev. 2), while peak dose occurs significantly earlier around 13,000 years at Tank 12 in the GoldSim™ model (see Figure 7.1-15; SRR-CWDA-2010-00093, Rev. 2).

Additionally, the NRC staff notes that while the seven intruder well locations selected by DOE for evaluation in the 10 CFR 61.42 analysis appear to be reasonable, the location of elevated intruder well concentrations will change over time as the projected inventories used in the HTF PA for each tank type are replaced with final inventories. Finally, the NRC staff notes technical concerns with respect to DOE's selection of the 1-m (3-ft) compliance boundary in Section 4.2.11. The NRC staff plans to monitor DOE's support for the selected compliance case (Case A) and the likelihood of alternative cases. The NRC staff also plans to monitor the impact of the selection of the compliance boundary[23] or compliance wells, as HTF closure progresses.

DOE indicates in the CC-INT-1 response that only sensitivity run radionuclides (identified in Table 5.2-9 of SRR-CWDA-2010-00128, Rev. 1) are considered (SRR-CWDA-2013-00106, Rev. 1). Lack of consideration of other key parameters may be problematic because short-lived radionuclides such as Sr-90 and Cs-137 are not listed as sensitivity run radionuclides in Table 5.2-9 in the HTF PA (SRR-CWDA-2010-00128, Rev. 1)[24]. Although Sr-90 is listed in CC-INT-1 figure legends, Cs-137 is not listed and it is not clear if Sr-90 contributions from both drill cuttings and *groundwater* sources are considered. Therefore, DOE's evaluation of the contributions of short-lived radionuclides to intruder dose may be incomplete.

In general, the NRC staff believes the potential dose contributions of short-lived radionuclides, such as Cs-137 and Sr-90, have not been fully evaluated by DOE in alternative cases (see Section 4.2.9). Of particular concern to the NRC staff is the risk-significant inventory of Sr-90 located in HTF tank annuli, such as Tanks 9, 10, 14, and 16. Evidence of the potential risk significance of Sr-90 in tank waste is found in alternative case results. For example, HTF PA Table 5.6-35 includes two realizations for Case D that have a large Sr-90 dose in the range of 10s of mSv/yr (1,000s of mrem/yr) for 10 CFR 61.41, member of the public, doses evaluated at 100 m from the disposal facility boundary. While the assumed Sr-90 inventory in the Case D realizations are up to an order of magnitude greater than the Tank 9, 10, 14, and 16 annuli inventories, the higher dose in the Case D realizations are due to a higher assumed inventory. This dose would be offset by even higher doses when exposure point concentrations are evaluated at 1 m (3 ft) from the disposal facility boundary [or 1 m (3 ft) from HTF tank sources], as would be the case in an inadvertent intruder analysis. Another important factor associated with the high Sr-90 dose in the Case D realizations is the lack of an infiltration-reducing closure cap. However, Sr-90 inventories associated with failed liners that are fully or partially submerged may experience risk-significant release early on in the performance period due to groundwater infiltration and/or drainage, irrespective of assumptions regarding the performance of the closure cap.

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[23] See additional discussion in Section 4.2.11 "NRC Evaluation of Hydrology and Far-Field Transport" regarding selection of the compliance boundaries.

[24] U-233 was also found to be a key radionuclide for inadvertent intruder dose reported in the HTF PA. However, this radionuclide is similarly omitted from the list of sensitivity run radionuclides in Table 5.2-9 in the HTF PA (SRR-CWDA-2010-00128, Rev. 1). The contributions of U-233 to the Case E results are, therefore, not clear.

Finally, other uncertainties associated with groundwater-dependent pathways discussed in the 10 CFR 61.41 evaluation (e.g., solubility of residual waste in the CZ, release of annulus and sand pad inventories, sorption of radionuclides to cementitious material and soil) also apply to the NRC staff's evaluation of 10 CFR 61.42. Although a higher allowable dose of 5 mSv/yr (500 mrem/yr) is considered acceptable for protection of the inadvertent intruder compared to the 10 CFR 61.41 member of the public analysis[25], differences in the level of attenuation between the 1-m (3-ft) and 100-m (330-ft) compliance boundaries may make compliance with the 10 CFR 61.42 performance objective more limiting. In other words, attenuation between the 1-m (3-ft) and 100-m (330-ft) compliance points may be greater than a factor of 20 (the difference between the allowable dose for the 10 CFR 61.41 and 61.42 performance objectives), particularly for short-lived, slow-moving radionuclides. The NRC staff plans to monitor technical issues related to the 10 CFR 61.41 evaluation, as well as technical issues unique to the 10 CFR 61.42 evaluation, as HTF closure progresses to assess compliance with the 10 CFR 61.42 performance objective.

4.2.18 Uncertainty and Sensitivity Analyses

DOE includes extensive uncertainty and sensitivity analyses in its HTF PA documentation (SRR-CWDA-2010-00128, Rev. 1) to study (1) the sensitivity of key modeling results to changes in parameter values, (2) the uncertainty in dose projections, and (3) the contributions of various natural and engineered barriers to reducing HTF risk. DOE uses the PORFLOW™ and GoldSim™ codes[26] to develop HTF PA models from which uncertainty and sensitivity analyses are conducted.

Additional details regarding HTF PORFLOW™ model construction are provided in Sections 4.2.8 and 4.2.10. The deterministic HTF PORFLOW™ model is used to support a sensitivity analysis in which one parameter is varied at a time to identify the most influential parameters affecting model outputs, such as the dose to a member of the public. The deterministic HTF PORFLOW™ model is also used by DOE to perform a barrier analysis to better understand how multiple barriers work alone, or in concert with other barriers, to reduce risk.

DOE uses the software program GoldSim™ to construct a probabilistic model, which is described in the next section. In a probabilistic model, multiple parameter values are varied at the same time to better understand the uncertainty in dose projections over a plausible range of parameter space. Uncertain parameter values are sampled using Latin Hypercube methods (McKay et al., 1979; SAND-79-1473) from pre-determined distributions. Each combination or

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[25] The inadvertent intruder dose limit of 5 mSv/yr (500 mrem/yr) represents a factor of 20 higher allowable dose compared to the 0.25 mSv/yr (25 mrem/yr) dose limit specified in 10 CFR 61.41 for the member of the public.

[26] PORFLOW™ and GoldSim™ *codes* are used by DOE to develop *models* of contaminant flow and transport at HTF to demonstrate compliance with the performance objectives in 10 CFR Part 61, Subpart C. The NRC staff's reference to PORFLOW™ and GoldSim™ models in this section is meant to refer to models constructed by DOE for HTF using these codes, and is not meant to imply that the PORFLOW™ and GoldSim™ codes contain built-in models for HTF.

set of sampled parameter values, called realizations, are run in GoldSim™. Typically, thousands of realizations are run and statistical analysis is performed on the resulting output. Results extracted from the probabilistic model are also interrogated using a "Gradient Boosting Algorithm" (Friedman, 2002) to identify important parameter values that have the most influence on dose to a member of the public and other important modeling outputs.

Deterministic PORFLOW™ and probabilistic GoldSim™ models developed by DOE for HTF are also used to evaluate, in a limited sense, conceptual model uncertainty through the evaluation of alternative cases (see Table 4-11). Conceptual model uncertainty rests, for example, with a lack of knowledge regarding how a system should be represented or how it might evolve over time (e.g., matrix, fracture, or hybrid approach to modeling flow and transport; or variable sequencing of engineered barrier failure or failure paths). DOE considers different states or conditions of HTF engineered barriers such as early tank grout failure or the bypassing of concrete materials underneath the tank that would otherwise tend to attenuate or reduce the impacts of releases from the tanks. Alternative cases are evaluated in PORFLOW™ and GoldSim™ and presented individually. Alternative cases are also evaluated in combination in GoldSim™. The combined "All Cases" results are weighted by the expected likelihood of occurrence for each alternative case.

The GoldSim™ models can be considered an abstraction or simplification of the more complex, multi-dimensional PORFLOW™ models. Benchmarking is performed to compare intermediate and final outputs extracted from the 3-D PORFLOW™ and 1-D GoldSim™ models. Adjustments are then made to the 1-D GoldSim™ model to better align the probabilistic modeling results with the more complex PORFLOW™ models. The following sections present a summary of (1) DOE's approach to performing uncertainty and sensitivity analyses for the HTF, (2) uncertainty and sensitivity analysis results, and (3) discussion regarding key findings related to HTF barriers and performance.

4.2.18.1 Probabilistic Model Overview

DOE prepared a probabilistic uncertainty and sensitivity analysis to support the HTF PA (SRR-CWDA-2010-00128, Rev. 1) and draft basis document for the WD for HTF (DOE/SRS-WD-2013-001, Rev. 0) using the GoldSim™ modeling platform. GoldSim™ is a graphics-based, object-oriented computer program designed to carry out dynamic, probabilistic simulations (GoldSim Technology Group LLC, 2010). GoldSim™ contains many built-in functions and tools to allow users to perform a wide range of simulations related to topics as diverse as financial planning and contaminant transport modeling. The NRC staff recognizes that the resource demand associated with execution of the probabilistic analysis using the deterministic PORFLOW™ models is likely prohibitive given (1) the size of (number of elements in) the near- and far-field models, (2) the number of radionuclide chains being simulated, and (3) the number of scenarios being considered in DOE's HTF PA. The computational burden of executing a probabilistic analysis for a complex problem that is the scope of the HTF facility is somewhat alleviated through the use of simpler system-level models such as DOE's HTF GoldSim™ models, because the GoldSim™ code is more efficient at exploring important model sensitivities and predictive uncertainties than the PORFLOW™ code. Nonetheless, use of a simpler model, such as GoldSim™, to perform the probabilistic analysis is not without its own limitations and disadvantages.

Table 4-11 Summary of Alternative Cases Evaluated in Supplemental Deterministic and Probabilistic Analyses

	A	B	C	D	E
Description (Successive Degradation)	Base case	Case A with fast flow path through tank	Case B with faster chemical transition times*	Case B with fast flow through tank *and basemat*	Case D with faster chemical transition times*
Tank Grout—Hydraulic Failure	Variable by tank type (e.g., complete degradation at 13,000 years [Type I] to 64,000 years [Type IV])†	Instantaneous degradation at 500 years	Same as base case (Case A)	Instantaneous degradation at 500 years	Same as base case (Case A)
Tank Grout or CZ—Chemical Failure	Variable by tank type and radionuclide based on the number of displaced pore volumes flowing through reduced grout‡	Faster chemical transition times compared to base case (Case A)	Fastest chemical transition times (i.e., significantly faster chemical transition times than Cases A, B, and D)	Faster chemical transition times compared to base case (Case A)	Fastest chemical transition times (i.e., significantly faster chemical transition times than the Cases A, B, and D)
Steel Liner Failure	Variable by tank type (e.g., 3,600 yrs [median, Type IV] and 11,000 years [median, Type I]). Median values are used in deterministic analysis	More rapid failure than base case (e.g., 75 years [median, Type IV] and 2500 years [median, Type II]. Median values are used in the deterministic analysis	Same as Case B	Same as Case B	Same as Case B

Table 4-11 (continued) Summary of Alternative Cases Evaluated in Supplemental Deterministic and Probabilistic Analyses

	A	B	C	D	E
Probability	0.75	0.05	0.15	0.0125	0.0375

*DOE assumes in Cases C and E that the cementitious materials hydraulically degrade slowly over time similar to Case A; however, unlike Case A, groundwater is assumed to flow around and by-pass the reducing and buffering capacity of the tank grout leading to rapid chemical transitions to generally higher solubility. Because in Cases B and D, DOE assumes the cementitious materials fail relatively early in the simulation, after 500 years, groundwater is assumed to flow through the tank grout leading to more prolonged chemical transitions compared to Cases C and E (see Section 4.2.8 for additional details on chemical transitions), although the chemical transitions times are faster than Case A.

†DOE assumes, in deterministic Case A, that tank grout degrades slowly over time (e.g., beginning at 800 years and ending at 64,000 years for Type IV tanks). During the degradation period, the conductivity of the tank grout continues to increase until the tank grout is no longer a barrier to flow (see Section 4.2.8 for additional details). Different hydraulic failure rates are assumed in the probabilistic GoldSim™ model including either: (1) base case; or (2) two times slower, or (3) two times faster hydraulic failure rates compared to the base case.

‡The solubility and mobility of key radionuclides is variable over time. DOE considers three chemical states that reflect the chemical evolution of the tank grout. These chemical states are named, in chronological order, Reduced Region II, Oxidized Region II (ORII), and Oxidized Region III (ORIII). The solubility and K_d of each radionuclide can be different for each of the three chemical states. Chemical transitions occur after a fixed number of replacements (or flushes) of the grout pore water with infiltrating groundwater. For example, the first chemical transition to ORII occurs at 523 pore volumes and the second transition to ORIII occurs at 2,119 pore volumes for non-submerged tanks (see Section 4.2.8 for additional details).

For example, GoldSim™ is not designed to efficiently 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 that is described in more detail in Section 4.2.17.

In addition to the evaluation of parameter uncertainty, the probabilistic assessment also considers alternative scenarios or cases related to engineered barrier and vadose zone performance. As indicated in Section 4.2.8, DOE considers alternative cases primarily related to potential variability in engineered barrier performance and waste release. DOE simulates and provides results for alternative configurations (Cases B–E) in its deterministic and probabilistic assessments, thereby, supplementing the results of its compliance case, Case A. Table 4-11 provides summary information regarding the alternative configurations.

Modifications to Previous Probabilistic Models

Modifications were made to the FTF GoldSim™ model to include additional complexity for the HTF GoldSim™ model. For example, FTF probabilistic models did not account for upwards transport of key radionuclides from the contaminated zone into the tank grout above the contaminated zone via advection and diffusion. In certain cases[27], this phenomenon was found to be important in PORFLOW™ near-field modeling simulations, as it led to the sequestration of key radionuclides in the tank system. To account for upward transport in the HTF model, three concentric sets of 20 mixing cells each that represent the tank grout are connected in series including: (1) an inner cylinder that experiences primarily downward flow, (2) an outer cylinder that captures upward flow prior to liner failure, and (3) a fast flow pathway along the tank wall. The inner and outer cylinder representation is consistent with the flow field observed in PORFLOW™ modeling output for Type IV tanks, where downwards flow is significant in the center of the tank and upward flow is more significant along the tank wall prior to liner failure.

Because Type I and II tanks[28] at HTF have experienced leaks from their primary liners into their annuli, a more complex model is needed to simulate the transport of radionuclides through the tank vaults. Figure 4-27 provides a schematic of the transport pathways included for Type I or Type II tanks in the HTF GoldSim™ model. Prior to liner failure, the controlling release pathway out of a tank annulus is through the vault wall. After liner failure, the remaining inventory is released from the annulus to the underlying basemat.

[27]Upward transport of radionuclides into the tank grout is primarily important (1) for Type IV tanks that have no top liner to prevent in-leakage and (2) at early times before the liner fails.
[28] Tanks 9-12 are Type I tanks, Tanks 13-16 are Type II tanks.

Finally, the HTF tank basemats are represented with up to 30 vertical mixing cells[29] to better simulate the transport of key radionuclides out of the tank vaults. DOE observed that the flux results from previous versions of the probabilistic GoldSim™ model, which represented the basemat with fewer cells, were significantly different than similar PORFLOW™ flux results for certain highly-sorbing radionuclides such as Np-237. In the updated GoldSim™ model, all tank types have at least two sets of concentric vertical mixing cells representing a fast-flow cylinder in the center of the basemat and an outer cylinder representing the remaining portion of the basemat underneath the primary liner. Type I and II tanks require additional vertical transport pathways representing that portion of the basemat underneath (1) the annulus and (2) the vault wall (see Figure 4-27). The cylinder under the vault wall takes the material properties of the wall and appears, for at least some tanks, to actually be an extension of the vault wall and not part of the basemat.

To model flow through the unsaturated zone, DOE performs a parameter study of (1) the presence and characteristics of a fast flow path through the tank system (partial, full, or no fast flow path), (2) liner failure time (time zero, early, moderate, or late liner failure), (3) cementitious material degradation (slow, nominal, or fast cementitious material degradation), and (4) presence of an infiltration-reducing cap (nominal cap or no-cap) to produce 72 sets of flow rates for each tank type (Type I, II, IIIA, and IV) and PORFLOW™ time interval (40 total time intervals). Table 4-12 summarizes the cases examined for the parameteric study. The compliance case, Case A, represents a case with no fast-flow pathway.

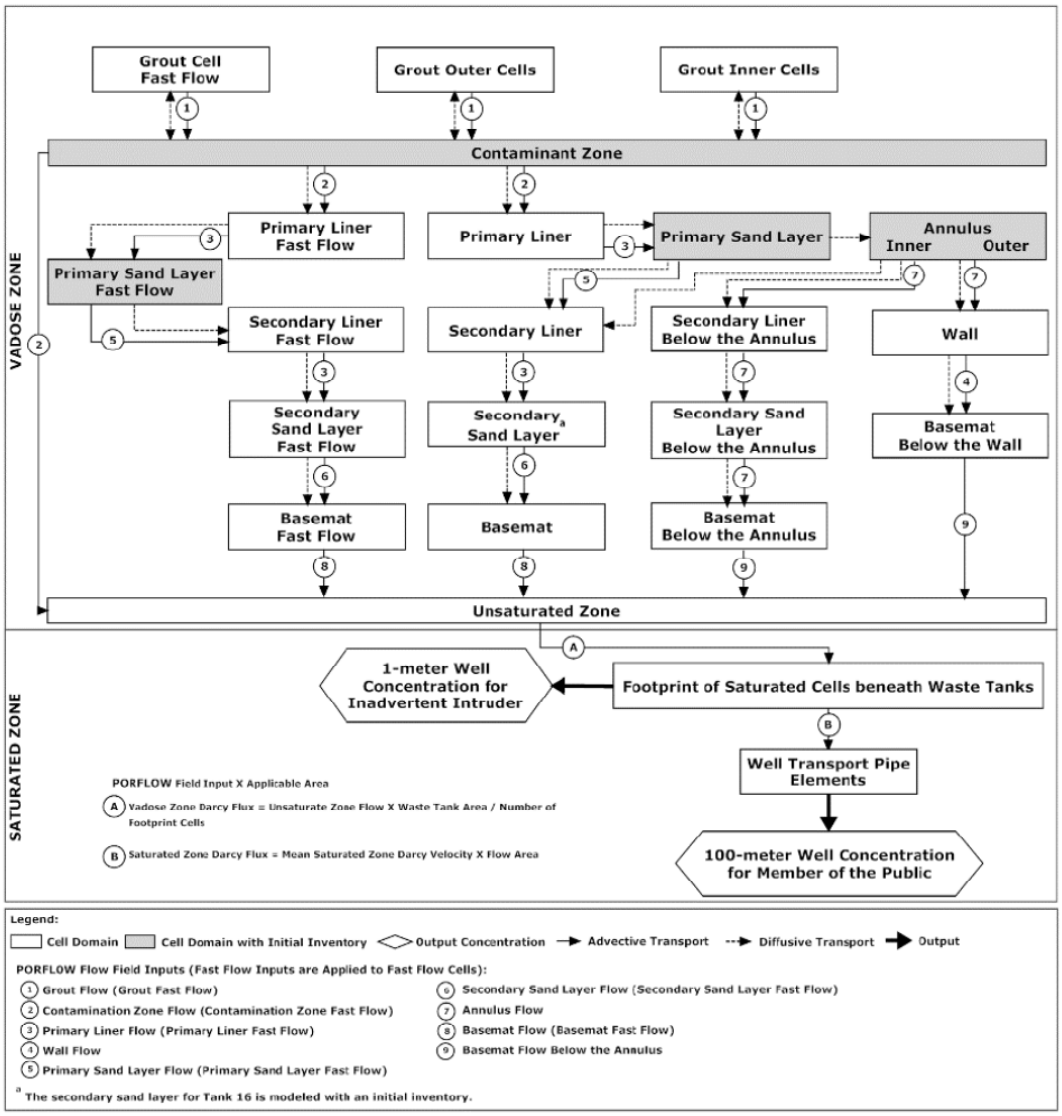
Cases B and C include a partial fast-flow pathway (Cases B and C are grouped together), while Cases D and E include a full fast-flow pathway (Cases D and E are grouped together) consistent with the conceptual models for these alternative cases. The sampled liner failure times for a particular realization may differ from the liner failure time evaluated in the parametric study. Flow in the selected parametric case is scaled from time-zero to the liner failure time to match the sampled failure time to maintain consistency.

Probabilistic Model Flow Velocity Abstraction

One primary difference between the alternative cases evaluated in PORFLOW™ in the deterministic analysis and the 72 parametric cases evaluated in GoldSim™ in the probabilistic analysis is that the cementitious materials are not assumed to fail at 500 years in Cases B and D. Rather, only the base case hydraulic degradation times and a fast and slow case that are a factor of two higher or lower than the base case are simulated. Additionally, chemical transition times are not listed as an independent parameter in the study. Chemical transition times appear to be calculated for the 72 parametric flow cases in GoldSim™ based on the PORFLOW™-generated flows. However, the NRC staff was unable to confirm, prior to preparation of this TER, which material (the tank grout or the contaminated zone) controls the

[29] The number of vertical mixing cells used to represent the basemat for each tank type was determined during benchmarking (i.e., the number of vertical mixing cells was varied to obtain a better match between PORFLOW™ and GoldSim™ simulations).

Figure 4-27 Schematic of Type I and II HTF Tank Transport Pathways



Adapted from Figure 3.1-2 in SRR-CWDA-2010-00093, Rev. 2.

solubility of radionuclides in the contaminated zone, as cases representing different conceptual models for groundwater conditioning are grouped together in the probabilistic analysis.[30]

[30] In deterministic analyses, Cases C and E chemical transitions are controlled by the contaminated zone chemistry, while Cases B and D are controlled by the tank grout chemistry. However, in the probabilistic analysis, Cases B and C are lumped together and Cases D and E are lumped together making it difficult to determine what material controls the contaminated zone chemical transitions and solubility changes.

Table 4-12 Parametric Flow Cases

	Steel Liner Lifetime	Cementitious Material Hydraulic Degradation	Closure Cap	Presence of Fast Flow*	Number of Combinations
	Initially Failed	Half Nominal (Slow)	Nominal	Case A	
	Early Failure	Nominal	No Cap	Case B/C	
	Moderate	Double Nominal (Fast)	—	Case D/E	
	Late Failure	—	—	—	
No. of Choices	**4**	**3**	**2**	**3**	**72 per tank type**

*The "Case" is sampled consistent with the probabilities listed in Table 4-11. The "Case" determines which flow field is selected when running the HTF GoldSim™ model probabilistically. However, the Cases run in the HTF GoldSim™ model are not necessarily consistent with the descriptions in Table 4-11 that apply to the PORFLOW™, deterministically-run Cases. For example, the cementitious material failure times reported in Table 4-11 include only the base case failure time or a 500 year failure time, and do not include the "slow" and "fast" hydraulic failure times evaluated in the parametric flow study. It appears that the chemical transition times are a function of the flow rates used in the parametric flow study; however, the NRC staff was unable to determine if contaminated zone chemical transition times are controlled by the tank grout or contaminated zone chemistry.

To model flow through the saturated zone, Darcy velocities are also calculated for each tank based on PORFLOW™ modeling results. Stream trace calculations that provide data on travel times to the compliance boundary for a conservative tracer are used to calculate saturated flow velocities. It is important to note that DOE uses lateral distance to the compliance boundary to calculate the Darcy velocity. Therefore, the harmonically-averaged Darcy velocity used in GoldSim™ is biased low compared to the actual PORFLOW™ modeled velocity as it does not account for vertical transport through the aquifer that may significantly increase travel distance to the point of compliance. However, the same lateral distance used to calculate the Darcy velocities is used as the pathway length, thereby, ensuring the actual travel times to the point of compliance are accurate. Because Darcy velocity also influences the degree of aquifer mixing, uncertainty in the Darcy velocity can also be important to the results of the analysis. In fact, Darcy velocity is the subject of a separate DOE sensitivity analysis conducted in GoldSim™, as described in more detail below.

DOE only models flow in the UTRA in the probabilistic GoldSim™ model. In the case of a well completed in the Gordon Aquifer, DOE assumes a simple ratio between the UTRA concentration actually being simulated in the GoldSim™ model and the concentration in the Gordon Aquifer. The Gordon Aquifer concentration and dose is assumed to be a factor of 100 less than the UTRA concentration based on simulations run in PORFLOW™. DOE estimates that the probability of well completion in the Gordon Aquifer is 44 percent, based on an investigation of local records. Thus, the peak dose in 44 percent of the realizations contributes little to the average peak dose (i.e., the average peak dose is approximately 44 percent less than it would be if the location of maximum exposure were evaluated in all realizations).

4.2.18.2 Probabilistic Model Benchmarking

As discussed above, the purpose of the benchmarking process is to align the PORFLOW™ and GoldSim™ modeling results to make comparisons between the deterministic and probabilistic analyses more meaningful. Three modeling outputs are used as metrics by which to assess agreement between the two independent models and to better align GoldSim™ to PORFLOW™ modeling results: (1) contaminant flux to the saturated zone, (2) contaminant concentration or dose at the 100-m (330-ft) point of assessment, and (3) public dose versus time.

Although the DOE 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 (PORFLOW™) predictive model results are used to make adjustments to the GoldSim™ model. An underlying assumption in the benchmarking process, therefore, is that the more complex PORFLOW™ model adequately represents the HTF flow and transport system such that adjustments should only be applied to the simpler, more conservative GoldSim™ model to produce comparable results. The NRC staff performs a separate evaluation of the adequacy of the PORFLOW™ models in Sections 4.2.9 and 4.2.11.

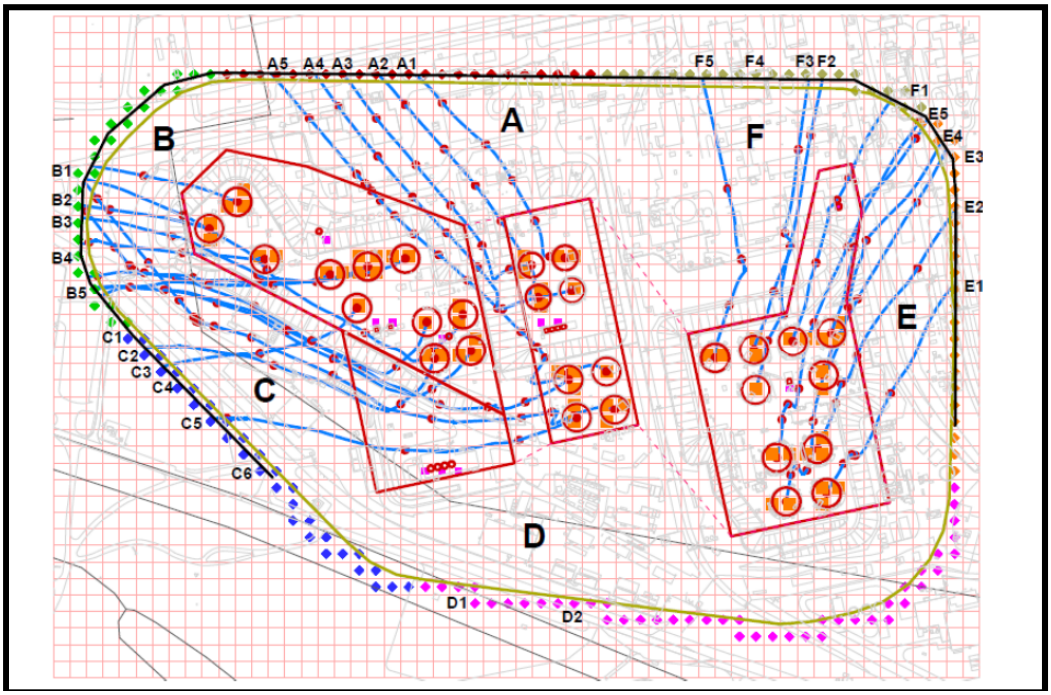
DOE uses a built-in plume function[31] to assess the contributions of HTF sources to well concentrations along the 100-m (330-ft) boundary in the one-dimensional GoldSim™ model, even when the HTF sources are not located directly upgradient from the evaluation wells. The contributions of various HTF sources to any particular well location is based on parameters affecting dispersion, as well as the horizontal and lateral distance between the evaluation well and the HTF source plume centerlines along the 100-m (330-ft) boundary. DOE evaluation wells at the 100-m (330-ft) boundary are illustrated in Figure 4-28.

DOE varies transverse and horizontal dispersivities for western and eastern sources to better match GoldSim™ to PORFLOW™ results (Response to CC-FF-2; SRR-CWDA-2013-00106, Rev. 1). For example, plumes emanating from western sources tend to spread more laterally due to divergent flow. Therefore, DOE applies a higher horizontal transverse dispersivity to western sources. On the other hand, eastern sources tend to spread more vertically compared to western sources (Response to CC-FF-1; SRR-CWDA-2013-00106, Rev. 1). Therefore, DOE applies a higher vertical transverse dispersivity to eastern sources.

DOE also applies a correction to the GoldSim™-calculated well concentrations at the 100-m (330-ft) boundary to account for additional mixing and dilution at the end of the flow paths due to velocity gradients (Response to CC-FF-3; SRR-CWDA-2013-00106, Rev. 1). A factor reduction is applied based on the ratio of the Darcy velocity along the path length to the 100-m (330-ft) boundary and the Darcy velocity at the 100-m (330-ft) boundary (Shaffner, 2013f [ML13199A413]). DOE provides estimates of effective dilution factors for HTF sources in response to CC-FF-4 (SRR-CWDA-2013-00106, Rev. 1) that range from a factor of 24 (Tank 36) to a factor of 118 (Tank 16). The DOE reported dilution factors do not consider longitudinal dispersion that are a function of the source term. For pulse inputs greater than

[31] The plume function is built in to the GoldSim™ code.

Figure 4-28 Well Locations Evaluated in the Probabilistic GoldSim™ Model Based on PORFLOW™ Generated Stream Traces



Adapted from Figure 4.4-54 in SRR-CWDA-2010-00128, Rev. 1.

25 years, the effect of longitudinal dispersion on plume attenuation is insignificant (Response to CC-FF-4; SRR-CWDA-2013-00106, Rev. 1).

DOE also applies a benchmarking factor in the near-field model. A factor of 0.08 is applied to the contaminant flux exiting the annulus to the vault wall. Although DOE attempted to mimic the convective flow cell observed in the PORFLOW™ modeling output, DOE found it was difficult to match the actual mass flux from the annulus to the vault wall. A factor of 8 percent seemed to work well and was ultimately used by DOE as a result of the benchmarking process.

4.2.18.3 Probabilistic Sensitivity Analysis and Results

To identify those most risk-significant parameters and processes affecting the HTF 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 identify both 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 or what is referred to as “endpoints” such as the peak dose to a member of the public 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 the time period has a significant effect on the results of the analysis.

Once a GBM 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 coefficient of determination (R^2) of the linear regression of the model output versus the GBM predictions. The R^2 values for the HTF model are high, indicating an acceptable degree of predictive power of the GBM in fitting the GoldSim™ model. Base case endpoints have R^2 values that range from 0.98 to 0.99, Case D endpoints have R^2 values that range from 0.93 to 0.98, and "All Cases" endpoints have R^2 values of 0.75 to 0.94.

Three GoldSim™ models were evaluated in the probabilistic sensitivity analysis: (1) Case A, (2) Case D, and (3) "All Cases". Each model was run for a 20,000-yr time period and 1,000 model realizations. Results are presented for each of the three models that indicate the most influential parameters affecting (1) peak member of the public dose within 10,000 and 20,000 years, and (2) peak concentrations of I-129, Np-237, Ra-226, and Tc-99. A summary of the HTF PA sensitivity analysis results is provided in Table 4-13.

Sensitivity analysis results show the importance to the peak dose within 20,000 years for the inventory (particularly the inventory of Type III Tank 36 located near the HTF boundary) of the following: technetium solubility, neptunium and iodine cement K_ds, radium K_d in sandy soil, beef transfer factor for technetium, and the water well completion stratum. The water well completion stratum is listed as an important parameter in each probabilistic model owing to its well defined influence on the results. The well-completion stratum determines in which aquifer the evaluation well is located. The GoldSim™ model does not actually simulate contaminant flow and transport to the Gordon Aquifer, but rather reduces the maximum UTRA well concentration by a factor of 100 to represent the maximum Gordon Aquifer well concentration. However, the NRC staff evaluates compliance with the performance objectives in 10 CFR 61.41 and 10 CFR 61.42 at the point of maximum exposure at or beyond a 100-m (330-ft) buffer zone surrounding the disposal facility, irrespective of the probability of the exposure. Therefore, while DOE evaluates the well completion stratum as an uncertain parameter in its probabilistic assessment, compliance with the performance objectives should be evaluated at the point of maximum exposure at or beyond the 100-m (330-ft) buffer zone for any viable aquifer/unit. If DOE were to ignore Gordon Aquifer doses, the result is a peak of the mean dose that is approximately a factor of two lower.

Additional Probabilistic Model Sensitivity Analysis

DOE performed additional sensitivity analysis using the probabilistic GoldSim™ model to study the impact of flow field uncertainty on the results. Because data are not available to justify modification of the PORFLOW™ flow model and GoldSim™ flows are based on PORFLOW™ model results, DOE took what it considered a simple but conservative approach to evaluating flow uncertainty. To study the impact of the groundwater divide on the results, DOE sums the maximum dose concentrations from individual sectors for each time step to represent a case in which source plumes overlap in space rather than diverge. While this scenario is physically unrealistic and exaggerated, DOE believes that the scenario provides a simple method for evaluating the potential cumulative impact of multiple sources converging in space in a bounding analysis. To study the impact of flow velocity uncertainty on the results, DOE performed additional sensitivity analyses that varied Darcy velocity at values of (1) 1.1 m/yr or

Table 4-13 Summary of Sensitivity Analysis Results

Input Parameter	End Point of Interest
Case A	
Tank 12 Inventory Multiplier for Tc-99 Water Well Completion Stratum	Peak Dose Within 10,000 years
Water Well Completion Stratum Technetium Solubility in ORII Cement Tank 36 Inventory Multiplier for Tc-99 Beef Transfer Factor for Technetium	Peak Dose Within 20,000 years
Water Well Completion Stratum Radium K_d in Sandy Soil Technetium Solubility in ORII Cement Tank 36 Inventory Multiplier for Tc-99 Neptunium K_d in ORIII Cement Tank 36 Inventory Multiplier for I-129 Tank 32 Inventory Multiplier for Pu-238 Tank 9 Inventory Multiplier for Np-237 Iodine K_d in ORII Cement	Peak Concentration of I-129, Np-237, Ra-226, and Tc-99
Case D	
Tank 36 Inventory Multiplier for Tc-99	Peak Dose Within 10,000 years
Water Well Completion Stratum Lead K_d in ORIII Cement Technetium Solubility in ORII Cement Tank 36 Inventory Multiplier for Tc-99	Peak Dose Within 20,000 years
All Cases	
Water Well Completion Stratum	Peak Dose Within 10,000 years
Water Well Completion Stratum Technetium Solubility in ORII Cement Tank 36 Inventory Multiplier for Tc-99 Beef Transfer Factor for Technetium	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 in this table. 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 (5 percent) of the endpoint variability.	

3.6 ft/yr (minimum), (2) 2.2 m/yr or 7.4 ft/yr (mean), and (3) 4.3 m/yr or 14 ft/yr (maximum), as well as summing the dose contributions from all sectors.

DOE indicates that summing the sector concentrations alone has limited effect on the peak dose. Because the contributions of five sectors are summed as if only one sector is present, the maximum theoretical dose is a factor of five higher. However, because the peak dose from each sector occurs at different times and some sectors have significantly lower peak doses compared to the maximum sector dose, the dose increase is only a factor of 2 to 3 higher over

10,000- and 20,000-year evaluation periods, respectively (see Table 4-14). In the Darcy velocity sensitivity study, DOE assumes that the Darcy velocity for all sources is the same value at either the minimum, mean, or maximum value. The largest increase in peak dose over the base case is the early peak from Tc-99 from the Type II sand pads under the minimum Darcy velocity case (factor of 7 times higher dose than the base case) at a value of 0.033 mSv/yr (3.3 mrem/yr). The minimum Darcy velocity case also leads to a higher 20,000-year peak dose about a factor of 3.3 times higher than the base case dose and 1.3 times higher than the nominal sum of all sectors case (the case where the Darcy velocity was the same as the base case). The maximum and mean Darcy velocity cases result in peak doses greater than the base case, but less than the nominal sum of all sectors case in the 10,000-year evaluation period. The maximum and mean Darcy velocity cases result in lower peak doses in 10,000 years, because the higher Darcy velocities resulted in a greater degree of dilution. However, the maximum and mean Darcy velocity cases have a peak dose greater than the nominal sum of all sectors case for the 20,000-year period. In the maximum Darcy velocity case, the peak 20,000-year dose is dominated by Ra-226. DOE indicates that a higher velocity leads to more rapid breakthrough of Ra-226 from multiple sources, leading to a higher dose. Table 4-14 provides summary results from the flow field sensitivity analysis. DOE concludes that the analysis shows that a change in water table divide that could change flow directions and velocities would be unlikely to lead to a peak dose that exceeded the performance objectives.

Table 4-14 Peak Dose Results for Probabilistic GoldSim™ Model Sensitivity Cases on Flow Field

Scenario Description	Early Peak Dose (mrem/yr)*	Time of Early Peak (years)	10,000 Year Peak Dose (mrem/yr)*	Time of 10,000 Year Peak (years)	20,000 Year Peak Dose (mrem/yr)*	Time of 20,000 Year Peak (years)
Base case	0.5	840	3.8	9,800	12	17,000
Sum of All Sectors	1.7	840	5.7	9,800	31	20,000
Sum of All Sectors (Minimum)	3.3	940	5.6	9,800	40	13,000
Sum of All Sectors (Mean)	1.9	800	4.4	9,800	37	19,000
Sum of All Sectors (Maximum)	1.0	750	4.0	9,700	37	18,000

*To convert mrem/yr to mSv/yr divide by 100.

4.2.18.4 Deterministic Sensitivity Analysis and Results

DOE performed a series of deterministic sensitivity analyses including evaluation of (1) alternative cases and (2) parameter sensitivity analysis. Alternative cases include the alternative tank configurations B-E summarized in Table 4-11, as well as other cases such as a no closure cap case and a synergistic case. The no closure cap case considers higher infiltration rates 42 cm/yr (16.5 in/yr) from the beginning of the simulation. The synergistic case considers a fast flow path through the tank grout, earlier steel liner failure times, and higher solubility for key radioelements such as plutonium, technetium, uranium and neptunium. However, the synergistic case is only evaluated for 20,000 years with the dose increasing at the end of the simulation period. Additionally, the starting point of the synergistic case is Case C, a case in which there is no communication between the tank fast pathway and the tank vault (see Table 4-11). Finally, basemat and natural system K_ds are not addressed in the synergistic case. Results of the alternative cases are presented in Table 4-15. Results of the parameter sensitivity analysis are provided in Table 4-16.

The only alternative waste tank case with a dose greater than 0.25 mSv/yr (25 mrem/yr) within the period of compliance (10,000 years) is Case E. The highest Case E dose occurs in Sector B, primarily from Np-237. Case E is characterized by rapid contaminated zone chemical transitions expedited by flow through a fast flow path that runs the entire length of the tank system. Besides solubility control, the primary barrier limiting Np-237 release in the base case is attenuation in the basemats. In the absence of solubility control, the tank basemats are the only significant barrier to Np-237 release. With rapid chemical transitions to higher solubility and the fast flow path bypassing the attenuating properties of the tank basemat, the dose from Np-237 is significantly larger in Case E compared to the base case, Case A (see Table 4-15).

Table 4-15 Deterministic Waste Tank Case Results

Case	1,000-Year Evaluation Period		10,000-Year Evaluation Period		100,000-Year Evaluation Period	
	Peak Dose (mrem/yr)*	Timing of Peak Dose (yr)	Peak Dose (mrem/yr)*	Timing of Peak Dose (yr)	Peak Dose (mrem/yr)*	Timing of Peak Dose (yr)
A	0.3	700	4	8,800	124	91,000
B	12	1,000	14	1,100	125	81,000
C	2.1	480	16	7,000	124	84,000
D	12	1,000	18	6,300	123	82,000
E	3.7	840	239	2,300	239	2,300
No Cap	0.7	690	4.7	8,800	9.4	16,000†
Synergistic	2.7	970	5.7	10,000	13	20,000†

* To convert to mSv/yr, multiply by 0.01.
† The No Cap and Synergistic cases are only evaluated for 20,000 years.

Table 4-16 Summary of Deterministic Sensitivity Analysis Results

Parameter	Description	Results
Grout Transition Time	Two times faster and slower transitions. Two cases are simulated: (1) Case A and (2) Case C	Changes in dose spikes that occur earlier in the simulation period are more significant than changes in peak dose over longer periods. Faster grout transitions lead to earlier releases that are higher in some cases (e.g., spike in Tc-99 release in Case C before 10,000 years*) and lower in other cases (e.g., spike in Ra-226 release in Case A around 20,000 years for fast transition and 45,000 years for slow transition†).
Key Radionuclide Solubility	Pu: 1.6×10^{-10} - 1.1×10^{-07} mol/L Tc: 1.3×10^{-12} - 1.1×10^{-08} mol/L Np: 3.0×10^{-13} - 1.0×10^{-04} mol/L U: 1.8×10^{-10} - 3.4×10^{-04} mol/L	Significantly higher peak doses for Tc-99 and Pu-239 for the high end solubility. Tc-99 peak dose is 0.4 mSv/yr (40 mrem/yr) at around 12,000 years‡ compared to 0.04 mSv/yr (4 mrem/yr) in Case A. Pu-239 peak dose is about 1.4 mSv/yr (140 mrem/yr) at around 80,000 years§ compared to insignificant dose in Case A (not reported in PA).
Calcareous Zones, Soil K_d Values	Two times lower effective porosity and bulk soil density in the saturated zone to reflect preferential flow and decreased sorption.	The peak dose in 20,000 years is significantly higher at 0.2 mSv/yr (20 mrem/yr) versus 0.05 mSv/yr (5 mrem/yr). Significantly higher dose from I-129 at 13,000 years and Ra-226 at 20,000 years.
	Two and four times lower K_d in saturated soil.	The peak dose in 20,000 years is significantly higher at 0.6 mSv/yr (60 mrem/yr) at 13,500 years compared to 0.1 mSv/yr (10 mrem/yr) in Case A due to Ra-226.
Waste Tank Liner Failure Times	Four different liner failure times (immediate, early, moderate, and late) for four different tank types, for three different cases fast flow cases (A, B/C, D/E).	Late failures typically lead to lower peak doses. In some cases moderate liner failure times lead to higher peak doses compared to early liner failure times in 20,000 years. No tank with significant annular contamination is simulated.
Cement Degradation	Slow, normal, and fast cement degradation rates.¶	Faster degradation times generally lead to earlier and higher peak doses. No tank with significant annular contamination and no tank with an initially failed liner is simulated.
Water Ingestion Rate	Two times lower and higher water ingestion rate.	Peak dose is almost linearly related to water ingestion rate.

*See Figure 5.6-65 in SRR-CWDA-2010-00128, Rev. 1.
†See Figure 5.6-64 in SRR-CWDA-2010-00128, Rev. 1.
‡See Figure 5.6-71 in SRR-CWDA-2010-00128, Rev. 1.
§See Figure 5.6-71 in SRR-CWDA-2010-00128, Rev. 1.
It is difficult to tell where in relation to other failure curves the immediate failure doses lie (expected to be on top of early failure curve). See Figures 5.6-80 through 5.6-91 in SRR-CWDA-2010-000128, Rev. 1.
¶Tanks 11, 13, 39, and 22 represent Type I, II, III/IIIA, and IV tanks, respectively.

4.2.18.5 Barrier Analysis

As stated in Section 5.6.6 of the HTF PA (SRR-CWDA-2010-00128, Rev. 1), the purpose of the barrier analysis is to study the contributions of various barriers to minimizing the release of radioactivity out of the waste tanks (and vadose zone for non-submerged tanks). Several representative tanks (see Table 4-17) and radionuclides are analyzed representing various half-lives and unique transport behavior (see Table 4-17). The barrier analysis assesses radionuclide flux out of the basemats for submerged tanks and flux out of the vadose zone for non-submerged tanks for the following barriers: (1) vadose zone, (2) closure cap, (3) contaminated zone (chemical), (4) tank liner (hydraulic), and (5) cementitious materials (roof, wall, tank grout, basemat).

Natural attenuation in the saturated zone is outside the scope of the barrier analysis, although the vadose zone is included. The barrier analysis was initially performed and presented in the Revision 0 HTF PA (SRR-CWDA_2010-00128, Rev. 0). DOE indicates the underlying model assumptions and performance affecting properties of the barriers are similar between the Revision 0 and Revision 1 HTF PAs, suggesting the barrier analysis results remain valid. The NRC staff would note that the base case modeling parameters between HTF PA revisions are significantly different in some cases, possibly leading to different conclusions regarding the importance of barriers when comparing nominal versus alternative cases. For example, if the nominal parameters are more degraded (more conservative) in Revision 0 compared to Revision 1, then differences between the nominal (Rev. 0) case and alternative cases representing different levels of degradation may be less pronounced. If, on the other hand, the nominal (Rev. 0) parameters are less degraded (less conservative) in Revision 0 compared to Revision 1, then differences between the nominal case and alternative cases with more degraded barriers may be more pronounced.

Table 4-17 Tanks and Radionuclides Evaluated in the Barrier Analysis

Tank Type	Tank No.	Submerged	Failed Liner	Significant Annulus Inventory
Type I	9	×		×
Type I	12	×	×	
Type II	13	×		
Type II	15	×	×	
Type IV	21			
Type IIIA	36			
Radionuclide	**Half-Life (years)**	**Significant Solubility Control**	**Significant Basemat Sorption**	**Redox Sensitive**
Tc-99	$2.1\times10^{+05}$	×	×	×
Ra-226*	1,600	×	×	
Pu-239	24,000	×	×	×
I-129	$1.6\times10^{+07}$			
Np-237†	$2.1\times10^{+06}$	×	×	

* Produced through ingrowth. Predecessors include Pu-238, U-234, Th-230.
† Produced through ingrowth. Predecessors include Cm-245, Pu-241, Am-241.

Barrier case flux is provided for a 20,000-year evaluation period. The evaluation of flux for a 20,000-year period presents two challenges: (1) the significance of barriers to peak dose is not always clear because flux (not dose) is provided and large differences in peak flux can nonetheless be inconsequential in terms of a dose output, and (2) limiting the evaluation to 20,000 years makes it more difficult to determine barriers to timing versus barriers to the magnitude of peak dose. Sometimes large changes in flux are observed, but it is difficult to tell if these large changes are due to differences in the peak flux or due to differences in the timing of the peak release. If DOE had reported the absolute peak flux (or dose) over time, the NRC staff could have been gleaned information on the impact of the barrier to the magnitude of peak dose from the analysis. Similarly, if DOE had presented flux versus time curves for an evaluation period that captured the peak flux, the NRC staff could discern the impact of barriers on the timing of peak release. Nonetheless, the flux curves presented in this analysis provide valuable risk insights.

Up to three different levels of performance for barriers are assumed: (1) nominal, (2) partially degraded, and (3) fully degraded. Partially degraded conditions are only assumed for steel liners. A description of the levels of performance and the barrier analysis cases is provided in Tables 5.6-42 and 5.6-43 (page 684) of DOE's HTF PA (SRR-CWDA-2010-00128, Rev. 1) and summarized by the NRC staff below in Tables 4-18 and 4-19 for ease of reference.

Finally, it is also important to note that cementitious material K_ds are not varied in the barrier analysis. Only cementitious material chemical transition times appear to be varied, as the barrier analysis focuses on hydraulic, rather than chemical failure of the cementitious materials. Because the FTF PA (SRS-REG-2007-0002, Rev. 1) and supplemental analyses[32] shows the importance of the tank basemats in attenuating releases of key radioelements such as plutonium and neptunium from FTF tanks, the NRC also expects the tank basemats to be one of the most important barriers to waste release at HTF. Although solubility control may be more important for plutonium and neptunium in DOE's HTF PA compared to the FTF PA (i.e., DOE generally assumes lower solubility for key radionuclides in the HTF PA compared to the FTF PA, particularly for Oxidized Region III, based on updated solubility modeling), if solubility control is less than assumed in the HTF PA, the tank basemats could compensate for chemical failure of the contaminated zone. DOE's probabilistic sensitivity analyses also show the importance of cementitious material and natural system sorption. Thus, while basemat sorption and saturated zone attenuation are not studied in DOE's barrier analysis, the NRC staff thinks these two barriers can also be important to HTF performance.

The barrier analysis results show the importance of the (1) steel liner, (2) contaminated zone, and (3) tank grout to waste release. Steel liners primarily affect the timing of the release; however, depending on the tank type and radionuclide, the magnitude can also be affected. The steel liner prevents flow and transport out of the waste tanks, but also allows mass to build up behind the liner. For example, in some cases Ra-226 peak flux increases with prolonged steel liner lifetimes due to longer time for ingrowth from Ra-226 parents, Pu-238, U-234, and

[32] See also the NRC staff's independent assessment of FTF barrier performance, including the basemat, in the FTF Monitoring Plan (Camper, 2013a [ML12345A322]).

Table 4-18 Barrier Analysis Performance Levels

Material Zone	Nominal (N)	Partially Degraded (P)	Fully Degraded (F)
Closure Cap	Base case	N/A	Infiltration constant at 16.5 in/yr
Grout	Base case	N/A	Hydraulic properties of failed grout at time=0 yrs. Chemical properties unchanged. High flow throughout grout imparts reducing capacity onto CZ.
Waste Tank Concrete	Base case	N/A	Hydraulic properties of failed concrete at time=0 yrs. Initial chemical properties unchanged. Chemical transition times are a function of "failed" flow fields.
Contaminated Zone	Base case	N/A	No solubility control Tc-99 and Ra-226. High solubility for remaining radionuclides.
Waste Tank Liner	Base case	Early failure (for tanks that are not already assumed to be initially failed).	No liner at time=0 years.
Vadose Zone (only applies for unsubmerged tanks)	Base case	N/A	Lower K_d values compared to base case, as specified in SRR-CWDA-2012-00080, Rev. 0.

Th-230. In other cases, key radionuclides can be released at higher flux rates with later steel liner failure times due to greater degradation and higher flow in the tank system upon steel liner failure. On the other hand, earlier liner failure can lead to higher flux due to less decay of relatively short-lived radionuclides (radionuclides with a half-life comparable to or less than the steel liner failure time), or can make risk-significant releases within the period of compliance possible. Earlier liner failure also leads to faster chemical transition times, which have also been shown to be important to the peak flux of radionuclides from the tanks.

DOE finds solubility control in the contaminated zone to be important to Tc-99 and Pu-239 peak flux (and Np-237 in later years). The impact of no solubility control of Tc-99 after liner failure is quite pronounced, particularly for tanks with no annulus contamination (otherwise, release of soluble Tc-99 from tank annuli can overshadow release of Tc-99 from the contaminated zone). Pu-239 release is also found to be an order of magnitude higher in Case 6 compared to Case 2

Table 4-19 Barrier Analysis Case Descriptions

Material	Case A	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8	Case 9	No Cap
	Base Case	Failed Case	Steer Liner Cases			CZ Cases		Vadose Zone Cases		Cap
Closure Cap	N	F	F	F	N	F	N	N	F	F
Cementitious	N	F	F	F	N	F	N	N	F	N
CZ	N	N	N	N	N	F	F	N	N	N
Liner	N	F	N	P	P	F	N	N	F	N
Vadose Zone	N	F	F	F	N	F	N	F	N	N

Notes: N – Nominal Performance
P – Partially Degraded Performance
F – Fully Degraded Performance

for a Tank 15 simulation (Case 6 assumes no solubility control, while Case 2 assumes base case solubility control). Solubility control in the contaminated zone is not found to be important for I-129 and Ra-226, presumably because I-129 is not assumed to be solubility-limited and Ra-226 solubility limits are relatively high. DOE states that Np-237 flux is not sensitive to changes in the contaminated zone conditions and is more affected at later times, perhaps due to solubility control of Np-237 parents: Cm-245, Pu-241, and Am-241 (SRR-CWDA-2012-00080, Rev. 0). The NRC notes that the lack of sensitivity of Np-237 release to increases in solubility may be related to basemat performance (i.e., attenuation in the basemat has been shown to be rather significant in FTF analyses). The basemat K_d for neptunium is very high at values of 10,000 L/kg and 5,000 L/kg for reduced and oxidized conditions, respectively. In fact, neptunium flux does not peak in the 20,000-year evaluation period, due in large part to attenuation through the basemat (see Figure 3.3-4 in SRR-CWDA-2012-00080, Rev. 0).

DOE indicates that the tank grout can also delay the timing of release, although the effect on magnitude is less pronounced. Less clear are DOE statements in the HTF PA regarding the impact of hydraulic degradation of the tank grout on contaminated zone solubility, because solubility control in the contaminated zone is listed as a separate barrier[33]. Related to hydraulic degradation of the tank and annular grout, DOE also finds the transition time of the annulus grout important to the timing of Tc-99 release in Type I and II tanks with a significant Tc-99 annular inventory. Faster hydraulic degradation of the grout leads to faster chemical transition times. The K_d of Tc-99 in reducing grout is 5,000 L/kg, compared to 0.8 L/kg in oxidized grout, explaining the potential for rapid release of Tc-99 held up in reduced grout.

[33] DOE notes on page 685 of the HTF PA that Tc-99 release is indirectly affected by degradation of the waste tank grout and that degraded waste tank grout has the ability to impart its reducing capacity onto the CZ (SRR-CWDA-2010-00128, Rev. 1).

DOE's HTF PA indicates that variable grout hydraulic conductivity can have a marked influence on the flow direction through the tank system, leading to the redirection of flow through the annulus (Page 685; SRR-CWDA-2010-00128, Rev. 1). A supporting reference to the HTF PA indicates that in the absence of higher flow rates in Case C, upward diffusion of Tc-99, I-129, and Ra-226 from the contaminated zone becomes an important process leading to the storage of these radionuclides in the reducing tank grout (SRR-CWDA-2012-00080, Rev. 0). The reference goes on to state that higher release rates in Case C, after liner failure, are due to transport from the tank grout through the annulus and vault wall. However, given the relatively high flux of Tc-99 after liner failure for Tanks 13 and 15 in Case C that are well above the Tc-99 solubility limit, it appears the initial release following liner failure is sourced from the annulus inventory, rather than from the contaminated zone inventory (see Figures 3.2-1 and 3.2-2 in SRR-CWDA-2012-00080, Rev. 0). While release of the inventory associated with the CZ is controlled by solubility limits, inventory associated with the annulus is not subjected to solubility controls. Instead, the migration of annular inventory is controlled by K_ds.

DOE indicates that the closure cap is important to limiting flow through the tanks for the first few thousand years (SRR-CWDA-2010-000128, Rev. 1). For example, SRR-CWDA-2012-00080, Rev. 0 shows an earlier Tc-99 peak flux that is almost an order of magnitude higher from Tank 15, a tank with an initially failed liner, in the no cap case compared to Case A (see Figure 3.1-2 page 18). The report SRR-CWDA-2012-00080, Rev. 0 indicates that the closure cap is a moderately effective barrier for mobile radionuclides not greatly influenced by sorption onto oxidized cementitious materials (Tc-99, Ra-226, and I-129). The impact of the closure cap on flux is stated to be greater for slow-moving radionuclides, such as Pu-239 and Np-237, that sorb to oxidized cementitious materials. However, the NRC staff would note that the two order of magnitude difference in the Np-237 flux in the no-cap case versus Case A results is at least partially attributable to the length of the evaluation period. The Np-237 flux is still increasing at the end of the 20,000-year evaluation period with the flux curve for nominal conditions lagging behind the no cap case. Therefore, the absolute peak flux for Np-237 cannot be compared for these two cases. Nonetheless, some key insights are clear—the higher long-term infiltration rate of 42 cm/yr (16.5 in/yr) assumed in the no-cap case (1) increases flux out of the tanks, and (2) leads to faster chemical transition times compared to Case A, with Case A using a lower maximum infiltration rate of 30 cm/yr (12 in/yr) after the cap is assumed to be fully degraded.

Finally, DOE indicates that the vadose zone dampens radionuclide releases especially for those radionuclides with a high soil K_d value (e.g., isotopes of plutonium and radium). However, DOE indicates that this vadose zone barrier is less effective compared to other barriers. The NRC staff notes that the magnitude of radium peak flux appears to be more impacted by the vadose zone K_ds compared to plutonium. The vadose zone K_d affects primarily the timing of plutonium peak flux, rather than the magnitude.

4.2.18.6 Probabilistic Uncertainty Analysis and Results

DOE performed a probabilistic uncertainty analysis using the GoldSim™ modeling platform, as described earlier. HTF PA (SRR-CWDA-2010-00128, Rev. 1) Sections 4.4.4.2 and 5.6 provide additional details regarding DOE's probabilistic assessment. Figures 5.6-31 through 5.6-34 in the HTF PA provide statistical dose versus time curves for the member of the public for different Cases (A, D, and All Cases) and evaluation periods (10,000 years and 100,000 years). DOE

notes that the median dose is significantly lower than the peak of the mean dose. DOE indicates that its selection of conservative parameter distributions biases the results high with parameter values sampled in the tails of the distributions, leading to high dose realizations that strongly influence the peak of the mean dose.

Tables 5.6-32 and 5.6-33 in DOE's HTF PA (SRR-CWDA-2010-00128, Rev. 1) provide information on the peak-of-the-mean dose and mean-of-the-peak doses, as summarized in Table 4-20 below. A comparison of the peak-of-the-mean dose and the mean of the peak doses is instructive as it provides information about the potential for risk dilution in the probabilistic model. For example, if generic data or professional judgment are used to determine parameter distributions in a probabilistic analysis and the data are (1) unbiased compared to the "true" unknown data and (2) the parameter being observed is important only to the magnitude (not the timing) of the output, then the impact is mainly on the perceived variance of the output, but not necessarily on the mean of the output. On the other hand, the use of generic distributions that have a strong influence on the timing of the output could lead to risk dilution, a phenomenon that occurs when the evaluation of uncertainty reduces the magnitude of projected impacts, as discussed in technical literature (see Mohanty and Codell, 2004; OECD, 2005). For example, consider a case where the magnitude of radionuclide release and dose are fairly certain, but the timing of release and peak dose are uncertain. The use of overly broad distributions for parameters (i.e., high variance) affecting only the timing of release and peak dose would lead to a situation where the peak of the mean dose curve is significantly lower than the average peak dose. This appears to be true of DOE's probabilistic assessment, based on the results presented in Table 4-20. The mean of the peak doses is approximately an order of magnitude greater than the peak of the mean dose within a 10,000-year evaluation period. Because the peak of the mean dose is much more comparable to the mean of the peak doses during a 100,000-year evaluation period, it appears the uncertainty in the peak dose is related primarily to the timing and not necessarily the magnitude of the peak dose.

One of the most important parameters influencing the timing of peak dose is the time to steel liner failure. Significant differences exist between the timing of steel liner failure for two classes of tanks: (1) tanks with initially intact liners, and (2) tanks with initially failed liners. This leads to some dispersion of HTF tank release impacts in the deterministic simulation, although tanks of the same type (i.e., I (intact liner), I (no liner), II (intact liner), II (no liner), III/IIIA, and IV) are assumed to fail at the same time. Still, the cumulative impact of the failure of tanks of the same type at the same time is offset by the divergence of flow paths at HTF (i.e., centerlines of source plumes do not tend to overlap, see Figure 4-28 that shows the divergence of flow from HTF tanks). Not shown in Figure 4-28 is the difference in vertical profile with some source plumes located deeper in the aquifer and some plumes located vertically higher in the aquifer. Steel liner failure times generally occur earlier in alternative configurations within the compliance period, whereas only Type IV (and one Type I and three Type II tanks that are assumed to be initially failed) fail within the compliance period in Case A—all other tanks of Type I, II, III/IIA fail after the 10,000-year compliance period in the base case.

In the probabilistic analysis, steel liner failure times for alternative cases span a large range (see Table 4-21). On the other hand, steel liner failure times occur in a narrow range for Case A, with the exception of Type IV tanks. In many cases (e.g., when no solubility control is assumed for technetium under oxidized conditions), Tc-99 dominates the peak dose. Because Tc-99 is mobile in cementitious and natural materials under oxidizing conditions, the Tc-99 peak is

Table 4-20 Peak of the Mean Dose and Mean of the Peak Doses for Member-of-the-Public Evaluation in Probabilistic Analysis

Analysis	Peak of the Mean Dose (Median/95th Percentile)		Mean of the Peak Doses (Median/95th Percentile)
	mrem/yr*	yr	mrem/yr*
10,000-Year Evaluation Period			
Case A	13† (2/24)	8,240	85 (9/520)
Case D	35‡ (12/112)	9,620	210 (31/980)
All Cases	15§ (6/58)	9,750	220 (28/1000)
100,000-Year Evaluation Period			
All Cases	205□ (12/112)	67,300	530 (310/1900)

* To convert to mSv/yr multiply by 0.01
† HTF Transport Model v0.025 CaseA r1000 s2t.gsm
‡ HTF Transport Model v0.025 CaseD r1000 s2t.gsm
§ HTF Transport Model v0.025 CaseAll r1000 s04t. gsm
□ HTF Transport Model v0.025 100ky CaseAll r1000 s1.gsm

typically sharp when no solubility control is assumed. After liner failure, Tc-99 releases are further spread out in time due to assumptions regarding chemical transition times. Under these circumstances, the sharp Tc-99 peak doses do not tend to overlap from realization to realization, thereby enhancing the potential for risk dilution.

Another key parameter affecting the timing of peak dose is the distribution coefficient. Distribution coefficients are applied to cementitious and natural system materials. DOE assumes a lognormal distribution with geometric standard deviations and maximum and minimum values varying by material. For example, DOE assumes a cementitious material geometric standard deviation of 0.25 times the geometric mean for geometric means above 4.0, and a value of 1.0001, otherwise. DOE assumes minimum and maximum values of 0.5 and 1.5 the geometric mean for cementitious materials, respectively. For sandy soils (i.e., vadose zone and UTRA and Gordon aquifers), DOE assumes a geometric standard deviation of 0.375 times the geometric mean for geometric means above 2.7 L/kg, and a value of 1.0001, otherwise. DOE assumes minimum and maximum values of 0.25 and 1.75 the geometric mean for sandy soils, respectively. These distributions are well constrained and are thus, not expected to contribute unduly to risk dilution.[34] However, at later times, Ra-226 dominates the peak dose in many cases, and the dose versus time curves for Ra-226, while broad, are also spread out in

[34]See Section 4.2.9 for additional evaluation of DOE's treatment of distribution coefficient uncertainty.

Table 4-21 Summary of Steel Liner Failure Times Assumed in the Deterministic and Probabilistic Analyses

Parametric Flow Cases								
	No Fast Flow (Case A)				Partial and Full Fast Flow Path Cases (Cases B-E)			
	Type I	Type II	Type III/IIIA	Type IV	Type I	Type II	Type III/IIIA	Type IV
Initially Failed	0	0	0	0	0	0	0	0
Early	2,100	2,500	3,100	500	100	100	100	75
Moderate	11,000	13,000	13,000	3,600	1,100	2,500	2,100	1,000
Late	15,000	15,000	15,000	8,000	11,000	12,000	12,000	3,600
PORFLOW™ Deterministic Model (Tanks With Initially Intact Liners Only)								
Median	11,000	13,000	13,000	3,600	1,100	2,500	2,100	75
GoldSim™ Sampled Steel Liner Failure Times (Tanks With Initially In-Tact Liners Only)								
Mean	11,000	13,000	13,000	4,800	2,400	4,300	4,000	200
2.5%-ile	8,900	12,000	12,000	650	240	470	400	45
97.5%-ile	13,000	13,000	13,000	10,000	12,000	13,000	13,000	1,100

time due to a series of assumptions regarding steel liner failure times, transport through cementitious materials, and transport through the natural system.

Another key parameter affecting the timing of peak dose is chemical transition times, which are tightly constrained to ± 30 percent for the first transition and ± 50 percent for the second transition. From a risk dilution perspective, the narrow parameter distribution for chemical transition is beneficial (i.e., risk dilution would be more pronounced with a larger range of chemical transition times). However, there is limited support for the longevity of chemical conditions assumed in DOE's base case and earlier transitions could lead to significantly higher peak doses for certain key radionuclides within the compliance period. Table 4-22 provides information on the chemical transition times assumed in the deterministic and probabilistic analyses. For most tanks in the deterministic analysis, the chemical transitions to higher solubility occur after the period of compliance. One exception is Type I tanks with no liner that experience more rapid chemical transitions due to saturated groundwater cross-flow that helps deplete the reducing and buffering capacity of the tank grout more quickly. The first chemical

transition for Type II tanks with no liner and Type IV tanks also occurs at the end of the 10,000-year compliance period, although the second chemical transition occurs beyond the compliance period.

Realizations of Interest

DOE evaluates the top five highest realizations from the Case A, Case D, and All Cases probabilistic models within a 10,000-year evaluation period. Tc-99 dominates the peak dose in the Case A model with the following primary parameters affecting magnitude of peak dose: (1) Tc-99 inventory multiplier, (2) technetium beef transfer factor, and (3) technetium solubility for Oxidized Region II or III
DOE indicates that the highest doses occur in Sector A from a tank with no liner, Tank 12. The peak dose ranges from 17 to 35 mSv/yr (1,700 to 3,500 mrem/yr) compared to the base case value of 0.04 mSv/yr (4 mrem/yr). An important parameter affecting the timing of the peak dose is the chemical transition times to no solubility control in Oxidized Region III that allow the peak dose to occur within 10,000 years. It is also important to note that in Case A, drinking water ingestion is the primary pathway, while the primary pathway for all top five Case A realizations is beef ingestion, emphasizing the importance of the selection of the beef transfer factor to the results of the analysis.

Table 4-22 Summary of Chemical Transition Times

Waste Tank Position	Transition	Number of Pore Volumes Required (CZ Transition Time [Yr])		
		Deterministic	Minimum	Maximum
Partially and Non-Submerged	Reduced Region II to Oxidized Region II	523 (7,900 Type IV 9,200 Type II NL* 15,000 Type II 16,000 Type III)	366	680
	Oxidized Region II to Oxidized Region III	2,119 (19,000 Type II NL* 22,000 Type IV 25,000 Type II 28,000 Type III)	1,060	3,179
Submerged (Type I tanks)	Reduced Region II to Oxidized Region II (Condition C to D)	1,787 (7,700 Type I NL* 12,000 Type I)	1,251	2,323
	Oxidized Region II to Oxidized Region III (Condition D to Oxidized Region III)	2,442 (8,100 Type I NL* 12,000 Type I)	1,221	3,663
* Notes: NL=no liner				

With respect to the highest Case D realizations, the peak dose ranges from 43 to 150 mSv/yr (4,300 to 15,000 mrem/yr) compared to the deterministic Case D value of 0.18 mSv/yr (18 mrem/yr) considering a 10,000-year evaluation period. The key contributors to peak dose in these realizations are either Sr-90 or Tc-99, compared to Np-237 and Ra-226 in the deterministic analysis. DOE indicates that the high dose realizations with Sr-90 as the key contributor are sourced from Tank 15, a tank with an initially failed liner. The realizations with Tc-99 as the key contributor are sourced from Tank 36, which is located near the boundary. The Sr-90 high dose realizations of 150 and 43 mSv/yr (15,000 and 4,300 mrem/yr) occur in the first 200 years before Sr-90 decays to non-risk-significant levels. The high Sr-90 doses are facilitated by (1) an initially failed liner in Tank 15, (2) no cap to initially reduce infiltration, (3) a high inventory, (4) a lower K_d in sandy soils, and/or (5) a high transfer factor for fish ingestion. The high Tc-99 dose within 10,000 years is facilitated by (1) no solubility control in Oxidized Region II, (2) the beef transfer factor, (3) the Tc-99 inventory multiplier, and (4) the transition time to Oxidized Region II occurring within 10,000 years.

Finally, the highest doses reported for the "All Cases" model are dominated by Tc-99 with peak doses between 25 and 30 mSv/yr (2,500 to 3,000 mrem/yr) occurring within about 6,000 years. These results are similar to Case A results, but occur slightly earlier due to faster steel liner failure and chemical transitions associated with alternative configurations. However, with only 1,000 realizations evaluated, the highest risk realizations reported for Case D are not reproduced. Neither are Case E results, which have been shown to be significantly higher than other alternative cases, well represented (e.g., realizations with doses dominated by Np-237), because only a total of approximately 38 Case E realizations are evaluated. Nonetheless, the "All Cases" model demonstrates that high peak Tc-99 doses can occur from Type III and Type IV tanks with early liner failure and chemical transition times to no solubility control. No additional insights beyond those reported for the Case A results are noted by DOE.

4.2.19 NRC Evaluation of Uncertainty and Sensitivity Analyses and Summary of Major Findings

With respect to the level of analysis performed by DOE to study the importance of key barriers and parameters in DOE's HTF PA, the 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 levels of engineered barrier and vadose zone performance on flux out of the near-field model. The information provided by DOE in its HTF PA greatly assists the NRC staff with gaining a better understanding of overall system performance.

With respect to the implementation of DOE's probabilistic model, the NRC staff 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. DOE's compliance demonstration is heavily weighted towards the base case configuration (Case A). The NRC staff is concerned with the level of support for key modeling assumptions in DOE's base case. Much of the discussion in the preceding sections of Chapter 4 of this TER provides additional details regarding the NRC staff's specific concerns. As shown in Table 4-3, an exceedance of the performance objective is projected to occur at some point in the future from Ra-226, although it seems likely that the peak dose from Ra-226 will occur after the 10,000-year compliance period due to ingrowth from

Ra-226 parents: Pu-238, U-234, and Th-230. On the other hand, the uncertainty and sensitivity analyses show the potential for other radionuclides to dominate the peak dose earlier in time and at levels potentially above the performance objectives, should assumptions and approaches relied on for Case A prove to be invalid.

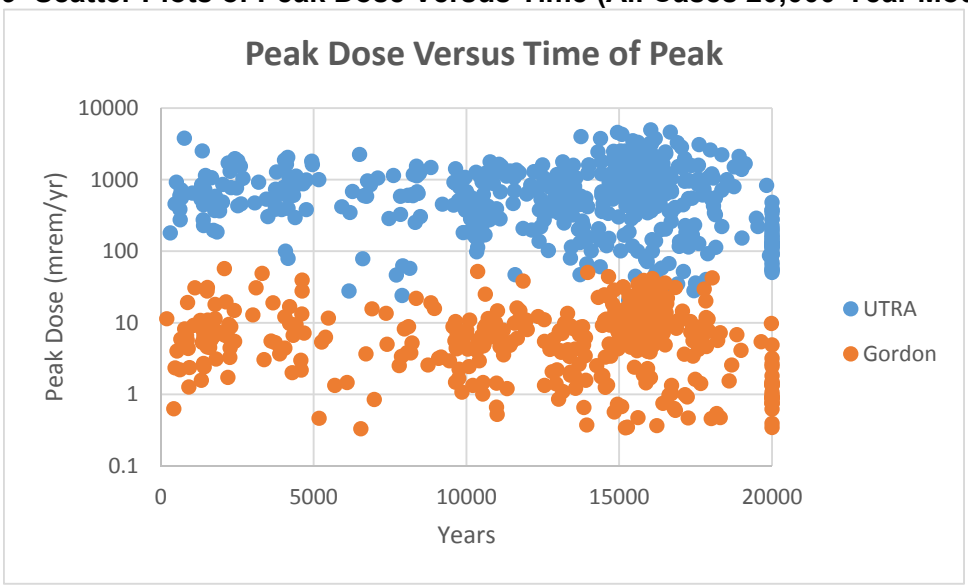
The NRC staff extracted output from DOE's probabilistic GoldSim™ model to produce scatter plots of peak dose versus time in Figures 4-29 and 4-30 for the "All Cases" probabilistic model run for a 20,000- and 100,000-year simulation period, respectively. Figure 4-29(a) shows two sets of realizations that are similar in shape but that differ by the magnitude of the peak dose with the bottom set representing results for the Gordon Aquifer that is nominally set at a value of 100 times less than the dose computed for the UTRA. A comparison of the peak dose versus time for a 20,000-year simulation period reveals that the average peak dose is 8.9 mSv/yr (890 mrem/yr) when the Gordon Aquifer wells are removed compared to an average peak dose of 5 mSv/yr (500 mrem/yr) for all realizations or approximately 44 percent less, as expected. A comparison of the peak dose versus time for a 100,000-year simulation period reveals that the average peak dose is 9.8 mSv/yr (980 mrem/yr) when the Gordon Aquifer wells are removed compared to an average peak dose of 5.3 mSv/yr (530 mrem/yr) for all realizations. The average time of peak dose is around 13,000 years for the 20,000-year simulation period and around 30,000 years for the 100,000-year simulation period. While all peak doses are greater than around 0.4 mSv/yr (40 mrem/yr) in the 20,000 year-simulation, if the full magnitude of the peak is allowed to occur with a longer simulation period, the lowest peak dose is higher at 1.6 mSv/yr (160 mrem/yr) for the 100,000-year simulation period. Figures 4-29(b) and 4-30(b) show histograms of the time of peak dose for the 20,000- and 100,000- year simulation periods, respectively. Remarkably, every realization with the UTRA selected as the well completion stratum is above the dose limit of 0.25 mSv/yr (25 mrem/yr) for the 10 CFR 61.41 performance objective. The high dose results are accentuated by assumptions regarding loss of technetium solubility control and the discrete failure of the steel liners that lead to exaggerated spikes in the Tc-99 dose. Nonetheless, DOE's probabilistic results show that both the timing and magnitude of peak dose within the compliance period appear to be fairly uncertain.

4.2.19.1 NRC Evaluation of Uncertainty Related to Timing and Magnitude of Peak Dose

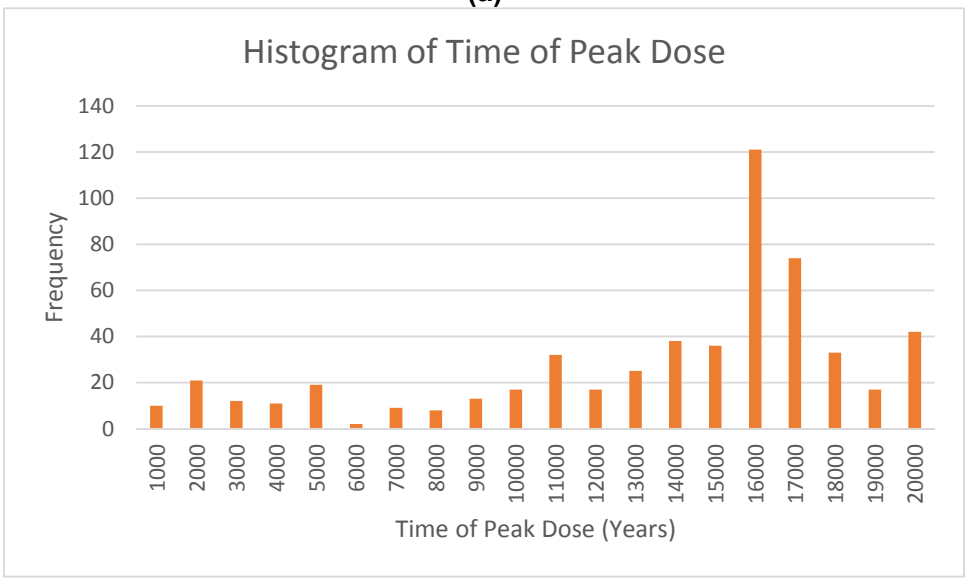
Factors Important to Timing of Peak Dose

Similar to concerns raised in the NRC staff's FTF TER and Monitoring Plan (Camper, 2011 [ML112371751]; Camper, 2013a [ML12345A322]) and in HTF RAIs (Mohseni, 2013a [ML13196A135]), the NRC staff's primary concern with DOE's HTF PA rests with the lack of support for key modeling assumptions. With regard to barriers that delay the timing of peak dose, the NRC staff continues to have concerns with assumptions related to (1) steel liner and (2) chemical barrier performance, both of which serve to delay the timing of peak dose. The probabilistic analysis results are also heavily weighted by the base case, Case A, although less so in the HTF PA compared to the FTF PA. While faster steel liner failure times can be realized for tanks with initially failed liners and in alternative cases (Cases B through E), Cases B through E account for only 25 percent of the realizations in DOE's HTF probabilistic assessment. Thus, steel liner failure times are dominated by Case A that has relatively longer steel liner failure times for initially intact tanks (e.g., only a fraction of a percent of Type I, II, and III/IIIA tanks with initially intact liners fail within 10,000 years in Case A).

Figure 4-29 Scatter Plots of Peak Dose Versus Time (All Cases 20,000-Year Model*)



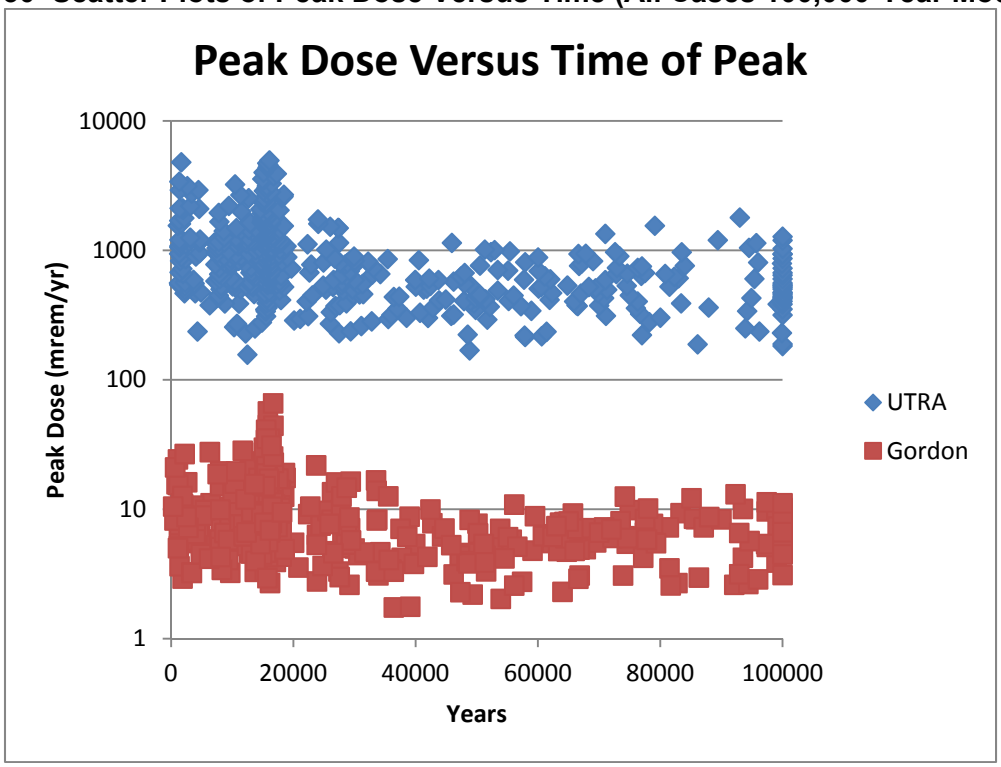
(a)



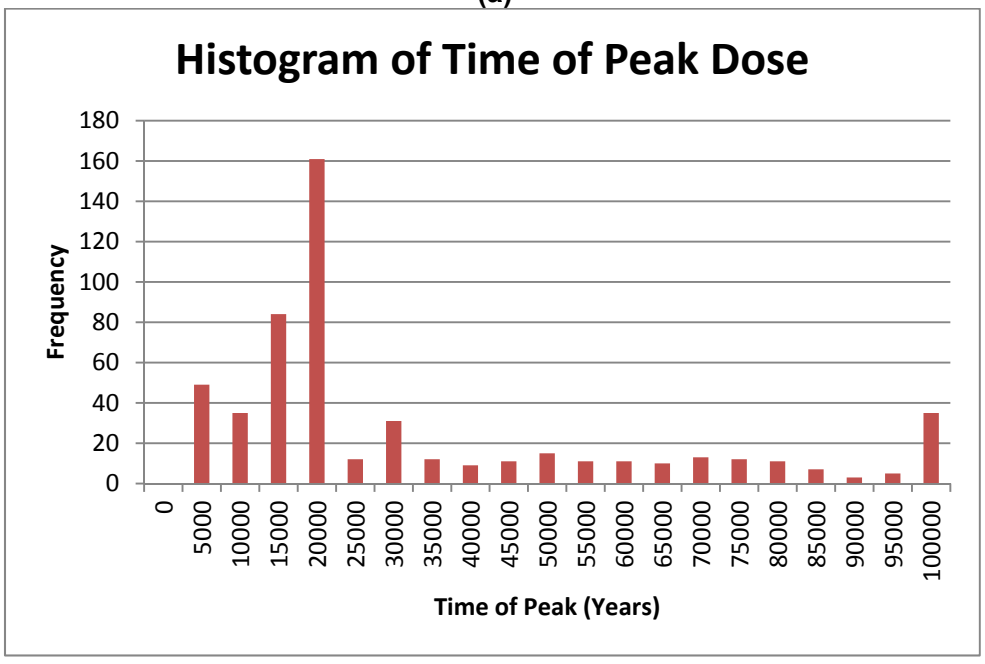
(b)

*The HTF Transport Model v0.025 CaseAll r1000 s01.gsm model file was evaluated

Figure 4-30 Scatter Plots of Peak Dose Versus Time (All Cases 100,000-Year Model)



(a)



(b)

Additionally, the NRC staff thinks that chemical transition times are not fully supported in the base case, Case A, and that the level of uncertainty in the performance of chemical barriers is understated[35]. The degradation mode for tank grout in most of the tanks at HTF is cracking due to corrosion product expansion. Further, shrinkage away from tank walls and cooling coils is expected. Yet, DOE does not consider preferential or bypassing flow pathways through cracks and shrinkage gaps in its base case when calculating chemical transition times. Furthermore, the chemical transition times for submerged tanks appears to require stronger technical bases considering they were developed with limited data that may not be representative of actual conditions (e.g., low dissolved oxygen concentration from background well P27D[36]).

The NRC staff thinks that the uncertainty in chemical transitions is likely understated in the base case, Case A. For example, DOE uses factors of ±30 and ±50 percent to represent uncertainty in the E_h and pH chemical transition times, respectively. These uncertainty ranges do not account for the presence of bypassing pathways through the tank grout in Case A, nor do they account for uncertainty in geochemical modeling used to calculate the chemical transition times (i.e., normative grout mineralogy controlling E_h and pH). Finally, the range of chemical transition times do not account for uncertainty in the ability of reducing tank grout to chemically condition infiltrating groundwater {e.g., independent research has shown that the E_h is not necessarily reflective of the reducing capacity of the tank grout (Pabalan et al., 2012 [ML12089A319])}. Interaction of groundwater with reducing tank grout may be physically or kinetically limited as discussed in more detail under Monitoring Factor 3.2 in the NRC staff's FTF Monitoring Plan (Camper, 2013a [ML12345A322]).

The majority of peak doses in the probabilistic analysis occur beyond the period of compliance. For example, the NRC staff's investigation of the 100,000 year "All Cases" probabilistic modeling results reveals that roughly 85 percent of the realizations have peak doses that occur beyond the 10,000-year compliance period. This result is due in part to: (1) potentially overly optimistic steel liner failure times, (2) prolonged chemical transition times, and (3) parameter distributions that may not sufficiently account for uncertainties that may hasten the time to steel liner and chemical barrier failure. It is also important to note that Ra-226 doses typically increase over time due to ingrowth from Pu-238, U-234, and Th-230, leading to peak doses that occur later in time in many cases.

With respect to annular waste, the NRC staff thinks that DOE has not adequately evaluated annular inventory risk in its HTF PA and supplemental RAI responses. DOE assumes minimal to no contact of the annular inventory with preferential pathways in deterministic and probabilistic models. Preferential pathways through the Tank 16 vault were implicated in the

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[35] While the NRC staff acknowledges that significantly faster chemical transition times occur for alternative Cases C and E, the probability of these cases is low and thus, the impact of faster chemical transitions is dampened.

[36] Well P27D dissolved oxygen concentration is a factor of 5 times less than other water table wells reported in Strom and Kaback (1992) for SRS. Well P27D does not appear to be representative of saturated groundwater near the water table surface and Type I tanks, as described in more detail in Section 4.2.9.

release of tank waste into the environment. Furthermore, annular waste in fully and partially submerged tanks may be released significantly earlier than what was assumed in DOE's HTF PA models, due to groundwater in-leakage and/or drainage, as discussed in more detail in Section 4.2.8. Delayed release of annular inventories is particularly important to reducing the risk from short-lived radionuclides such as Sr-90 and Cs-137.

The reducing grout appears to be a significant hydraulic barrier in DOE's analysis (see DOE's barrier analysis results). The timing of tank and annular grout hydraulic degradation affects chemical transition times. DOE assumes that tank and annular grout begin to degrade thousands of years after tank closure and degradation is not complete until tens of thousands of years into the future. Table 4-23 summarizes DOE's cementitious material degradation time assumptions in the HTF PA. Tank and grout hydraulic degradation times are especially important for tanks with initially failed liners, as the tank grout is the primary barrier limiting flow into the system. See additional discussion regarding assumptions related to steel liner failure times, chemical transition times, and hydraulic failure of cementitious materials in Sections 4.2.8 and 4.2.9.

Table 4-23 Summary of Cementitious Material Degradation Times (Years)

Cement Stage	Tank Type				
	Type I	**Type II**	**Type III**	**Type IIIA**	**Type IV**
Degrading HTF Reducing Grout	2,700 to 13,200	5,100 to 16,700	5,100 to 19,200	5,000 to 19,100	800 to 64,400
Degrading HTF Aged Concrete	1,350 to 2,700	2,550 to 5,100	2,550 to 5,100	2,500 to 5,000	400 to 800

Cementitious material K_ds also influence the timing of peak dose, but to a lesser degree than other parameters. Nonetheless, the NRC staff will continue to evaluate the impact of bypass flow through tank vaults that is particularly risk significant for plutonium and neptunium release. For example, the neptunium K_d of 10,000 L/kg is particularly risk significant and will be the focus of the NRC staff's monitoring in this area.

Finally, saturated zone travel times to the compliance boundary also affect the timing of peak dose. Travel times are a function of the Darcy velocities abstracted from the PORFLOW™ models, as well as natural system K_ds. Given the issues associated with the HTF groundwater model discussed in Section 4.2.11, the NRC staff will continue to monitor DOE's support for the HTF hydrogeological conceptual model and the Darcy velocities assumed in the HTF PA. With regard to natural system performance, the NRC staff continues to have concerns with the K_d

averaging approach used to simulate plutonium transport in the subsurface[37]. Furthermore, the radium K_d may also be important to the compliance demonstration and will continue to be evaluated in monitoring.

Factors Important to Magnitude of Peak Dose

The most important parameters affecting the magnitude of the peak dose are the solubility limits assigned to the three chemical states assumed in DOE's HTF PA: (1) Reduced Region II (or Condition C for submerged tanks), (2) Oxidized Region II (or Condition D for submerged tanks), and (3) Oxidized Region III. In the HTF PA, DOE limits the solubility of several key radionuclides based on assumptions regarding the solubility-limiting phase (e.g., iron co-precipitation) or the chemistry of the infiltrating groundwater in the various chemical states (e.g., the assumed E_h of infiltrating groundwater results in relatively low solubility limits for plutonium for all time in the base case). Furthermore, the NRC staff is concerned that DOE has not appropriately considered uncertainty in the geochemical modeling performed to determine solubility limits. As discussed in Section 4.2.9, the NRC staff concludes that DOE's solubility assumptions require additional support for the NRC staff to have reasonable assurance that the performance objectives can be met. Similar to concerns raised in the FTF TER (Camper, 2011 [ML112371751]) and Monitoring Plan (see Monitoring Factors 4.1 and 4.2, Camper, 2013a [ML12345A322]), the NRC staff continues to recommend that DOE provide additional support related to solubility-limiting phases and limits (see Section 4.2.9).

The NRC staff will also monitor DOE's efforts to obtain additional support for cementitious material and natural system K_ds for key radionuclides that are important to the compliance demonstration (see Monitoring Factors 3.5 and 4.1 in Camper, 2013a [ML12345A322]). Improved model calibration in the area of interest will also increase the technical defensibility of the HTF dose projections. Finally, the final tank inventories are also important to dose and will be a focus of monitoring at HTF (see Monitoring Factors 1.1, 1.2, and 1.3 in Camper, 2013a [ML12345A322]).

4.2.19.2 NRC Evaluation of the Hybrid Modeling Approach

With regard to the hybrid modeling approach, the NRC staff thinks that the information provided in the sensitivity and uncertainty analysis greatly improves understanding of overall system performance and is helpful with respect to informing the compliance decision. However, the uncertainty analysis should not be used to demonstrate compliance with the performance objectives because (1) there is limited support for the base case, and (2) there is limited support for the assignment of the likelihood of alternative cases and consequently, the averaging of alternative cases in the "All Cases" model. Therefore, the NRC staff recommends that DOE continue to present the results of alternative cases individually and provide qualitative information regarding the likelihood of alternative cases. With regard to the deterministic analysis, the NRC staff thinks that additional information is needed to support the compliance

[37] See the FTF Monitoring Plan (Camper, 2013a [ML12345A322), Monitoring Factor 4.1 and Appendix E, for additional information.

case, Case A. Ideally, supporting information would be in the form of additional experimental or field data, natural analogs, peer review, expert elicitation, and other forms of model support. Without this additional model support, it is difficult to argue the relative likelihood of the base case compared to alternative cases.

The NRC staff is also concerned that potentially risk-significant features of the disposal facility or important degradation processes that may compromise the effectiveness of HTF barriers are not included in the base case, Case A. While the effect of risk-significant FEPs are sometimes implicitly considered in probabilistic analysis, it is not clear that an adequate evaluation of high-risk cases has been conducted. Additionally, DOE typically performs deterministic sensitivity analyses in which only one parameter is varied at a time, or in which one or more barriers remains intact, making it difficult to evaluate the most important vulnerabilities in system performance. The following examples are provided for illustrative purposes:

- The deterministic, synergistic case represents a degraded engineered barrier case. However, the synergistic case represents a case with no bypassing pathway through the tank vault; and hydraulic degradation rates of cementitious materials and basemat K_ds are similar to the base case.

- DOE evaluates a "fully degraded" case in the deterministic barrier analysis, Case 7. However, the barrier analysis does not consider a key barrier to waste release (i.e., basemat sorption). For example, had neptunium basemat sorption been evaluated, DOE may have concluded that the basemat was very important to performance when comparing a case with both the basemat and solubility control versus just solubility control "turned off".

- Although the reducing grout is assumed to be failed at 500 years in alternative, deterministic Cases B and D, chemical transition times are controlled by the overlying grout, similar to the base case, leading to longer times to chemical transitions compared to Cases C and E. Faster chemical transitions in the contaminated zone occur in alternative, deterministic Cases C and E, although flow through the reducing tank and annular grout is limited for thousands of years, similar to the base case. Fast chemical transitions in the contaminated zone but prolonged chemical transitions in the annuli of Type I and II tanks in Cases C and E, may be important to reducing the cumulative impact of annular and tank releases for tanks with significant annular contamination.

- Although in the deterministic analysis, Case E has rapid chemical transitions and bypassing pathways through both the tank and vault, solubility is constrained to low values for most key radionuclides, dampening the potential peak dose in Case E with the notable exception of -237.

- Cumulative impacts of underperformance of both natural and engineered barriers are typically not considered simultaneously (e.g., only vadose zone K_ds and engineered barriers are considered in deterministic barrier analysis).

4.2.19.3 NRC Evaluation of Probabilistic Model Construction, Abstractions, and Benchmarking

The NRC staff notes several concerns associated with the abstraction of the three-dimensional PORFLOW™ model in the probabilistic one-dimensional GoldSim™ model in the following paragraphs. The NRC staff also notes potential deficiencies in both the PORFLOW™ and GoldSim™ models, as well as limitations in the benchmarking process in this section.

The NRC staff is concerned that assumptions regarding steel liner failure and cementitious material degradation in the parametric flow study affect flow through the fast-flow pathway in alternative configurations, and consequently, chemical transition times. For example, Cases C and E are inherently preferential flow cases, in which groundwater is assumed to bypass the reducing and buffering capacity of the tank grout, thereby expediting contaminated zone chemical transitions. However, Cases B and C, and Cases D and E are grouped together in the parametric flow study, making it difficult to determine if rapid chemical transitions actually occur in Cases C and E.

DOE applies a basemat bypass fraction to all cases irrespective of whether a fast-flow pathway exists through the basemat, which further adds ambiguity to the implementation of alternative cases simulated in the probabilistic analysis,. The basemat bypass fraction is sampled from a triangular distribution with an expected value of zero percent and a maximum value of ten percent. DOE indicated in a July 3rd, 2013, clarification teleconference (Shaffner, 2013f [ML13199A413]), that there is no correlation between bypass flow and bypass fraction.

Thus, while the parametric flow study was intended to provide information on the impact of flow field variability on the results, without a well-defined conceptual model and appropriate parameter correlations, obtaining clear signals from the probabilistic sensitivity analysis is difficult. A more effective method of evaluating the sensitivity of the results to alternative cases or the combination of parameters most influencing contaminant flow and transport may be needed to assist with interpretation of the results.

The NRC staff also has concerns with the lack of DOE's consideration of a bypassing pathway through tank annuli in deterministic and probabilistic analyses. No direct communication exists between the annulus waste and the fast-flow pathway that runs vertically through the center of the tank and basemat in Cases D and E for Type I tanks. Although annular inventories loaded in Type II tank sand pads intersect the fast-flow pathway that runs through the center of the tank and basemat, only a small fraction of the Type II sand pad inventory appears to be able to migrate through the sand pads and down and out the basemat through the fast-flow pathway (see Sections 4.2.9 and 4.2.10).

Groundwater in-leakage and out-leakage has been observed at HTF. In 1960, a risk-significant release of liquid high-level waste stored in Tank 16 occurred due to leakage from the primary liner that overtopped the annular pan in secondary containment before waste could be pumped out of the Tank 16 annulus. The release pathway into the environment was thought to be construction joints in the tank vaults (DP-1358). Additionally, groundwater in-leakage has also been observed in several FTF and HTF tanks (WSRC-TR-93-761; WSRC-STI-2009-00352; SRR-STI-2010-00283; DOE/SRS-WD-2013-001, Rev. 0; SRNS-STI-2008-00096; DPSPU 82-11-10; SRS-REG-2007-00002, Rev. 1 and SRR-CWDA-2010-00128, Rev. 1). Therefore, the NRC staff is concerned that DOE has not adequately evaluated known release pathways

through the tank vaults into the environment in its models. Lack of consideration of these release pathways appears to be more important for Type I and II tanks at HTF, due to the potentially risk-significant inventory expected to remain in HTF annuli.

The NRC staff's concerns related to DOE's evaluation of annular inventory risk also extend to chemical transition times assumed in the deterministic and probabilistic analyses. As indicated above, DOE only considers three rates of cementitious material degradation in its parametric flow case, with a maximum value of two times faster degradation of the annular grout (and tank grout). Because the release of certain radionuclides, such as Tc-99, is sensitive to the chemical conditions in the annulus, flow through the annulus and related chemical transition times may be important to the compliance demonstration.

DOE's assumptions regarding the failure time of the partial secondary liner in tanks with annular contamination (Type I and II tanks) may also be risk significant. DOE assumes that the primary liner and secondary liners fail at the same time. While this assumption may be conservative with respect to the release of waste from the contaminated zone, this assumption may not be conservative with respect to the release of inventory from the tank annuli, if secondary liners could reasonably be expected to fail prior to failure of the primary liner. This assumption is important for tanks with a significant annular inventory and an initially intact steel liner (i.e., Tanks 9 and 10).

While benchmarking is performed on Case A with generally good results, alternative cases reflecting significantly different flow and transport conditions are not benchmarked, making it difficult to conclude that GoldSim™ and PORFLOW™ model results are reasonably comparable. Furthermore, no short-lived radionuclides with significantly different transport characteristics are evaluated in benchmarking (only Ra-226, Tc-99, I-129, Cs-135, and Np-237 are evaluated).

While intruder doses near Tank 12 are included in benchmarking, significant differences are observed between breakthrough times simulated in PORFLOW™ and GoldSim™. The differences in breakthrough times would potentially lead a decision-maker to make different compliance decisions and are, therefore, especially risk significant. However, the differences in breakthrough times are not resolved during benchmarking, but are simply attributed to extended lateral flow paths through the tank vaults in the PORFLOW™ model. Because no strong technical basis is provided to conclude that the complicated flow paths observed in PORFLOW™ are reasonable, it is not clear that the breakthrough times simulated in PORFLOW™ are any more valid than the breakthrough times simulated in GoldSim™. Therefore, the NRC staff will continue to evaluate flow through the tank vaults to determine the reasonableness of the results as discussed in Section 4.2.17.

4.2.19.4 Uncertainty Analysis Results and Recommendations

As a result of the NRC staff's review of DOE's sensitivity and uncertainty analyses, the NRC staff concludes that there is significant uncertainty in the timing and magnitude of risk-significant releases and the projected dose in DOE's HTF PA. Therefore, the NRC staff requires additional information to have reasonable assurance that the performance objectives in 10 CFR Part 61, Subpart C can be met.

Because the uncertainty and sensitivity analysis is not used to demonstrate compliance, recommendations and conclusions specific to DOE's uncertainty and sensitivity analyses can be considered in future HTF PA updates (i.e., intermediate to long-term recommendations). However, because the probabilistic analysis informs the compliance demonstration and may provide more realistic assumptions in some important areas, DOE should use the results to inform areas where additional model support is needed.

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 set out in part 20". DOE's approach to demonstrating the protection of individuals during operations (10 CFR 61.43) is to crosswalk 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 10 CFR Part 20 standards 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 the exposure of individuals during operations are maintained ALARA. These include: (1) a documented radiation protection program, (2) a documented safety analysis, (3) design of the HTF, (4) regulatory and contractual enforcement mechanisms, and (5) access controls, training, dosimetry, and monitoring. These measures are described in the draft basis document for the WD for HTF (DOE/SRS-WD-2013-001, Rev. 0).

4.3.2 NRC Evaluation of Protection of Individuals During Operations

DOE provides adequate information that individuals will be protected during operations. DOE provides a detailed crosswalk of the relevant DOE regulations to those provided in 10 CFR Part 20, which is referenced in the 10 CFR 61.43 performance objective. The NRC staff agrees that an equivalent level of protection is provided by the relevant DOE regulations as 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 ALARA including: (1) a documented radiation protection program, (2) a documented safety analysis, (3) design of the HTF, (4) regulatory and contractual enforcement mechanisms, and (5) access controls, training, dosimetry, and monitoring.

In general, the activities at the HTF 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 staff agrees with DOE that the risk to the public during operations should be minimal, and that the relevant regulatory limits can be achieved.

4.3.3 Stability of the Disposal Facility Following Closure

SRS is located on the Atlantic Coastal Plain within the Aiken Plateau. This region has relatively low seismic activity (WSRC-IM-2004-00008, Rev. 1). 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 HTF has a surface elevation ranging from 85 to 101 m (280 to 332 ft) AMSL (SRNL-STI-2010-00148, Rev. 0) and waste tank working slab elevations ranging from 73 to 89 m (240 to 293 ft) AMSL. The calculated 100,000-year flood elevations for Fourmile Branch Basin near H-Area is 72.2 m (236.8 ft) AMSL (WSRC-TR-99-00369), which is at least 0.9 m (3.1 ft) below the minimum working slab elevation of 73 m (240 ft) AMSL, based on elevations provided in DOE's HTF PA (Table 3.2-1; SRR-CWDA-2010-00128, Rev. 1). The lowest elevation of the lower foundation layer at the bottom of the side slope of the engineered closure cap is approximately 85 m (280 ft) AMSL, which is approximately 13.5 m (44 ft) above the 10,000-year floodplain level (SRR-CWDA-2010-00128, Rev. 1).

Grouting of the HLW tanks and annular spaces (for Type I, II 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, Rev. 0). DOE's stability assessment concludes that the grouted tanks would not be susceptible to cracking due to settlement or seismic loading (T-CLC-F-00421, Rev. 0).

Once all 29 HLW tanks and ancillary equipment are cleaned and grouted, a closure cap will be installed, and a 100-yr period of institutional controls will begin (SRR-CWDA-2010-00128, Rev. 1). During this period, active maintenance of the HTF facility will include prevention of pine forest succession and reparation of any significant erosion. No active maintenance is assumed to be conducted beyond the institutional control period. Beyond the institutional control period, DOE's "SRS End State Vision" includes ownership and control of the entire site by the federal government, in perpetuity, and prohibits residential use of the site. However, DOE assumes no credit is taken for the "SRS End State Vision" after a 100-year period of institutional controls, as discussed in previous sections of this TER (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, Rev. 2). 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.

Much of SRS, including HTF, is underlain by calcareous sediment that has resulted in the presence of under-consolidated "soft zones" in the Santee Formation (WSRC-TR-99-4083, Rev. 0). In conjunction with these soft zones, layers of hardened sediment are commonly observed. These layers have been characterized as bridges or arches in a honeycomb-like structure that acts to redistribute stresses. Historically, some of the soft zones have consolidated, resulting in depressions on the land surface. These depressions, or "sinks", at the SRS site typically are 3 to 5 m (10 to 15 ft) deep. In Figure 2 of SRNL-TR-2012-00160, Rev. 0, a series of sinks can be observed that are approximately 60 m (200 ft) wide by 100 m (330 ft) long. Construction activities at the Vogtle Nuclear Power Plant, located approximately 19 km (12 mi) southwest of H-Area, have revealed features associated with calcareous sediment, including soft zones, caves, and collapse features (Larrahondo-Cruz, 2011; Ku, 2012). As discussed during a technical exchange (Shaffner, 2013c [ML13154A327]), DOE believes that at

H-Area these soft zones are less of a factor than in areas downdip (e.g., Vogtle Nuclear Power Plant). However, DOE notes in SRNL-TR-2012-00160, Rev. 0 that they will continue to work with Georgia Institute of Technology on a multi-year investigation of soft zones.

The potential for mass removal of carbonate material leading to subsidence within 100 years has been dismissed by DOE due to the Santee formation being located below the water table in a relatively stable geochemical environment (i.e., negligible dissolution of calcareous material) (WSRC-TR-99-4083, Rev. 0; WSRC-TR-94-0369, Rev. 1). Furthermore, soft zones have remained stable for 40,000 years, withstanding numerous earthquakes of design-basis magnitude (WSRC-TR-94-0369, Rev. 1; SRNL-TR-2012-00160, Rev. 0).

The stability of these soft zones has been investigated since the 1950s, with the understanding and treatment of these features having evolved greatly since preliminary evaluations (e.g., U.S. Army Corps of Engineers, 1952; WSRC-TR-99-4083, Rev. 0). Early approaches to stabilizing structures built in these regions included subsurface grouting, however, it was determined that grouting campaigns provided limited benefit in mitigating potential settlement from the soft zones (WSRC-TR-99-4083, Rev. 0). 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 HTF under closure conditions (i.e., tanks grouted and cover emplaced) has not been directly evaluated. However, analyses for numerous SRS facilities have been conducted that show soft zones are 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, Rev. 0). Dynamic stability analyses for a design basis earthquake indicate that soft zones would not be susceptible to collapse (SRR-CWDA-2010-00128, Rev. 1). Although soft zone collapse is not anticipated by DOE, a hypothetical collapse of a 2.4 m (8 ft) soft zone near the FTF is calculated to result in a surface settlement of approximately 10 cm (4 in) (K-CLC-F-00034, Rev. 0).

4.3.4 NRC Evaluation of Stability of the Disposal Facility Following Closure

4.3.4.1 NRC Evaluation of Site Stability

The NRC staff agrees that grouting of the HLW tanks and annular spaces will provide a monolithic structure, minimize void space, and prevent tank collapse and differential settlement that could occur due to consolidation. In the HTF PA, DOE relies on a stability analysis that was originally conducted for the nearby FTF (T-CLC-F-00421, Rev. 0). Although several differences exist between FTF and HTF (e.g., HTF includes Type II tanks, the geometry of closure caps vary), the NRC staff does not think that the conclusions of the report would be significantly different. The 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 are modeled as a uniform monolith with a long-term grout compressive strength of 12.4 MPa (1,800 psi) and the models do not take credit for any reinforcing steel in the vault concrete or the tank itself. Grout degradation mechanisms are 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 at FTF, as discussed further in Section 4.2.9.1 of the FTF TER (Camper, 2011 [ML112371751]).

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 indicate conservatism in the calculations and appear adequate on a design basis (WSRC-TR-99-4083, Rev. 0). Although dissolution of the calcareous sediment is likely to be a very slow process, DOE has not 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 (U.S. Army Corps of Engineers, 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 (U.S. Army Corps of Engineers, 1952; WSRC-RP-92-450). This has created a soil structure that is characterized by arching, under-consolidation, and historic, periodic collapses. The U.S. Army Corps of Engineers (1952) estimates that soft zones of calcareous material would occur in H-Area at elevations from 35 to 53 m (115 to 175 ft) AMSL, noting that they pumped 676 m^3 (23,889 ft^3) of grout into 17 boreholes in H-Area. DOE's calculations do not account for the potential removal of subsurface material that has resulted in subsidence observed at SRS and HTF. DOE should account for these processes and features or demonstrate that reasonably predicted future dissolution of calcareous sediment is insignificant to site performance. During monitoring, the NRC staff will continue to review research related to the long-term stability of soft zones.

4.3.4.2 NRC Review Results and Recommendations Related to Site Stability

Similar to what is noted in the NRC staff's FTF TER (Camper, 2011 [ML112371751]) and FTF Monitoring Plan (Camper, 2013a [ML12345A322]), the NRC staff notes the following for HTF:

- DOE has provided sufficient information for the NRC staff to perform a preliminary review of site stability.

- The 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.

The NRC staff recommends that DOE perform closure cap settlement and stability analyses during the monitoring period which includes the following (Medium Risk Significance, Intermediate Term):

- Site-specific settlement analysis for HTF that includes the increased overburden from tank grout and the closure cap

- Evaluation of vault and grout integrity that is consistent with observations and reasonable expectations of future degradation of cementitious materials

- Assessment of the potential subsidence due to ongoing dissolution of calcareous sediment in the Santee formation

4.4 NRC Evaluation (Criterion 3)

The NRC staff evaluated DOE's demonstration of compliance with NDAA Criterion 3 including DOE's approaches regarding classification of the waste remaining at HTF following closure and compliance with the 10 CFR Part 61 performance objectives. The NRC staff's review results related to Criterion 3 are as follows:

- DOE's methodology for classifying waste is generally an acceptable application of Category 3 in NUREG-1854. The NRC staff notes specific exceptions in Section 4.2.1.

- DOE develops reasonable exposure scenarios to evaluate releases of radioactivity and demonstrates that the performance objective in 10 CFR 61.41 could be met. However, the NRC staff notes uncertainties in the projected releases from HTF and, principally, that DOE has provided limited support for the performance of key barriers that are considered important to demonstrating the performance objectives can be met (see Section 4.2.3). The NRC staff also provides recommendations below for DOE to develop additional model support for the HTF PA and its demonstration of compliance with the performance objective at 10 CFR 61.41. Specifically, the NRC staff notes the following results of its review of the HTF PA, supporting documentation, and DOE responses to RAIs:

    - DOE's analysis to identify, screen, and disposition FEPs in its HTF PA, *ex post facto*, improves DOE's HTF PA documentation, however, the lack of transparency and traceability in screening decisions and disposition of included FEPs makes it difficult to determine whether FEPs are comprehensively evaluated in DOE's HTF PA (see Section 4.2.3).

    - DOE's approach for assessing closure cap performance is reasonable for planning purposes and DOE has provided sufficient information regarding long-term erosion protection of the closure cap (see Section 4.2.5).

    - DOE's approach to developing inventories for tanks that have yet to be cleaned is reasonable for the purposes of assessing HTF risk, prior to development of final inventories following waste retrieval activities (see Section 4.2.7).

    - Major degradation processes appear to be considered in DOE's cementitious material degradation modeling, although the NRC staff notes large uncertainties associated with modeling degradation processes over the long time periods relied on for performance in the HTF PA (See Section 4.2.9.1).

    - Major corrosion processes appear to be considered in the steel tank liner degradation modeling; however, DOE's assumptions regarding performance of the cementitious materials that are relied on as a barrier to steel liner corrosion may be overstated while the importance of aggressive corrosion processes understated in DOE's HTF PA based on various observations (see Section 4.2.9.2).

- Although DOE's overall approach to modeling waste release is reasonable, DOE's assumptions regarding solubility-limiting phases, solubility limits, and chemical transition times are risk significant and have not been confirmed through waste characterization and experimentation (see Section 4.2.9.3).

- DOE's approach to modeling waste release from the Type I and II tanks does not adequately assess the risk from the contamination located within the annular regions (see Sections 4.2.9.3 and 4.2.9.4).

- Basemat and natural system K_ds could be risk significant for certain key radionuclides. Additional characterization, experimentation, or modeling may reduce uncertainty in assignment of risk-significant parameters such as basemat and natural system K_ds (see Sections 4.2.9.4 and 4.2.11).

- DOE's far-field model presents an acceptable framework to facilitate decision-making regarding HTF closure; however, improvements in its calibration could reduce uncertainty and enhance confidence that the level of dilution and dispersion is not overstated (see Section 4.2.11).

- Selection of a more conservatively drawn compliance boundary may significantly increase dose projections for short-lived, moderately to strongly sorbing radiological constituents (see Section 4.2.11).

- DOE's dose methodology approach is generally well supported and adequate for the purposes of demonstrating compliance with the performance objectives in 10 CFR Part 61, Subpart C, although the NRC staff notes a few exceptions (see Section 4.2.13).

- 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[38], hybrid modeling approach, benchmarking process, and evaluation and interpretation of probabilistic modeling results (see Section 4.2.19).

- As described in Section 4.2.17, DOE develops reasonable exposure scenarios to evaluate potential impacts to the inadvertent intruder and demonstrate that performance objectives in 10 CFR 61.42 could be met. However, the NRC staff notes that the probabilistic analyses indicate that projected doses could exceed 5 mSv/yr (500 mrem/yr) within 10,000 years. 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 analyses.

[38]Informed by additional data collection, peer review, or experiments, etc.

- DOE can demonstrate compliance with protection of individuals during operations (10 CFR 61.43) as discussed in Section 4.3.2.

- DOE should conduct additional analyses to demonstrate stability of the site (10 CFR 61.44) as detailed in Section 4.3.4.

The NRC staff provides several recommendations in this TER, along with a rating of risk-significance and priority[39], where additional model support would be needed prior to tank closure to provide reasonable assurance that closure activities would comply with the 10 CFR Part 61 performance objectives. Summarized below are the NRC staff's primary recommendation related to Criterion 3, which is similar to the primary recommendation that was identified by the NRC staff in the FTF TER (Camper, 2011 [ML112371751]), and other key recommendations[40]:

1. DOE should conduct waste release experiments to:

    a. Increase experimental support for key modeling assumptions related to the behavior of tank fill grout over time, including the evolution of pH and E_h (High Risk Significance, Short-to-Intermediate Term);

    b. 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);

    c. Determine constant concentrations of elements of concern under conditions of exposure to local groundwater and grout leachate via static tests (High Risk Significance, Short and Intermediate Term); and

    d. Distinguish between releases from high-solubility compounds and low-solubility compounds via semi-dynamic leach tests (Medium-to-High Risk Significance, Intermediate Term).

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

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[39] 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 (1) optional or (2) contingent on results of other analyses.

[40] NRC staff recommendations with lower risk significance related to Criterion 3 are identified and discussed in individual sections of Chapter 4 of this TER.

2. DOE should conduct a more comprehensive analysis of contaminant release from the annular regions of Type I and II tanks (Medium-to-High Risk Significance, Short and Intermediate Term).

3. The 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-Medium Risk Significance, Short and Intermediate Term).

4. DOE should continue to evaluate the appropriateness of selected transport parameters (e.g., cementitious material and soil K_ds) and selection of sorption models during the monitoring period (Medium Risk Significance, Intermediate Term).

5. DOE should improve the calibration of the far-field model, particularly local to H-Area, and continue to study uncertainty in calibration targets (Medium Risk Significance, Intermediate Term).

6. DOE should perform a closure cap settlement and stability analysis during the monitoring period (Medium Risk Significance, Intermediate Term).

5. OVERALL NRC REVIEW RESULTS AND RECOMMENDATIONS

As discussed in detail in previous sections, the NRC staff has conducted a review of DOE's draft basis for the WD for HTF at the SRS. The NRC staff's evaluation is based on information presented in DOE's draft Basis for Section 3116 Determination (DOE/SRS-WD-2013-001, Rev. 0); DOE's responses to NRC staff's RAIs; supporting references; and information provided during meetings between the NRC staff and DOE. Regarding Criteria 2 and 3 of Section 3116 of the NDAA, the NRC staff has provided a number of recommendations, the implementation of which will strengthen DOE's basis for concluding the stabilized waste in HTF tanks and associated ancillary structures and equipment can meet the NDAA criteria at the time of closure. Given that Tank 16 will be one of the first HTF tanks to be closed, the extent of contamination in its annulus, and the previous release of waste from secondary containment, the NRC staff recommends that DOE should conduct a more comprehensive analysis of contaminant release from the annulus prior to grouting Tank 16.

The NDAA requires the NRC, in coordination with the State of South Carolina, to monitor DOE disposal actions and assess compliance with the performance objectives in 10 CFR 61, Subpart C. The NRC will coordinate with the SCDHEC to develop a monitoring plan by which NRC and the state will monitor DOE's disposal actions. The NRC's monitoring plan will provide additional details regarding the priority and sequencing of recommendations necessary for the NRC to have reasonable assurance that the performance objectives can be met. For example, many of the NRC staff's recommendations under Criterion 3 depend upon the results of key waste release experiments.

It should be noted that the NRC staff is providing consultation to DOE as required by Section 3116 of the NDAA, and the NRC staff is not providing regulatory approval in this action. DOE is responsible for determining whether the waste is HLW, in consultation with the NRC. This NRC staff assessment is a site-specific evaluation and is not a precedent for any future decisions regarding non-HLW or incidental WDs at SRS or other sites.

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7. REFERENCES

[ACRi, 2008] "PORFLOW User's Manual." Version 6.0. Bel Air, California: Analytical & Computational Research, Inc. (ACRi). July 2008.

[ADM.53, Rev. 0] "Maximum Extent Practical (MEP) Documentation Process." S4 Manual Procedure ADM.53, Rev. 0. Aiken, South Carolina: Savannah River Site. November 13, 2012.

[Alexander, 2013] Alexander, G. "Technical Review: Waste Release and Solubility Related Documents Prepared by United States Department of Energy to Support Final Basis for Section 3116 Determination for the F-Area Tank Farm Facility at Savannah River Site." Memorandum to Janelle Jesse. ADAMS Accession No. ML12272A082. Rockville, Maryland: U.S. Nuclear Regulatory Commission. May 31, 2013.

[Andrade et al., 1993] Andrade, C., Alonso, C., and Molina, F.J. "Cover Cracking As a Function of Bar Corrosion: Part I—Experimental Test." *Materials and Structures*. Vol. 26, No. 8. pp. 453–464. October 1993.

[Anderson and Woessner, 1992] Anderson, M.P. and Woessner, W.W. *Applied Groundwater Modeling, Simulation of Flow, and Advective Transport*. Salt Lake City, Utah: Academic Press, Inc. 1992.

[ASTM, 2002] "Standard Guide for Calibrating a Groundwater Flow Model Application." D-5981-96. West Conshohocken, Pennsylvania: ASTM International. 2002.

[ASTM, 2008] "Standard Test Methods for Determining Chemical Durability of Nuclear, Hazardous, and Mixed Waste Glasses and Multiphase Glass Ceramics: The Product Consistency Test (PCT)." C-1285-02. West Conshohocken, Pennsylvania: ASTM International. 2008.

[Barr, 2013a] Barr, C. "Technical Review of Final Inventory Documentation for Tanks 5 and 6 at F-Area Tank Farm Facility, Savannah River Site (Project No. PROJ0734)." Memorandum to Gregory Suber. ADAMS Accession No. ML13085A291. Rockville, Maryland: U.S. Nuclear Regulatory Commission. September 30, 2013.

[Barr, 2013b] Barr, C. "Technical Review of Tanks 5 and 6 Special Analysis at F-Tank Farm Facility, Savannah River Site (Project No. PROJ0734)." Memorandum to Gregory Suber. ADAMS Accession No. ML13273A299. Rockville, Maryland: U.S. Nuclear Regulatory Commission. October 4, 2013.

[Beres, 1990] Beres, D.A. "The Clean Air Act Assessment Package-1988 (CAP-88) A Dose and Risk Assessment Methodology for Radionuclide Emissions to Air—User's Manual." Vol. 1. Washington, DC: U.S. Environmental Protection Agency. October 1990.

[BNL-71537-2003] Sullivan, T.M. "Assessment of Release Rates for Radionuclides in Activated Concrete." BNL-71537-2003. Upton, New York: Brookhaven National Laboratory. 2003.

[C-ESR-G-00003, Rev. 5] Waltz, Jr., R.S. "SRS High Level Waste Tank Leaksite Information." C-ESR-G-00003, Rev. 5. Aiken, South Carolina: Savannah River Remediation LLC. January 4, 2011.

[Camper, 2005] Camper, L.W. "Technical Evaluation Report for Draft Waste Determination for Salt Waste Disposal." Memorandum to Charles E. Anderson. ADAMS Accession No. ML053010225. Rockville, Maryland: U.S. Nuclear Regulatory Commission. December 28, 2005.

[Camper, 2011] Camper, L.W. "Technical Evaluation Report for Draft Waste Determination for Savannah River Site F Area Tank Farm." Memorandum to Frank Marcinowski. ADAMS Accession No. ML112371751. Rockville, Maryland: U.S. Nuclear Regulatory Commission. October 27, 2011.

[Camper, 2012] Camper, L.W. "Technical Evaluation Report for the 2009 Performance Assessment for the Saltstone Facility at the Savannah River Site." Memorandum to M. Gilbertson. ADAMS Accession No. ML121020140. Rockville, Maryland: U.S. Nuclear Regulatory Commission. April 30, 2012.

[Camper, 2013a] Camper, L.W. "The U.S. Nuclear Regulatory Commission Planned Monitoring Activities for F-Area Tank Farm at the Savannah River Site." Memorandum to Mark A. Gilbertson. ADAMS Accession No. ML12345A322. Rockville, Maryland: U.S. Nuclear Regulatory Commission. January 23, 2013.

[Camper, 2013b] Camper, L.W. "The U.S. Nuclear Regulatory Commission Planned Monitoring Activities for the Saltstone Disposal Facility at the Savannah River Site." Memorandum to Mark A. Gilbertson. ADAMS Accession No. ML13100A076. Rockville, Maryland: U.S. Nuclear Regulatory Commission. August 30, 2013.

[Cantrell et al., 2006] Cantrell, K.J., Krupka, K.M., Deutsch, W.J., and Lindberg, M.J. "Residual Waste From Hanford Tanks 241-C-203 and 241-C-204.2. Contaminant Release Model." *Environmental Science and Technology*. Vol. 40. pp. 3,755–3,761. 2006.

[CBP-TR-2013-02, Rev. 0] Almond, P.M., Stefanko, D.B., and Langton, C.A. "Cementitious Barriers Partnership: Effect of Oxidation on Chromium Leaching and Redox Capacity of Slag-Containing Waste Forms." CBP-TR-2013-02, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. March 2013.

[CBU-PIT-2005-00099, Rev. 0] Hamm, B.A. "Tank Farm Zeolite Historical Review and Current Inventory Assessment." CBU-PIT-2005-00099, Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. June 10, 2005.

[CBU-PIT-2005-00228, Rev. 0] Hamm, B.A. "Savannah River Site High-Level Waste Tank Farm Closure Radionuclide Screening Process (First-Level) Development and Application." CBU-PIT-2005-00228, Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company, Savannah River Site. November 2006.

[Clear, 1976] Clear, K.C. "Time to Corrosion of Reinforcing Steel in Concrete Slabs." FHWA-RD-76-70. Washington, DC: Federal Highway Administration. April 1976.

[Demirkanli et al., 2007] Demirkanli, D.I., Molz, F.J., Kaplan, D.I., Fjeld, R.A., and Serkiz, S.M. "Modeling Long-Term Plutonium Transport in the Savannah River Site Vadose Zone." *Vadose Zone Journal*. Vol.6. pp. 344–353. 2007.

[DOE, 1990] "Application for a Hazardous Waste Part B Postclosure Care Permit, H-Area Hazardous Waste Management Facility." *Lithostratigraphic Sequence of the Hydrogeology Units at the GSA*. Rev. 2. Vol. 5, Book 2. p. 154. Aiken, South Carolina: U.S. Department of Energy, Savannah River Plant. December 1990.

[DOE/EIS-0303] "High-Level Waste Tank Closure Final Environmental Impact Statement." DOE/EIS-0303. Aiken, South Carolina: U.S. Department of Energy, Savannah River Operations Office. May 2002.

[DOE M 435.1-1] "Radioactive Waste Management Manual." DOE M 435.1-1. Change 1. Washington, DC: U.S. Department of Energy. January 9, 2007.

[DOE/SRS-WD-2005-001a] "Draft Basis for Section 3116 Determination Salt Waste Disposal Savannah River Site." DOE/SRS-WD-2005-001. Aiken, South Carolina: U.S. Department of Energy, Savannah River Site. February 2005.

[DOE/SRS-WD-2005-001b] "Basis for Section 3116 Determination for Salt Waste Disposal at the Savannah River Site." DOE/SRS-WD-2005-001. Aiken, South Carolina: U.S. Department of Energy, Savannah River Site. January 2006.

[DOE/SRS-WD-2010-001, Rev. 0] "Draft Basis Document for Section 3116 Determination for F-Tank Farm at the Savannah River Site." DOE/SRS-WD-2010-001, Rev. 0. Aiken, South Carolina: U.S. Department of Energy, Savannah River Site. September 2010.

[DOE/SRS-WD-2012-001, Rev. 0] "Basis for Section 3116 Determination for Closure of F-Tank Farm at the Savannah River Site." DOE/SRS-WD-2012-001, Rev. 0. Aiken, South Carolina: U.S. Department of Energy, Savannah River Site. March 2012.

[DOE/SRS-WD-2013-001, Rev. 0] "Draft Basis for Section 3116 Determination for Closure of H-Tank Farm at the Savannah River Site." DOE/SRS-WD-2013-001, Rev. 0. Aiken, South Carolina: U.S. Department of Energy, Savannah River Site. February 6, 2013.

[DP-1358] Poe, W.L. "Leakage From Waste Tank 16, Amount, Fate, and Impact." DP-1358. Aiken, South Carolina: E.I. du Pont de Nemours and Company and Savannah River Laboratory. November 1974.

[DPSP-80-17-21] Snell, E.B. "Second Tank 16 Annulus Cleaning Test Process and Equipment Description." Memorandum to J.H. Hershey. DPSP-80-17-21. Aiken, South Carolina: E.I. du Pont de Nemours and Company, Savannah River Plant. October 8, 1980.

[DPSPU 82-11-10] McNatt, F.G. "History of Waste Tank 20: 1959 Through 1974." DPSPU 82-11-10. Aiken, South Carolina: Savannah River Site. July 1982.

[ENG.50, Rev. 2] "LW Project and Closure Operating Plans." S4 Manual Procedure ENG.50, Rev. 2. Aiken, South Carolina: Savannah River Site. November 13, 2012.

[EPA, 1988] U.S. Environmental Protection Agency. "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." Federal Guidance Report No. 11. EPA-520/1-88-020. Washington, DC: U.S. Environmental Protection Agency, Office of Radiation Programs. 1988.

[EPA, 1993] U.S. Environmental Protection Agency. "External Exposure to Radionuclides in Air, Water, and Soil." Federal Guidance Report No. 12. EPA-402-R-93-081. Washington, DC: U.S. Environmental Protection Agency, Office of Radiation and Indoor Air. 1993.

[Evans, 1995] Evans, E.K. "Characteristics of Hydraulic Conductivity Heterogeneity in Tertiary Sediments Within the GSA." Thesis, Pennsylvania State University. Aiken, South Carolina: Savannah River Site. 1995.

[Felsher, 2013a] Felsher, H. "NRC's Comments on the SRS FTF Closure Module for Tanks 5F and 6F." Electronic Mail to Bridget Clarke. ADAMS Accession No. ML13081A050. Rockville, Maryland: U.S. Nuclear Regulatory Commission. March 21, 2013.

[Felsher, 2013b] Felsher, H. "Summary of the May 15, 2013 Clarifying Teleconference Call Between the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy Regarding F-Area Tank Farm Volume Mapping and Inventory for the Savannah River Site (Docket No. PROJ0734)." Note to File PROJ0734. ADAMS Accession No. ML13150A219. Rockville, Maryland: U.S. Nuclear Regulatory Commission. June 18, 2013.

[Flanders, 2006]. Flanders, S.C.. "Request for Additional Information on the Draft Section 3116 Determination for Closure of Tank 19 and Tank 18 at the Savannah River Site." Memorandum to M.A. Gilbertson. ADAMS Accession No. ML060800295. Rockville, Maryland: U.S. Nuclear Regulatory Commission. March 31, 2006.

[Friedman, 2002] Friedman, J. H. "Stochastic Gradient Boosting." *Computational Statistics & Data Analysis*. Vol. 38, No. 4. pp. 367-378. 2002.

[G-ESR-G-00051, Rev. 0] Zupon, D. "A Waste Removal, Balance of Program, Systems Engineering Evaluation Report." G-ESR-G-00051. Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. September 17, 2003.

[G-SQP-A-00012] "PORFLOW Software Quality Assurance Plan." G-SQP-A-00012. Aiken, South Carolina: Savannah River Site. June 18, 2007.

[G-STP-A-00009] "PORFLOW Software Test Plan." G-STP-A-00009. Aiken, South Carolina: Savannah River Site. June 2007.

[G-TR-G-00002, Rev. 0.] Whiteside, T. "Software Testing and Verification for PORFLOW Version 6.10.3." G-TR-G-00002. Rev. 0. Aiken, South Carolina: Savannah River Site. October 2007.

[Gelhar et al., 1992] Gelhar, L.W., Welty,C., and Rehfeldt, K.R. "A Critical Review of Data on Field-Scale Dispersion in Aquifers." *Water Resources Research*. Vol. 28, No. 7. pp. 1,955–1,974. 1992.

[GoldSim Technology Group LLC, 2010] "GoldSim Probabilistic Simulation Environment User's Guide." Version 10.5, Vols. 1 and 2. Issaquah, Washington: GoldSim Technology Group LLC. December 2010.

[Harris et al., 1997] Harris, A.W., Atkinson, A., Cole, G.B., Haworth, A., Nickerson, A.K., and Smith, A.C. "The Performance of Cementitious Barriers in Repositories." Harwell, United Kingdom: AEA Technology. 1997.

[Hobbs, 1999] Hobbs, D.T. "Precipitation of Uranium and Plutonium From Alkaline Salt Solutions. " *Nuclear Technology.* Vol. 128. pp. 103–112. 1999.

[IAEA, 2004] "Safety Assessment Methodologies for Near Surface Disposal Facilities, Results of a Co-Ordinated Research Project." Vol. 1. Vienna, Austria: International Atomic Energy Agency. July 2004.

[IAEA, 2010] "Handbook of Parameter Values for the Prediction of Radionuclide Transfer in Terrestrial and Freshwater Environments." Technical Report Series No. 472. Vienna, Austria: International Atomic Energy Agency. 2010.

[ICRP, 1995] "Radiation Protection: ICRP Publication #72—Age-Dependent Doses to Members of the Public From Intake of Radionuclides: Part 5 Compilation of Ingestion and Inhalation Dose Coefficients." Didcot Oxfordshire, United Kingdom: International Commission on Radiological Protection. September 1995.

[Johnson, 1942] Johnson, D.W. *The Origin of the Carolina Bays*. New York City, New York: Columbia University Press. 1942.

[K-CLC-F-00034, Rev. 0] McHood, M.D. "APSF Soft Zone Settlement Analysis." Calculation No. K-CLC-F-00034, Rev. 0. Aiken, South Carolina: Savannah River Site. 1998.

[Kaplan et al., 2006] Kaplan, D.I., Demirkanli, D.I., Gumapas, L., Powell, B.A., Fjeld, R.A., Molz, F.J., and Serkiz, S.M. "11-Year Field Study of Pu Migration from Pu III, IV, and VI Sources." *Environmental Science and Technology*. Vol. 40. pp. 443–448. 2006.

[King et al., 2010] King, W.D., Edwards, T.B., Hobbs, D.T., and Wilmarth, W.R. "Solubility of Uranium and Plutonium in Alkaline Savannah River Site High-Level Waste Solutions." *Separation Science and Technology.* Vol. 45. pp.1,793–1,800. 2010.

[Krupka et al., 2006] Krupka, K. et al. "Residual Waste From Hanford Tanks 241-C-203 and 241-C-204.1 Solids Characterization." *Environmental Science and Technology*. Vol. 40. pp. 3,749–3,754. 2006.

[Ku, 2012] Ku, T. "Geostatic Stress Evaluation by Directional Shear Wave Velocities, With Application Towards Geocharacterization at Aiken, SC." Ph.D. Dissertation. Atlanta, Georgia: Georgia Institute of Technology, School of Civil and Environmental Engineering. p. 351. 2012.

[Langmuir, 1997] Langmuir, D. *Aqueous Environmental Geochemistry*. Upper Saddle River, New Jersey: Prentice Hall. 1997.

[Larrahondo-Cruz, 2011] Larrahondo-Cruz J.M. "Carbonate Diagnoses and Chemical Weathering in the Southeastern United States: Some Implications on Geotechnical Behavior." Ph.D. Dissertation. Atlanta, Georgia: Georgia Institute of Technology, School of Civil and Environmental Engineering. p. 320. 2011.

[Lowman, 2010] Lowman, D. "Savannah River Site H-Area Tank Farm Input Packages/References." Note to PROJ0734. ADAMS Accession No. ML100970781. Rockville, Maryland: U.S. Nuclear Regulatory Commission. April 13, 2010.

[Marcinowski, 2007] Marcinowski, F. "Path Forward for Consultation on Section 3116 Determinations for Closure of Tanks at the Savannah River Site." Memorandum to L. Camper. ADAMS Accession No. ML080090405. Washington, DC: U.S. Department of Energy. December 20, 2007.

[May and Warne, 1999] May, J.H. and Warne, A.G. "Hydrogeologic and Geochemical Factors Required For The Development of Carolina Bays Along the Atlantic and Gulf of Mexico, Coastal Plain, USA." *Environmental and Engineering Geoscience.* Vol. 5, No. 3. pp. 261–270. 1999.

[McKay et al., 1979] McKay, M.D., Beckman, R.J., and Conover, W.J. "A Comparison of Three Methods for Selecting Values of Input Variables in the Analysis of Output From a Computer Code." *Technometrics*. Vol. 21, No. 2. pp. 239–245. May 1979.

[Mohanty and Codell, 2004] Mohanty, S and Codell, R B. "Ramifications of Risk Measures in Implementing Quantitative Performance Assessment for the Proposed Radioactive Waste Repository at Yucca Mountain, Nevada USA." *Risk Analysis*. Vol. 24. pp. 537–546. 2004.

[Mohseni, 2013a] Mohseni, A. "U.S. Nuclear Regulatory Commission Staff Comments and Request for Additional Information on the 'Draft Basis for Section 3116 Determination and Associated Performance Assessment for the H-Area Tank Farm at the Savannah River Site,' DOE/SRS-WD-2013-001, Rev. 0 and on 'Performance Assessment for the H-Area Tank Farm for the Savannah River Site,' SRR-CWDA-2012-00128, Rev. 1." Memorandum to William Levitan. ADAMS Accession No. ML13196A135. Rockville, Maryland: U.S. Nuclear Regulatory Commission. July 31, 2013.

[Mohseni, 2013b] Mohseni, A. "The U.S. Nuclear Regulatory Commission August 27–28, 2013, Onsite Observation Visit Report for the Savannah River Site F-Tank Farm Facility (Docket No. PROJ0734)." Memorandum to Jean Ridley. ADAMS Accession No. ML13267A452. Rockville, Maryland: U.S. Nuclear Regulatory Commission. November 12, 2013.

[Molina et al., 1993] Molina, F.J., Alonso, C., and Andrade, C. "Cover Cracking As a Function of Rebar Corrosion: Part 2—Numerical Model." *Materials and Structures*. Vol. 26, No. 9. pp. 532–548. November 1993.

[NBSIR 81-2228] Gerhold, W.F., Escalante, E., and Sanderson, B.T. "The Corrosion Behavior of Selected Stainless Steels in Soil Environments." NBSIR 81-2228. Gaithersburg, Maryland: National Bureau of Standards. 1981.

[NCRP-123] "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground: Volumes 1 and 2." NCRP-123 (Copyright). Bethesda, Maryland: National Council on Radiation Protection and Measurements (NCRP). January 1996.

[Neville, 1996] Neville, A.M. "Properties of Concrete." *Durability of Concrete*. Chapter 10. London, United Kingdom: John Wiley & Sons, Inc. 1996.

[NRC, 2001a] U.S. Nuclear Regulatory Commission. 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," *Code of Federal Regulations.* Washington, DC: Office of the Federal Register. January 2001.

[NRC, 2001b] U.S. Nuclear Regulatory Commission, 10 CFR Part 20, "Standards for Protection Against Radiation," *Code of Federal Regulations.* Washington, DC: Office of the Federal Register. January 2001.

[NUREG-0782] "Draft Environmental Impact Statement on 10 CFR Part 61, Licensing Requirements for Land Disposal of Radioactive Waste." Vol. 4. Washington, DC: U.S. Nuclear Regulatory Commission. September 1981.

[NUREG-1563] Kotra, J.P., Lee, M.P., Eisenberg, N.A., and DeWispelare, A.R. "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program." ADAMS Accession No. ML033500190. Washington, DC: U.S. Nuclear Regulatory Commission. November 1996.

[NUREG-1573] "A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities—Recommendations of NRC's Performance Assessment Working Group." NUREG-1573. ADAMS Accession No. ML003770778. Washington, DC: U.S. Nuclear Regulatory Commission. October 2000.

[NUREG-1623] Johnson, T.L. "Design of Erosion Protection for Long-Term Stabilization." NUREG-1623. Washington, DC: U.S. Nuclear Regulatory Commission. September 2002.

[NUREG-1757] Schmidt, D.W., Banovac, K.L., Buckley, J.T., Esh, D.W., Johnson, R.L., Kottan, J.J., McKenney, C.A., McLaughlin, T.G., and Schneider, S. "Consolidated NMSS Decommissioning Guidance, Characterization, Survey, and Determination of Radiological Criteria." NUREG-1757, Vol. 2, Rev. 1. Washington, DC: U.S. Nuclear Regulatory Commission. September 2003.

[NUREG-1854] Arlt, H., Bradford, B., Devaser, N., Esh, D., Fuller, M., Ridge, A., Brient, B., LaPlante, P., Mackin, P., Pearcy, E., Turner, D., and Winterle, J. "NRC Staff Guidance for Activities Related to US Department of Energy Waste Determination." NUREG-1854. Washington, DC: U.S. Nuclear Regulatory Commission. August 2007.

[NUREG/CR-5542] Walton, J.C., Plansky, L.E., and Smith, R.W. "Models for Estimation of Service Life of Concrete Barriers in Low-Level Radioactive Waste Disposal." NUREG/CR-5542. Washington, DC: U.S. Nuclear Regulatory Commission. September 1990.

[Pabalan et al., 2012] Pabalan, R.T., Alexander, G.W., and Waiting, D.J. "Experimental Study of Contaminant Release from Reducing Grout." ADAMS Accession No. ML12089A319. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. March 2012.

[Pabalan et al., 2009] Pabalan, R.T., Glasser, F.P., Pickett, D.A., Walter, G.R., Biswas, S., Juckett, M.R., Sabido, L.M., and Myers, J.L. "Review of Literature and Assessment of Factors Relevant to Performance of Grouted Systems for Radioactive Waste Disposal." CNWRA 2009-001. ADAMS Accession No. ML090980426. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2009.

[Painter and Pabalan, 2009] Painter, S.L. and Pabalan, R.T. "Estimated Longevity of Reducing Environment in Grouted Systems for Radioactive Waste Disposal." ADAMS Accession No. ML101160513. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2009.

[Papadakis and Fardis, 1989] Papadakis, V.G. and Fardis, M.N. "A Reaction Engineering Approach to the Problem of Concrete Carbonation." *AICHE Journal*. Vol. 35. pp. 1,639–1,650. 1989.

[Parks, 2013]. Parks, L. "Technical Review Updated Cost-Benefit Analysis for Removal of Additional Highly Radioactive Radionuclides From Tank 18." Memorandum to Gregory Suber. ADAMS Accession No. ML13080A401. Rockville, Maryland: U.S. Nuclear Regulatory Commission. March 21, 2013.

[Parrot, 1990] Parrott, L.J. "Damage Caused by Carbonation of Reinforced Concrete." *Materials and Structures*. Vol. 23. pp. 230–234. 1990.

[Perez-Lopez et al., 2005] Perez-Lopez, R., Cama, J. Nieto, J.M., and Ayora, C. "The Role of Coating in the Oxidative Dissolution of a Pyrite-Rich Sludge." *Proceedings of the 9th International Mine Water Association Congress*. pp. 575-579. 2005.

[PIT-MISC-0040] Davis, N.R. "Tank 19 Heel Removal Systems Engineering Plan." PIT-MISC-0040. Aiken, South Carolina: Savannah River Site. 1998.

[PIT-MISC-0089] "Savannah River Site End State Vision." PIT-MISC-0089. Aiken, South Carolina: Savannah River Site. July 2005.

[PNNL-21723] Cantrell, K.J. and Williams, B.D. "Equilibrium Solubility Model for Technetium Release From Saltstone Based on Anoxic Single-Pass Flow Through Experiments." PNNL-21723. Richland, Washington: Pacific Northwest National Laboratory. September 2012.

[PNNL-22977] Um, W., Jung, H.B., Wang, G., Westsik, Jr., J.H., and Peterson, R.A. "Characterization of Technetium Speciation in Cast Stone." PNNL-22977. Richland, Washington: Pacific Northwest National Laboratory. November 2013.

[PORTAGE-08-022, Rev. 0] "H-Area Tank Farm Model Development for Tanks in the Water Table: PORFLOW™ Version 6.20.0." PORTAGE-08-022. Rev. 0. Idaho Falls, Idaho: Portage Inc. November 2008.

[Prikryl and Pickett, 2007] Prikryl, J.D. and Pickett, D.A. "Recommended Site-Specific Sorption Coefficients for Reviewing Non-High-Level Waste Determinations at the Savannah River Site and Idaho National Laboratory." ADAMS Accession No. ML073510127. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. October 2007.

[Rumbaugh, 1998] Rumbaugh, J. "Applied Techniques for Model Calibration and Stochastic Simulation Course Notebook." Denver, Colorado: Environmental Simulations, Inc. 1998.

[Sadler, 1993] Sadler, M.F. "Construction Permit #17,424-IW, SRS F/H-Area, Aiken and Barnwell County." Columbia, South Carolina: South Carolina Department of Health and Environmental Control. January 25, 1993.

[SAND-79-1473] Iman, R.L., Davenport, J.M., and Zeigler, D.K. "Latin Hypercube Sampling (Program User's Guide). [LHC, in FORTRAN]." SAND-79-1473. Albuquerque, New Mexico: Sandia National Laboratory. 1980.

[Schroeder et al., 1994a] Schroeder, P.R., Lloyd, C.M, Zappi, P.A., and Aziz, N.M. "The Hydrologic Evaluation of Landfill Performance (HELP) Model: User's Guide for Version 3." EPA-600-R-94-168A. Cincinnati, Ohio: U.S. Environmental Protection Agency, Office of Research and Development. September 1994.

[Schroeder et al., 1994b] Schroeder, P.R., Dozier, T.S., Zappi, P.A., McEnroe, B.M., Sjostrom, J.W., and Peyton, R.L. "The Hydrologic Evaluation of Landfill Performance (HELP) Model: Engineering Documentation for Version 3." EPA-600-R-94-168B. Cincinnati, Ohio: U.S. Environmental Protection Agency, Office of Research and Development. September 1994.

[Shaffner, 2011a] Shaffner, J. "Summary of Teleconference Between U.S. Nuclear Regulatory Commission Staff and U.S. Department of Energy Representatives Concerning Requests for Additional Information Pertaining to the Draft Waste Determination and Related Performance Assessment Related to the Closure of the F Area Tank Farm at the Savannah River Site." Note to File PROJ0734. ADAMS Accession No. ML110250118. Rockville, Maryland: U.S. Nuclear Regulatory Commission. February 7, 2011.

[Shaffner, 2011b]. Shaffner, J. "Summary of Teleconference Between the U.S. Nuclear Regulatory Commission Staff and the U.S. Department of Energy Representatives Concerning Responses to Requests for Additional Information Related to the Closure of the F-Tank Farm, Savannah River Site." Note to File PROJ0734. ADAMS Accession No. ML112580225. Rockville, Maryland: U.S. Nuclear Regulatory Commission. September 23, 2011.

[Shaffner, 2012] Shaffner, J. "Summary of Telephone Conference Call Held on July 26, 2012, Between the U.S. Nuclear Regulatory Commission Staff and Department of Energy Representatives Concerning Requests for Additional Information/Clarification Pertaining to the Residual Waste Solubility Related to Removal of Highly Radioactive Radionuclides from Tank 18, F Area Tank Farm. Not to File PROJ0734. ADAMS Accession No. ML12236A370. Rockville, Maryland: U.S. Nuclear Regulatory Commission. September 7, 2012.

[Shaffner, 2013a] Shaffner, J. "Summary of Clarification Discussion Between the U.S. Nuclear Regulatory Commission Staff and the U.S. Department of Energy and Savannah River Remediation Staff Concerning Waste Release Potential Related to H Area Tank Farm at the Savannah River Site." Note to File PROJ0734H. ADAMS Accession No. ML13106A338. Rockville, Maryland: U.S. Nuclear Regulatory Commission. May 3, 2013.

[Shaffner, 2013b] Shaffner, J. "Summary of Clarification Discussion Between the U.S. Nuclear Regulatory Commission Staff and the U.S. Department of Energy and Savannah River Remediation Staff Concerning Flow and Transport of Water and Contaminants related to H-Area Tank Farm at the Savannah River Site." Memorandum to File. ADAMS Accession No. ML13126A127. Rockville, Maryland: U.S. Nuclear Regulatory Commission. May 10, 2013.

[Shaffner, 2013c] Shaffner, J. "Summary of Clarification Discussion Between U.S. Nuclear Regulatory Commission Staff and U.S. Department of Energy and Savannah River Remediation Staff Concerning Flow Modeling and Calibration Related to H Area Tank Farm at the Savannah River Site." Note to File PROJ0734. ADAMS Accession No. ML13154A327. Rockville, Maryland: U.S. Nuclear Regulatory Commission. July 5, 2013.

[Shaffner, 2013d] Shaffner, J. "Summary of Clarification Discussion Between the U.S. Nuclear Regulatory Commission Staff and U.S. Department of Energy and Savannah River Remediation Staff Concerning Flow Modeling and Calibration Related to H Area Tank Farm at the Savannah River Site." Note to File PROJ0734. ADAMS Accession No. ML13193A072. Rockville, Maryland: U.S. Nuclear Regulatory Commission. July 16, 2013.

[Shaffner, 2013e] Shaffner, J. "Summary of (1) Public Meeting Between NRC and DOE Concerning RAIs Related to Proposed Waste Determination for Closure of H Area Tank Farm at SRS and (2) Consultation Site Visit by NRC Staff to H Tank Farm." Note to File PROJ0734. ADAMS Accession No. ML13183A410. Rockville, Maryland: U.S. Nuclear Regulatory Commission. August 1, 2013.

[Shaffner, 2013f] Shaffner, J. "Summary of Clarification Discussion Between U.S. Nuclear Regulatory Commission Staff and U.S. Department of Energy and Savannah River Remediation Staff Concerning Performance Assessment Modelling Uncertainty/Sensitivity Related to H Area Tank Farm at the Savannah River Site." Note to File PROJ0734. ADAMS Accession No. ML13199A413. Rockville, Maryland: U.S. Nuclear Regulatory Commission. July 29, 2013.

[Shaffner, 2013g] Shaffner, J. "Follow-Up RAI Response Questions." Electronic Mail to Sherri Ross. ADAMS Accession No. ML14016A391. Rockville, Maryland: U.S. Nuclear Regulatory Commission. November 21, 2013.

[Siple, 1967] Siple, G.E. "Geology and Ground Water of the Savannah River Plant and Vicinity South Carolina." U.S. Geological Survey Water-Supply. Paper 1841. p. 113. 1967.

[SRM-SECY-92-391] "SECY-92-391 – Denial of PRM-60-4 – Petition for Rulemaking from the States of Washington and Oregon Regarding Classification of Radioactive Waste at Hanford." Staff Requirements Memorandum. SRM-SECY-92-391. ADAMS Accession Number ML003760236. Washington, DC: U.S. Nuclear Regulatory Commission. February 16, 1993.

[SRNL-RP-2013-00203, Rev. 0] Hobbs, D.T., Taylor-Pashow, K., Roberts, K., and C.Langton. "Task Technical and Quality Assurance Plan for Determining the Radionuclide Release from Tank Waste Residual Solids." SRNL-RP-2013-00203, Rev. 0 Aiken, South Carolina: Savannah River National Laboratory. 2013.

[SRNL-STI-2009-00150, Rev. 1] McDowell-Boyer, L.M., et al. "Distribution Coefficients (K_ds), K_dsDistributions, and Cellulose Degradation Product Correction Factors for the Composite Analysis." SRNL-STI-2009-00150, Rev. 1. Aiken, South Carolina: Savannah River National Laboratory. 2009.

[SRNL-STI-2009-00473, Rev. 0] Kaplan, D.I. "Geochemical Data Package for Performance Assessment Calculations Related to the Savannah River Site." SRNL-STI-2009-00473, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2010.

[SRNL-STI-2009-00492, Rev. 0] Poirier, M.R. and Fink, S.D. "Analysis of Samples From Tank 5F Chemical Cleaning." SRNL-STI-2009-00492, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2009.

[SRNL-STI-2009-00493, Rev. 0] Poirier, M.R. and Fink, S.D. "Analysis of Samples From Tank 6F Chemical Cleaning." SRNL-STI-2009-00493, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2010.

[SRNL-STI-2009-00634, Rev. 0] Miller, T.J., Powell, B.A., and Kaplan, D.I. "Neptunium IV and V Sorption to End-Member Subsurface Sediments of the Savannah River Site." SRNL-STI-2009-00634, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2009.

[SRNL-STI-2009-00636, Rev. 0] Lilley, M.S., Powell, B.A., and Kaplan, D.I. "Iodine, Neptunium, Plutonium, and Technetium Sorption to Saltstone and Cement Formulations Under Oxidizing and Reducing Conditions." SRNL-STI-2009-00636, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2009.

[SRNL-STI-2009-00637, Rev. 0] Roberts, K. and Kaplan, D.I. "Reduction Capacity of Saltstone and Saltstone Components." SRNL-STI-2009-00637, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2009.

[SRNL-STI-2010-00035, Rev. 0] Langton, C.A. Chemical Degradation Assessment for the H-Area Tank Farm Concrete Tanks and Fill Grouts." SRNL-STI-2010-00035, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2010.

[SRNL-STI-2010-00047] García-Díaz, B.L. "Life Estimation of High-Level Waste Tank Steel for H-Tank Farm Closure Performance Assessment." SRNL-STI-2010-00047. Aiken, South Carolina: Savannah River National Laboratory. 2010.

[SRNL-STI-2010-00148, Rev. 0] Jones, W.E., Millings, M.R., and Rambo, B.H. "Hydrogeologic Data Summary in Support of the H-Area Tank Farm Performance Assessment." SRNL-STI-2010-00148, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2010.

[SRNL-STI-2010-00447, Rev. 0] Jannik, G.T., Karapatakis, D.J., and Lee, P.L. "Land and Water Use Characteristics and Human Health Input Parameters for Use in Environmental Dosimetry and Risk Assessments at the Savannah River Site." SRNL-STI-2010-00447, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. August 6, 2010.

[SRNL-STI-2011-00011, Rev. 0] Kaplan, D.L. "Estimated Neptunium Sediment Sorption Values As a Function of pH and Measured Barium and Radium Kd Values." SRNL-STI-2011-00011, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2011.

[SRNL-STI-2011-00672] Almond, P.M., Kaplan, D.I., and Shine, E.P. "Variability of K_d Values in Cementitious Materials and Sediments." SRNL-STI-2011-00672. Aiken, South Carolina: Savannah River National Laboratory. 2012.

[SRNL-STI-2012-00087, Rev. 0] Denham, M. "Evolution of Chemical Conditions and Estimated Plutonium Solubility in the Residual Waste Layer During Post-Closure Aging of Tank 18." SRNL-STI-2012-00087, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2012.

[SRNL-STI-2012-00106, Rev. 0] Hobbs, D.T. "Form and Aging of Plutonium in Savannah River Site Waste Tank 18." SRNL-STI-2012-00106, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2012.

[SRNL-STI-2012-00123, Rev. 0] Hay, M.S., O'Rourke, P.E., and Ajo, H.M. "Scanning Electron Microscopy and X-Ray Diffraction Analysis of Tank 18 Samples." SRNL-STI-2012-00123, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2012.

[SRNL-STI-2012-00178, Rev. 0] Hay, M.S. and Reboul, S.H. "Characterization of Tank 16H Annulus Samples." SRNL-STI-2012-00178, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2012.

[SRNL-STI-2012-00404, Rev. 0] Denham, M. and Millings, M. "Evolution of Chemical Conditions and Estimated Solubility Controls on Radionuclides in the Residual Waste Layer During Post-Closure Aging of High-Level Waste Tank." SRNL-STI-2012-00404, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2012.

[SRNL-STI-2012-00465] Jordan, J.M., Flach, G.P., and Westbrook, M.L. Westbrook. "PORFLOW Modeling Supporting the H-Tank Farm Performance Assessment." SRNL-STI-2012-00465. Aiken, South Carolina: Savannah River National Laboratory. 2012.

[SRNL-STI-2012-00479, Rev. 0] Reboul, S.H. "Chemical Differences Between Sludge Solids at the F and H Area Tank Farms." SRNL-STI-2012-00479, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2012.

[SRNL-STI-2013-00541, Rev. 0] Langton, C.A. and Almond, P.M. "Cast Stone Oxidation Front Evaluation: Preliminary Results for Samples Exposed to Moist Air." SRNL-STI-2013-00541, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2013.

[SRNL-TR-2009-00019, Rev. 0] Kaplan, D.I. "Tc and Pu Distribution Coefficients, K_ds Values for the Saltstone Facility Performance Assessment." SRNL-TR-2009-00019, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2009.

[SRNL-TR-2010-00023, Rev. 0] Whiteside, T. "2010 Software Testing and Verification of PORFLOW™ Version 6.21.0. SRNL-TR-2010-00023, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2010.

[SRNL-TR-2010-00195, Rev. 0] Whiteside, T. "2010 Software Testing and Verification of PORFLOW™ Versions 6.30.0 and 6.30.1." SRNL-TR-2010-00195, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2010.

[SRNL-TR-2010-00213, Rev. 0] Whiteside, T. "2010 Software Testing and Verification of PORFLOW™ Versions 6.30.1 and 6.30.2. SRNL-TR-2010-00213, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2010.

[SRNL-TR-2012-00160, Rev. 0] Bagwell, L. "A Review of Subsurface Soft Zones at Savannah River Site With Emphasis on H-Area Tank Farm (U)." SRNL-TR-2012-00160, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2012.

[SRNS-TR-2008-00003, Rev. 0] Whiteside, T. "Software Testing and Verification for PORFLOW™." SRNS-TR-2008-00003, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2008.

[SRNS-STI-2008-00096] Wiersma, B.J. "An Assessment of the Service History and Corrosion Susceptibility of Type IV Waste Tanks." SRNS-STI-2008-00096. Aiken, South Carolina: Savannah River Nuclear Solutions LLC. 2008.

[SRR-CES-2009-00022, Rev. 0] Clendenen, G.B. "Strategy for Tanks 5 and 6 Phase II Mechanical Sludge Removal." SRR-CES-2009-00022, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2009.

[SRR-CWDA-2009-00017, Rev. 0] "Performance Assessment for the Saltstone Disposal Facility at the Savannah River Site." SRR-CWDA-2009-00017, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. October 2009.

[SRR-CWDA-2009-00054, Rev. 0] "Comment Response Matrix for Nuclear Regulatory Commission (NRC) Comments on the F-Tank Farm Performance Assessment (SRS-REG-2007-00002, Revision 0, dated June 27, 2008)." SRR-CWDA-2009-00054, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. March 31, 2010.

[SRR-CWDA-2010-00019, Rev. 0] Watkins, D.R. "H-Area Tank Farm Grout and Concrete Degradation Modeling Information." Memorandum to K.H. Rosenberger. SRR-CWDA-2010-00019, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. March 15, 2010.

[SRR-CWDA-2010-00023, Rev. 3] Dean, B. "H-Area Tank Farm Closure Inventory for Use in Performance Assessment Modeling." SRR-CWDA-2010-00023, Rev. 3. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-CWDA-2010-00033, Rev. 1] "Comment Response Matrix for Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAIs) on the Saltstone Disposal Facility Performance Assessment (SRR-CWDA-2009-00017, Rev. 0, dated October 29, 2009)." SRR-CWDA-2010-00033, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. July 23, 2010.

[SRR-CWDA-2010-00093, Rev. 2] Lester, B. "H-Area Tank Farm Stochastic Fate and Transport Model." SRR-CWDA-2010-00093, Rev. 2. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-CWDA-2010-00128, Rev. 0] "Performance Assessment for the F-Tank Farm at the Savannah River Site." SRR-CWDA-2010-00128, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2008.

[SRR-CWDA-2010-00128, Rev. 1] "Performance Assessment for the H-Area Tank Farm at the Savannah River Site." SRR-CWDA-2010-00128, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-CWDA-2011-00005, Rev. 1] Stephens, B.A. and Gilbreath, K.D. "Tank 6F History of Waste Removal 1964 Through 2010." SRR-CWDA-2011-00005, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2011.

[SRR-CWDA-2011-00022, Rev. 0] "Industrial Wastewater General Closure Plan for H-Area Waste Tank Systems." SRR-CWDA-2011-00022, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2010.

[SRR-CWDA-2011-00033, Rev. 1] Stephens, B.A. and Gilbreath, K.D. "Tank 5F History of Waste Removal 1959 Through 2010." SRR-CWDA-2011-00033, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2011.

[SRR-CWDA-2011-00044, Rev. 1] "Comment Response Matrix for Nuclear Regulatory Commission RAI-2009-02 Second Request for Additional Information (RAI) on the Saltstone Disposal Facility Performance Assessment." SRR-CWDA-2011-00044, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2011.

[SRR-CWDA-2011-00050, Rev. 1] Pavletich, J.P. "Liquid Waste Tank Residuals Sampling and Analysis Program Plan." SRR-CWDA-2011-00050, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-CWDA-2011-00050, Rev. 2] Pavletich, J.P. "Liquid Waste Tank Residuals Sampling and Analysis Program Plan." SRR-CWDA-2011-00050, Rev. 2. Aiken, South Carolina: Savannah River Remediation LLC. 2013.

[SRR-CWDA-2011-00054, Rev. 1] “Comment Response Matrix for United States Nuclear Regulatory Commission Staff Comments on the Draft Basis for Section 3116 Determination and Associated Performance Assessment for the F-Tank Farm at the Savannah River Site.” SRR-CWDA-2011-00054, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2011.

[SRR-CWDA-2011-00091, Rev. 0] ”Documentation of Removal of Highly Radioactive Radionuclides in Waste Tanks 18 and 19.” SRR-CWDA-2011-00091, Rev. 0 Aiken, South Carolina: Savannah River Remediation LLC. 2011.

[SRR-CWDA-2011-00117, Rev. 0] Pavletich, J.P. “Liquid Waste Tank Residuals Sampling-Quality Assurance Program Plan.” SRR-CWDA-2011-00117, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-CWDA-2011-00126, Rev. 0] Marusich, C.M. “Tank 16 History of Waste Removal 1959 Through 2010.” SRR-CWDA-20122-00126, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2011.

[SRR-CWDA-2012-00011, Rev. 0] Hommel, S. “Features, Events, and Processes for Liquid Waste Performance Assessments.” SRR-CWDA-2012-00011, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-CWDA-2012-00044, Rev. 1] Hommel, S. “Evaluation of Features, Events, and Processes in the H-Area Tank Farm Performance Assessment.” SRR-CWDA-2012-00044, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-CWDA-2012-00071, Rev. 0] “Industrial Wastewater Closure Module for the Liquid Waste Tanks 5F and 6F F-Area Tank Farm, Savannah River Site.” SRR-CWDA-2012-00071, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2013.

[SRR-CWDA-2012-00080, Rev. 0] “Barrier Analysis Report from the Performance Assessment for the H-Area Tank Farm at the Savannah River Site.” SRR-CWDA-2012-00080, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-CWDA-2012-00109, Rev. 0] Sheppard, R.E. “Determination of HTF Site Specific Factors for Waste Classification.” SRR-CWDA-2012-00109, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-CWDA-2013-00049, Rev. 1] “Savannah River Site Liquid Waste Facilities Performance Assessment Maintenance Program-FY2013 Implementation Plan.” SRR-CWDA-2013-00049, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2013.

[SRR-CWDA-2013-00086, Rev. 0] “Follow-Up in Support of U.S. Nuclear Regulatory Commission HTF Consultation Clarifying Phone Calls.” SRR-CWDA-2013-00086, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2013.

[SRR-CWDA-2013-00106, Rev. 1] "Comment Response Matrix for NRC Staff RAI on the Draft Basis Section 3116 Determination and Revision 1 Associated Performance Assessment for HTF at SRS." SRR-CWDA-2013-00106, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2013.

[SRR-CWDA-2013-00144, Rev. 0] "Comment Response Matrix for NRC Staff Clarification Questions on DOE Responses to Request for Additional Information on the Draft Basis for Section 3116 Determination and Associated Performance Assessment for the H-Area Tank Farm at the Savannah River Site." SRR-CWDA-2013-00144, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2013.

[SRR-ESH-2013-00078] Allen, P.M. "SRR Annual Radioactive Waste Tank Inspection Program-CY2012," Memorandum to J.P. deBessonet and R.H. Pope. SRR-ESH-2013-00078. Aiken, South Carolina: Savannah River Remediation LLC. June 27, 2013.

[SRR-LWE-2009-00014, Rev. 0] Colleran, H.Q. "Information on the Radiological and Chemical Characterization of the Savannah River Site Tank Waste." SRR-LWE-2009-0014, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. August 31, 2009.

[SRR-LWE-2010-00240, Rev. 1] "Tank Mapping Methodology." SRR-LWE-2010-00240, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-LWE-2011-00201, Rev. 0]. Pasala, N.R. "Information on the Radiological and Chemical Characterization of the Savannah River Site Tank Waste." SRR-LWE-2011-00201, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2011.

[SRR-LWE-2012-00039, Rev. 0] Clark, D.J. and Clark, J.L. "Estimation of Waste Material in the Tank 16 Annulus Following Sampling in November 2011." Memorandum to G.C. Arthur. SRR-LWE-2012-00039, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-LWE-2012-00082] Tisler, A. "CY2011 Annual SCDHEC Technology Briefing." Presentation to South Carolina Department of Health and Environmental Control. SRR-LWE-2012-00082. Aiken, South Carolina: Savannah River Remediation LLC. April 30, 2012.

[SRR-LWE-2013-00057, Rev. 0] Pavletich, J.P. "Tank 16 Sampling and Analysis Plan." SRR-LWE-2013-00057, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. 2013.

[SRR-LWE-2013-00077] Tisler, A. "Annual SCDHEC Technology Briefing." Presentation to South Carolina Department of Health and Environmental Control. SRR-LWE-2013-00077. Aiken, South Carolina: Savannah River Remediation LLC. April 17, 2013.

[SRR-LWP-2009-00001, Rev. 17] Chew, D.P. and Hamm, B.A. "Liquid Waste System Plan." SRR-LWP-2009-00001, Rev. 17. Aiken, South Carolina: Savannah River Remediation LLC. 2012.

[SRR-STI-2010-00283] Waltz, R.S. and West, W.R. "Annual Radioactive Waste Tank Inspection Program-2009." SRR-STI-2010-00283. Aiken, South Carolina: Savannah River Remediation LLC. 2010.

[SRR-STI-2013-00198] Subramanian, K. and Vitali, J. "Tank 12 Bulk Oxalic Acid Cleaning Flowsheet Strategy," Presentation to SCDHEC. SRR-STI-2013-00198. Aiken, South Carolina: Savannah River Remediation LLC. 2013.

[SRS-REG-2007-00002, Rev. 1] "Performance Assessment for the F-Tank Farm at the Savannah River Site." SRS-REG-2007-00002, Rev. 1. Aiken, South Carolina: Savannah River Site. 2010.

[T-CLC-F-00421, Rev. 0] Carey, S. "Structural Assessment of F-Area Tank Farm After Final Closure." Calculation No. T-CLC-F-00421, Rev. 0. Aiken, South Carolina: Savannah River Site. December 2007.

[Turner et al., 2002] Turner, D.R., Bertetti, F.P., and Pabalan, R.T. "Role of Radionuclide Sorption in High-Level Waste Performance Assessment: Approaches for the Abstraction of Detailed Models." *Soil Science Society of America Special Publication No. 59*. Madison, Wisconsin: Soil Science Society of America, Inc. 2002.

[U.S. Army Corps of Engineers,1952] "Geologic Engineering Investigations." Vicksburg, Mississippi: U.S. Army Corps of Engineers, Waterways Experiment Station. 1952.

[U-ESR-H-00107, Rev. 0] Clark, D.J. "Tank 16H: Preliminary Evaluation of Cessation of Annulus Waste Removal Activities." U-ESR-H-00107, Rev. 0. Aiken, South Carolina: Savannah River Remediation LLC. March 7, 2013.

[USGS, 2011]. U.S. Geological Survey (USGS). "Earthquake Hazards Program: Historic Earthquakes." <http://earthquake.usgs.gov/earthquakes/states/events/1886_09_01_iso.php>. 2011.

[USGS, 2013]. U.S. Geological Survey (USGS). "M2.8–9km E of Lincolnton, Georgia (BETA)." <http://comcat.cr.usgs.gov/earthquakes/eventpage/se042613a#summary>. 2013.

[USGS, 2014a] U.S. Geological Survey. "M4.1 – Georgia, USA." <http://earthquake.usgs.gov/earthquakes/dyfi/events/se/022814c/us/index.html>. 2014.

[USGS, 2014b] U.S. Geological Survey. "M3.0 – Georgia, USA." <http://earthquake.usgs.gov/earthquakes/dyfi/events/us/c000mrls/us/index.html>. 2014.

[V-ESR-G-00003, Rev. 1] Caldwell, T.B. "Waste Removal Technology Baseline: Description of the Technology Development." V-ESR-G-00003, Rev. 1. Aiken, South Carolina: Savannah River Remediation LLC. 2011.

[WSRC-IM-2004-00008, Rev. 1] "DSA Support Document—Site Characteristics and Program Descriptions (Formerly the Generic Safety Analysis Report): Chapter 1—Site Characteristics." WSRC-IM-2004-00008. Rev. 1. Aiken, South Carolina: Washington Savannah River Company. June 2007.

[WSRC-OS-94-42] "Federal Facility Agreement for the Savannah River Site." WSRC-OS-94-42. Aiken, South Carolina: Westinghouse Savannah River Company. August 16, 1993.

[WSRC-RP-93-00287] Thayer, P.A., Smits, A.D., Harris, M.K., Amidon, M.B., and Lewis, C.M. "Final Report: Stratigraphic Maps of the General Separations Area (GSA), Savannah River Site (SRS), Aiken, South Carolina." WSRC-RP-93-287. Aiken, South Carolina: Westinghouse Savannah River Company. 1993.

[WSRC-RP-94-54] Thayer, P.A., Smits, A.D., Parker, W.H., Conner, K.R., Harris, M.K., and Amidon, M.B. "Petrographic Analysis of Mixed Carbonate-Clastic Hydrostratigraphic Units in the General Separations Area (GSA), Savannah River Site (SRS), Aiken, South Carolina." WSRC-RP-94-54. Aiken, South Carolina: Westinghouse Savannah River Company. 1993.

[WSRC-RP-92-450] Strom, R.N. and Kaback, D.S. "Groundwater Geochemistry of the Savannah River Site and Vicinity." WSRC-RP-92-450. Aiken, South Carolina: Westinghouse Savannah River Company. 1992.

[WSRC-RP-2001-00024, Rev. 0] "HLW Tank 18 Waste Removal Systems Engineering Evaluation." WSRC-RP-2001-00024, Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. 2001.

[WSRC-SQP-A-00028, Rev. 0] Collard, L.B. "Software Quality Assurance Plan for the PORFLOW™ Code." WSRC-SQP-A-00028. Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. 2002.

[WSRC-STI-2006-00198, Rev. 0] Phifer, M.A., Millings, M.R., and Flach, G.P. "Hydraulic Property Data Package for the E-Area and Z-Area Soils, Cementitious Materials, and Waste Zones." WSRC-STI-2006-00198, Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. 2006.

[WSRC-STI-2007-00061, Rev. 2] Subramanian, K.H. "Life Estimation of High-Level Waste Tank Steel for F-Tank Farm Closure Performance Assessment." WSRC-STI-2007-00061, Rev. 2. Aiken, South Carolina: Westinghouse Savannah River Company. 2008.

[WSRC-STI-2007-00150, Rev. 0] Aleman, S.E. "PORFLOW™ Testing and Verification Document." WSRC-STI-2007-00150, Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. 2007.

[WSRC-STI-2007-00184, Rev. 2] Phifer, M.A., Jones, W., Nelson, E.A., Denham, M.E., Lewis, M.R., and Shine, E.P. "FTF Closure Cap Concept and Infiltration Estimates." WSRC-STI-2007-00184, Rev. 2. Aiken, South Carolina: Westinghouse Savannah River Company. October 2007.

[WSRC-STI-2007-00192, Rev. 1] Hay, M.S., Crapse, K.P, Fink, S.D., and Pareizs, J.M. "Characterization and Actual Waste Tests With Tank 5F Samples." WSRC-STI-2007-00192, Rev. 1. Aiken, South Carolina: Savannah River National Laboratory. 2007.

[WSRC-STI-2007-00369, Rev. 0] Dixon, K. and Phifer, M. "Hydraulic and Physical Properties of Tank Grouts and Base Mat Surrogate Concrete for FTF Closure." WSRC-STI-2007-00369, Rev 0. Aiken, South Carolina: Westinghouse Savannah River Company. 2007.

[WSRC-STI-2007-00460] Subramanian, K.H. "Life Estimation of Transfer Lines for Tank Farm Closure Performance Assessment." WSRC-STI-2007-00460. Aiken, South Carolina: Westinghouse Savannah River Company. 2007.

[WSRC-STI-2008-00203, Rev. 0] Hay, M.S. "Characterization of Samples from Tank 16H Annulus." WSRC-STI-2008-00203, Rev. 0. Aiken, South Carolina: Savannah River National Laboratory. 2008.

[WSRC-STI-2008-00285] Grogan, K.P., Fjeld, R.A., Kaplan, D.I., Shine, G.P., DeVol, T.A., Coates, J. and Seaman, J. "Distribution of the Sorption Coefficients (K_dsValues) in the SRS Subsurface Environment." WSRC-STI-2008-00285. Aiken, South Carolina: Washington Savannah River Company. 2008.

[WSRC-STI-2009-00352] Waltz, R.S. and West, W.R. "Annual Radioactive Waste Tank Inspection Program-2008." WSRC-STI-2009-00352. Aiken, South Carolina: Washingon Savannah River Company. 2009.

[WSRC-TR-92-42-004] Skinner, K. "Integrated Geologic Analysis of the K-Reactor Area." WSRC-TR-92-42-004. Aiken, South Carolina: Westinghouse Savannah River Company. January 1992.

[WSRC-TR-93-761] Wiersma, B.J. and Shurrab, M.S. "A Visual Assessment of the Concrete Vaults Which Surround Underground Waste Storage Tanks (U).´ WSRC-TR-93-761. Aiken, South Carolina: Westinghouse Savannah River Company. 1993.

[WSRC-TR-94-0369, Rev. 1] Cumbest, R. "In-Tank Processing Geotechnical Summary Report." WSRC-TR-94-0369, Rev. 1. Aiken, South Carolina: Westinghouse Savannah River Company. 1994.

[WSRC-TR-95-0046] Denham, M.E. "SRS Geology/Hydrogeology Environmental Information Document." WSRC-TR-95-0046. Aiken, South Carolina: Westinghouse Savannah River Company. 1999.

[WSRC-TR-96-0037] Flach, G.P., Hamm, L.L., Harris, M.K., Thayer, P.A., Haselow, J.S. and Smits, A.D. "Groundwater Flow and Tritium Migration From the SRS Old Burial Ground to Fourmile Branch." WSRC-TR-96-0037. Aiken, South Carolina: Westinghouse Savannah River Company. 1996.

[WSRC-TR-96-0399, Vol. 1, Rev. 1] Flach, G.P. and Harris, M.K. "Integrated Hydrogeological Modeling of the General Separations Area, Vol. 1: Hydrogeologic Framework." WSRC-TR-96-0399. Rev. 1. Aiken, South Carolina: Westinghouse Savannah River Company. 1999.

[WSRC-TR-96-0399, Vol. 2, Rev. 1] Flach, G.P. and Harris, M.K. "Integrated Hydrogeological Modeling of the GSA, Vol. 2: Groundwater Flow Model." WSRC-TR-96-0399. Rev. 1. Aiken, South Carolina: Westinghouse Savannah River Company. 1999.

[WSRC-TR-99-00369] Chen, K.F. "Flood Hazard Recurrence Frequencies for C-, F-, E-, H-, Y-, and Z-Areas." WSRC-TR-99-00369. Aiken, South Carolina: Westinghouse Savannah River Company. 1999.

[WSRC-TR-99-4083, Rev. 0] Aadland, R.K., Cumbest, R.J., Whaley, K., Gelinas, R.L., Parker, W., Syms, F.H., and Rothammer, C. "Significance of Soft Zone Sediments at the SRS: Historical Review of Significant Investigations and Current Understanding of Soft Zone Origin, Extent and Stability." WSRC-TR-99-4083, Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. 1999.

[WSRC-TR-2003-00250, Rev. 0] Hiergesell, R.A. and Jones, W.E. "An Updated Regional Water Table of the Savannah River Site and Related Coverages." WSRC-TR-2003-00250, Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. 2003.

[WSRC-TR-2003-00432, Rev. 0] Flach, G.P. and Millings, M.R. "Unreviewed Disposal Question: Evaluation of Unanalyzed Silt and Engineered Trench Locations." WSRC-TR-2003-00432, Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. 2003.

[WSRC-TR-2004-00106, Rev. 0] Flach, G.P. "Groundwater Flow Model of the General Separations Area Using PORFLOW™." WSRC-TR-2004-00106, Rev. 0. Aiken, South Carolina: Westinghouse Savannah River Company. 2004.

[WSRC-TR-2005-00201] Wike, L.D., Martin, F.D., Nelson, E.A., Halverson, N.V., Mayer, J.J., Paller, M.H., Riley, R.S., Serrato, M.G., and Specht, W.L.. "SRS Ecology Environmental Information Document." WSRC-TR-2005-00201. Aiken, South Carolina: Westinghouse Savannah River Company. 2006.