SRR-CWDA-2013-00106
Revision 1

Comment Response Matrix
for
NRC Staff 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

October 2013

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Prepared for U.S. Department of Energy Under Contract No. DE-AC09-09SR22505

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ACRONYMS

BWRE	Bulk Waste Removal Efforts
CC	Clarifying Comments
CSH	Calcium Silicate Hydrate
DOE	U.S. Department of Energy
DWPF	Defense Waste Processing Facility
ECC	Enhanced Chemical Cleaning
EPA	U.S. Environmental Protection Agency
ERDMS	Environmental Restoration Data Management System
FEP	Features, Events, and Processes
FTF	F-Tank Farm
FTF PA	F-Tank Farm Performance Assessment[1]
GSA	General Separations Area
HRR	Highly Radioactive Radionuclide
HTF	H-Tank Farm
IHI	Inadvertent Human Intruder
MOP	Member of the Public
NDAA	Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005
NRC	U.S. Nuclear Regulatory Commission
OA	Oxalic Acid
PA	Performance Assessment
RAI	Request for Additional Information
SA	Special Analysis
SCDHEC	South Carolina Department of Health and Environmental Control
SLP	Standard Slurry Pump
SMP	Submersible Mixer Pump
SRR	Savannah River Remediation LLC
SRS	Savannah River Site
TCCZ	Tan Clay Confining Zone
TER	Technical Evaluation Report
UTR	Upper Three Runs
UTRA-LZ	Upper Three Runs Aquifer-Lower Zone
UTRA-UZ	Upper Three Runs Aquifer-Upper Zone
WCR	Water-Cement Ratio
WCS	Waste Characterization System

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[1] Throughout the responses provided in this document the *Performance Assessment for the F-Tank Farm at the Savannah River Site*, SRS-REG-2007-00002, is referred to as the FTF PA.

Executive Summary

In accordance with the *Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005* (NDAA), Section 3116, certain waste from reprocessing of spent nuclear fuel is not high-level waste if the Secretary of Energy, in consultation with the U.S. Nuclear Regulatory Commission (NRC), determines that the criteria in NDAA Section 3116(a) are met. On February 6, 2013, the U.S. Department of Energy (DOE) submitted the *Draft Basis for Section 3116 Determination for Closure of H-Tank Farm at the Savannah River Site*, DOE/SRS-WD-2013-001 (hereinafter referred to as: Draft HTF 3116 Basis Document) to support the consultation process for the stabilized residuals in waste tanks and ancillary structures, those waste tanks, and the ancillary structures (including integral equipment) at the Savannah River Site (SRS) H-Tank Farm (HTF) at the time of closure. [ML13044A309]

Prior to submittal of the Draft HTF 3116 Basis Document, DOE interacted with the NRC beginning in 2010 in development of the *Performance Assessment for the H-Tank Farm at the Savannah River Site*, SRR-CWDA-2010-00128 (hereinafter referred to as: HTF PA), the major technical reference document supporting the conclusions contained within the Draft HTF 3116 Basis Document. The interactions included extensive discussion (i.e., scoping meetings) between DOE and NRC on the fundamental technical bases, approaches, and key parameter values prior to development of the HTF PA. [ML100970781]

To support the NRC consultative role, after issuance of the Draft HTF 3116 Basis Document, NRC and DOE engaged in a series of technical exchanges and public meetings to clarify the approaches and rationales documented in the Draft HTF 3116 Basis Document and HTF PA. These clarifications were intended to provide NRC staff an improved understanding of the approaches and supporting technical bases developed by DOE. [ML13086A080, ML13106A338, ML13120A496, ML13154A327, ML13193A072, ML13199A413, ML13183A410] NRC staff comments on both the Draft HTF 3116 Basis Document and the HTF PA in the form of requests for additional information (RAI) or clarifying comments (CC) were provided to DOE on July 31, 2013. [ML13196A135] On August 29, 2013, NRC and DOE held a joint public meeting in Aiken, South Carolina to discuss and clarify the intent of the NRC comments and RAIs. [ML13218A556, ML13246A133]

DOE responses to the NRC Staff comments, RAIs and CCs, are provided in this response document. DOE submits these responses to facilitate NRC's completion of a Technical Evaluation Report (TER) for consultation regarding HTF at SRS.

Maximum Extent Practical (MEP)

RAI-MEP-1

Given that oxalic acid cleaning has significant downstream impacts, DOE should clarify its limitations as part of the technology baseline.

Basis

Section 5.3 of V-ESR-G-00003, Rev. 1 describes the chemical cleaning process using oxalic acid as part of the technology baseline. One of the challenges of working with oxalic acid is the formation of oxalates, which have downstream impacts. SRR-STI-2010-00015, Rev. 0 describes that for every tank that undergoes chemical cleaning about 51,000 kg of new sodium oxalates solids will be created for feed to Defense Waste Processing Facility (DWPF) as well as 1900 m^3 (500,000 gal) of salt waste. DOE describes the following oxalic acid impacts: additional wash cycles for DWPF feed, increased likelihood of feed breaks to DWPF, extension of the operating life of the entire Liquid Waste System, and evaporator foaming and scaling problems. WSRC-TR-2004-00317, Rev. 0 discusses potential limits on the use of oxalic acid due to downstream impacts on the liquid waste system. Specifically, the "sludge batch can contain about 10 % (by weight) of total solids as sodium oxalate before increasing the number of canisters produced or changing sludge processing", and "10 % (by weight) sodium oxalate in total solids amounts to disposal of 1 to 6 sludge heels depending on waste type of sludge heel cleaned and specific sludge batch."

DOE had been pursuing enhanced chemical cleaning technology which would have destroyed or oxidized the oxalates before introduction to the destination waste tank through a separate oxidation process. However, DOE stated in the technical exchange teleconference with the NRC staff on May 16, 2013 that this technology was not currently being funded, primarily due to nuclear safety concerns (ML13193A072).

In the teleconference, DOE referred NRC staff to Appendix B of the draft Basis for Waste Determination, which describes the process for documenting the removal to the maximum extent practical and includes how DOE plans to characterize the liquid waste system status as part of the technology evaluation. Appendix B states that DOE will consider storage space, compatibility of the waste, downstream impacts, status of the salt waste, impact on future waste streams, and available equipment when characterizing the liquid waste system.

Given that the use of oxalic acid has potential negative impacts in many of these areas; DOE has stated that its use will be carefully controlled. It would be useful at this point in time to assess the practicality of oxalic acid remaining as part of the technology baseline evaluating the considerations listed in Appendix B of the draft Basis for Waste Determination.

Path Forward

Please clarify how limitations surrounding oxalates 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. If oxalic acid is not likely to remain as part of the technology baseline, please clarify other technologies DOE may be pursuing as an alternative given that enhanced chemical cleaning is no longer being pursued.

Response RAI-MEP-1

To date, DOE has performed OA cleaning on FTF Tanks 5 and 6, and HTF Tanks 12, 16, and 24. Tank 16 was the first waste tank to undergo OA cleaning, which took place in 1979 as a demonstration project. This first-of-a-kind application of OA cleaning reduced the total estimated solids in the Tank 16 primary tank, from 5,250 gallons to 3,700 gallons. Subsequent cleaning with water washes removed additional solids and reduced the volume to an estimated 300 gallons. [SRR-CWDA-2013-00041] Because only limited sampling and analyses have been performed for actual radionuclide concentrations, the effectiveness of OA cleaning for HRR removal in Tank 16 is difficult to determine at this time. Final characterization of the Tank 16 residuals, as well as a final volume determination, is in progress. When the final Tank 16 primary tank residual inventory is determined, DOE will be able to better evaluate the effectiveness of OA cleaning on HRRs in Tank 16.

The second application of OA cleaning took place in 1985 on heel material in HTF Tank 24. Tank 24 was cleaned using an OA flowsheet similar to the one used on Tank 16; however, removal effectiveness was less than expected because Tank 24 contained spent zeolite resins from the cesium removal column operations within the waste tank. The removal effectiveness of the spent zeolite resin was much lower than expected due to chemical changes in the resins during the time spent in the high caustic environment. [DPST-85-782] Therefore, the effectiveness and use of OA to clean HTF waste tanks containing zeolite (i.e., Tanks 24, 32, 38, 40, 42, and 51) may be limited.

The next two waste tanks to undergo chemical cleaning with OA were FTF Tanks 5 and 6. The OA cleaning in Tank 5 reduced the residual solids volume in the primary tank from approximately 3,500 gallons to 3,300 gallons, and in the Tank 6 primary tank from approximately 6,000 gallons to 3,500 gallons. However, a significant portion of those remaining solids was non-radioactive oxalate compounds that formed during the chemical cleaning process. Additional solids were removed from Tanks 5 and 6 during subsequent mechanical feed and bleed removal campaigns. [SRR-CWDA-2012-00071] Based on limited sampling performed during the OA cleaning campaigns, OA has proven to be relatively effective at preferentially removing uranium isotopes, not very effective at removal of plutonium and americium isotopes, and has varying degrees of effectiveness on removal of other HRRs. [SRNL-STI-2009-00492, SRNL-STI-2009-00493]

In June and July 2013, HTF Tank 12 became the most recent waste tank to undergo OA cleaning campaigns. In an effort to reduce oxalate formation, DOE utilized lessons learned from the Tank 5 and 6 chemical cleaning campaigns to develop the Tank 12 OA cleaning flowsheet. The Tank 12 flowsheet was developed to maximize contact of residuals with OA, provide adequate mixing, control pH to prevent oxalate precipitation, and included an extensive pre-wash treatment regime to target sodium oxalate solubility. [SRR-STI-2013-00198] In order to gain additional understanding of the HRR removal effectiveness of OA cleaning, process sampling was performed prior to the start of OA cleaning as well as during each of the acid addition campaigns. Many of the laboratory analyses are still ongoing. DOE is currently evaluating the impact of the Tank 12 heel removal operations to date to determine whether to perform additional cleaning or proceed towards discussions with SCDHEC and EPA on ceasing waste removal activities and entering the sampling and analysis phase. At this time, because analysis of the Tank 12 process samples has not been completed and the final residual volume and radionuclide concentrations have not been determined, the impact of the revised flowsheet on the reduction of oxalate formation is not known.

As described in the NRC *Basis* for this RAI, DOE is concerned with the downstream impacts to the Liquid Waste System of oxalate formation during OA cleaning as the material is removed from the waste tank being cleaned and placed into other waste tanks. Furthermore, DOE has commissioned a study which will evaluate OA cleaning against the downstream impacts and will use this study to gain information on the impacts of OA cleaning on the Liquid Waste System versus the benefits gained by utilizing the process. DOE intends to use information gained from the final residuals inventory for Tank 16, the sample analyses from the recent Tank 12 chemical cleaning campaigns, data from previous chemical cleaning campaigns in Tanks 16, 24, 5 and 6, and existing research information to evaluate the overall effectiveness of OA cleaning. The intent of this evaluation is to determine the role of the existing OA cleaning process, as well as the Enhanced Chemical Cleaning process, in future heel removal efforts for both HTF and FTF.

RAI-MEP-2

DOE should update the process or strategy for considering developments in waste tank cleaning technologies that occur after the waste determination process has been completed.

Basis

During the FTF consultation, NRC staff requested that DOE document its strategy for considering and selecting technologies in addition to what had been previously documented in the 2003 Systems Engineering Evaluation (G-ESR-G-00051, Rev. 0). In response to the NRC staff comments, DOE provided V-ESR-G-00003, Rev. 1, which provided a description of the waste removal technology selection process, the baseline technologies, and consideration of future technologies used to support DOE's demonstration that NDAA Criterion 2 would be met. Given recent DOE technology decisions (e.g., no longer funding enhanced chemical cleaning), DOE should update their documentation of the technology baseline. Also, although emphasis on highly radioactive radionuclide removal is discussed in V-ESR-G-00003, Rev. 1, recent technology applications and selections do not appear to target highly radioactive radionuclides. For example, during oxalic acid cleaning for Tank 5F less than 10% of the plutonium and americium isotopes were removed (SRNL-STI-2009-00492, Rev. 0). A new technology presented in the draft Basis for Waste Determination is low temperature aluminum dissolution which was used in Tank 12H after bulk mechanical removal and prior to oxalic acid cleaning. Low temperature aluminum dissolution dissolves only aluminum and results showed minimal leaching of other metals; therefore, this technology does not target removal of highly radioactive radionuclides directly (X-CLC-H-00921, Rev. 0). DOE indicated in the June 5, 2013 meeting that low temperature aluminum dissolution was never intended to target highly radioactive radionuclides directly, but instead was meant to change the rheology of the waste in order to facilitate future removal of highly radioactive radionuclides (ML13183A410).

Path Forward

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. Include a clear description of whether or not the technology is intended to remove highly radioactive radionuclides and how it accomplishes their removal if intended to do so. DOE should also update the status of several technologies in its documentation of the technology baseline that were discussed in SRR-LWE-2013-00077 (e.g., the low-volume pump, robotic arm). Finally, DOE should indicate how more recent information is considered in the technology selection process and the potential for technologies to target highly radioactive radionuclide removal.

Response RAI-MEP-2

As discussed in the Draft HTF 3116 Basis Document, DOE is utilizing mechanical, chemical, and vacuum heel removal technologies at SRS in various combinations and sequences depending on the unique characteristics of the waste and the conditions in each waste tank. Information on the process for evaluating and selecting available waste removal technologies for each waste tank is provided in *Waste Removal Technology Baseline: Technology Development Description*, V-ESR-G-00003. This document discusses the use of mixer pumps as the baseline mechanical technology for waste tanks with cooling coils, followed by OA cleaning as a chemical means of waste tank cleaning. For waste tanks without cooling coils, the Mantis technology, which was utilized in the cleaning of FTF Tank 18 and Tank 19, is currently considered the baseline technology.

As discussed in both the Draft HTF 3116 Basis Document and the technology baseline document, DOE continues to evaluate emerging waste tank cleaning technologies for applicability in the SRS waste tanks. As required by *Industrial Wastewater General Closure Plan for H-Area Waste Tank System*, SRR-CWDA-2011-00022, DOE provides annual updates on new waste removal and characterization technologies to SCDHEC. The updates include sharing of information and lessons learned from SRS, other DOE sites, and other published reports as part of the DOE technology development program. Since issuance of *Waste Removal Technology Baseline: Technology Development Description*, V-ESR-G-00003, 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. In the area of new mixing technologies, DOE continues to evaluate alternative mixers with the desire to be able to carry out waste removal on a given waste tank utilizing one mixer pump design that is efficient for all phases of waste removal. DOE is evaluating a more economical pump design, when compared to the current SMP and SLP designs that would have similar waste removal performance to the SMPs or SLPs. DOE has also considered applicability of low-level mixing pumps as discussed in the response to RAI-MEP-6 and is following the development of mixing models as discussed in the response to RAI-MEP-3. To date, the SMP and SLP designs are still considered state of the art and continue to be utilized in cleaning of the SRS waste tanks.

Relative to waste removal technologies applicable to waste tanks without cooling coils, DOE continues to monitor developments of the MARS hydraulic arm being used for waste removal at the Hanford site, for potential deployment at SRS. The MARS hydraulic arm is mounted in a riser in the tank top and can position a water sluicer or vacuum device throughout the tank. The MARS technology could only be used in the SRS Type IV tanks because of cooling coil interference in the Type I, II, III, and IIIA tank designs. Currently, waste removal in the remaining HTF Type IV tanks is not planned to occur until approximately 2022.

As discussed in the response to RAI-MEP-1, DOE has utilized OA cleaning in a number of waste tanks and is currently evaluating the effectiveness and future of chemical cleaning with OA. At the time *Waste Removal Technology Baseline: Technology Development Description*, V-ESR-G-00003, was issued, DOE was actively pursuing ECC, a new OA cleaning technology aimed at reducing the impacts of oxalate formation during OA cleaning by destroying or oxidizing the oxalates before they are introduced into a waste tank. As discussed in the technology baseline document, DOE's plan at the time was to utilize FTF Tank 8 as the prototype waste tank for ECC deployment. Since issuance of the technology baseline document in June 2011, due to funding constraints and uncertainty in OA performance in actual waste tanks, DOE has suspended work on the ECC project. As stated in the *Basis* section above, OA performance in FTF Tanks 5 and 6 clearly demonstrated that it is not effective in solubilizing many of the HRRs; its effect on the rheology of the material and improvement in the associated mechanical removal of these insoluble solids is still being studied. As discussed in the response to RAI-MEP-1, DOE is evaluating the future use of OA chemical cleaning, including the use of the ECC technology.

Another chemical cleaning technology discussed in the Draft HTF 3116 Basis Document is Low Temperature Aluminum Dissolution, which DOE has deployed 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 water washing, subsequently 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. The benefit of Low Temperature Aluminum Dissolution in regards to HRR removal is that it appears to change rheology of the waste and therefore allow for more efficient waste removal with the mixer pumps. Low Temperature Aluminum Dissolution was deployed on Tank 12 to lower the residual solids volume to below 10,000 gallons prior to OA chemical cleaning. [X-CLC-H-00921] Final evaluation of the impact of Low Temperature Aluminum Dissolution in Tank 12 is currently ongoing and will be documented as part of the development of the Tank 12 Closure Module and Final Removal Report.

DOE has now completed, or is in the process of performing, heel removal activities utilizing the current baseline technologies outlined in *Waste Removal Technology Baseline: Technology Development Description*, V-ESR-G-00003, in six SRS waste Tanks (i.e., Tanks 5, 6, 12, 16, 18, and 19) . As discussed in the Response to RAI-MEP-1, DOE is evaluating the effectiveness of OA chemical cleaning based on information gathered during deployment of the technology in actual waste tanks. Based on the results of the evaluation, DOE will determine what changes, if any, need to be made to the current baseline technologies and documentation regarding waste removal technologies for use at SRS will be updated accordingly.

RAI-MEP-3

DOE's approach to developing an implementation strategy for a cleaning technology including the steps and criteria used in decision-making should be documented.

Basis

NUREG-1854 states that NRC staff should evaluate DOE's selection and *application* of removal technologies. DOE has a defined a generic process for *selecting* technologies in Appendix B of the draft Basis for Waste Determination. However, this process does not describe DOE's approach for designing the *application* of a specific technology. For example, prior to implementing the cleaning technology selected for sludge removal (e.g., submersible mixing pumps or Bingham slurry pumps), DOE develops a mixing strategy (i.e., the number, location, indexing of pumps, etc.). In the prior cleaning experience, DOE has adjusted its mixing strategy for specific tanks by moving a pump to a different riser, adding a pump(s), changing the indexing, or otherwise removing obstacles (e.g., cutting cooling coils).

DOE’s process for determining an initial strategy or adjusting a particular strategy is not well documented and it is not clear that consistency is applied in the decision making surrounding the mixing strategy. For example, there seem to be inconsistencies in the decision process for mixing strategies for different tanks. Section 3.3.3 of SRR-CWDA-2011-00126, Rev. 0 states that DOE installed three Bingham slurry pumps in the Tank 16H risers for mechanical sludge removal campaigns 3-5 whereas only one pump had been used for the first two campaigns. It is not clear why DOE’s initial strategy for Tank 16H did not include three Bingham slurry pumps.

As another example, mechanical sludge removal campaigns in Tank 6F were accomplished with two submersible mixing pumps. Mounds remained under risers 1 and 5 following the mechanical sludge removal in August, 2007 due to the limited effective cleaning radius of the submersible mixing pumps. DOE has anticipated that the effective cleaning radius was 15 m (50 ft) and, thus, planned for only two submersible mixing pumps; however, the actual effective cleaning radius was less than anticipated as demonstrated in Tanks 5F and 6F. In the third Tank 5F mechanical sludge removal campaign in February, 2008, DOE added a third submersible mixing pump to facilitate waste retrieval under riser 1. It was not clear to NRC staff why a third or fourth submersible mixing pump was not also added to Tank 6F to facilitate waste retrieval during mechanical sludge removal. SRR-CWDA-2012-00071, Rev. 0 (page 94) indicates that a third submersible mixing pump was added to Tank 6F later, during mechanical feed and bleed. NRC staff inquired about the apparent inconsistencies with Tank 5F and 6F cleaning campaigns in comments on the Tanks 5F and 6F Closure Module (ML13081A051). DOE clarified that only four total submersible mixing pumps can be connected at any one time in the area where Tanks 5F and 6F are because of electrical limitations (ML13191A132). Since two submersible mixing pumps were being used in Tank 5F and two in Tank 6F, the third submersible mixing pump during mechanical sludge removal in Tank 6F could not be used while cleaning was still being conducted in Tank 5F absent significant changes in the electrical setup. While this explanation clarifies the limitations DOE encountered during cleaning Tanks 5F and 6F, it is not clear how this lesson learned regarding actual effective cleaning radius of submersible mixing pumps will be applied for future tank cleaning. It is also not clear how potential options for the transfer pumps (e.g., location, number, etc.) are considered in the planning stages to maximize solids removal.

In cases such as this, decisions to stop waste removal activities based on programmatic or schedule constraints should be supported by an evaluation of the costs and potential benefits of continuing removal operations. A technology that DOE presented in a briefing to the state of

South Carolina in April 2013 (SRR-LWE-2013-00077) may assist DOE in predicting the effectiveness or benefits of certain mixing strategies. Specifically, DOE is cooperating with Hanford on the development of mixing models that can predict different slurry behavior. DOE stated that this is still in the beginning stages, but is a growing area with potential to enhance cleaning efforts at the Savannah River Site. In Tanks 5F and 6F, DOE considered the addition of a fourth submersible mixing pump, but decided that the costs would have outweighed the benefits, noting that the benefits were unknown for the amount of residual solids remaining in Tank 5F and 6F (SRR-CWDA-2010-00157, Rev. 0; SRR-CWDA-2011-00033, Rev. 1; SRR-CWDA-2011-00005, Rev. 1). The mixing model might have aided in predicting potential benefits of an additional pump(s).

Path Forward

Describe DOE's generic approach to developing an implementation strategy for a cleaning technology, including the steps and criteria used in decision making. Include a specific example of how this generic approach is implemented with the development of the pumping/mixing strategy for a specific tank. For example, DOE should indicate how limitations in the effective cleaning radius of submersible mixing pumps identified during Tanks 5F and 6F cleaning will be considered for HTF tanks yet to be cleaned. DOE should also clarify how lessons learned from past cleaning experience will be considered in deciding the mixing strategies (types of pumps, number of pumps, location, etc.) for future cleaning. Include the process that will be followed for deciding when and how to adjust the initial strategy to make it more effective if needed (i.e., the practicality of installing additional pumps, moving pumps, cutting cooling coils, etc. would be helpful). Finally, clarify the timeline of the use of the mixing model at HTF that is being developed for Hanford.

Response RAI-MEP-3

Currently, during Bulk Waste Removal Efforts (BWRE), DOE will utilize either SMPs or SLPs of varying quantities, or in the case of waste tanks with no sludge present, may not deploy any mixer pumps. In general, the purpose of BWRE is to extract the majority of the waste, which reduces the radiological and environmental risk inherent in storing large volumes of waste material. During BWRE, the waste removal activities are designed and timed to optimize operation of the sludge and salt treatment processes.

Because of the different approaches that may be deployed to support BWRE, the type, and number of mixer pumps, which remain in a waste tank at the conclusion of BWRE will differ from waste tank to waste tank. In addition, the actual volume, chemical properties, radiological constituents, and physical properties of the waste remaining in a waste tank prior to heel removal will also vary from waste tank to waste tank. Therefore, at the initiation of heel removal activities each waste tank has a unique set of existing conditions that must be taken into consideration in establishing an optimal and tailored waste removal strategy that also considers Liquid Waste System conditions and intra-area specific logistical capabilities (e.g., electrical feed limitations).

DOE has considered use of mixer pumps (e.g., SMPs and SLPs) as the baseline mechanical technology and OA cleaning as the baseline chemical technology, for use in heel removal in the SRS waste tanks. To determine the type, quantity, and location of mixer pumps to be deployed initially for heel removal, DOE takes into consideration a number of factors including such things as:

- The type, quantity, and mechanical condition of mixer pumps (e.g., SMP or SLP), if any, already installed;
- The existing risers available for installation of additional mixer pumps or existing equipment that could be removed to support installation of mixer pumps. DOE must evaluate the impact of relocating existing equipment relative to cost, available storage location, if applicable, and potential hazards to workers associated with equipment removal (e.g., radiological dose);
- Specific cooling coil arrangement in the waste tank relative to riser location and waste location within the waste tank;
- Availability of mixer pumps, SLP or SMP, which must be balanced with other risk reduction activities, such as DWPF feed preparation, salt waste disposal feed preparation, nuclear safety considerations, and other waste tank cleaning activities;
- Current knowledge of the mixing technology (e.g., expected effective cleaning radius under deployed conditions); and
- Electrical systems available to power the pumps, including drives and/or sharing of power distribution with other activities.

Note that deployment of SLPs in Tank 16 and SMPs in Tanks 5 and 6 represent the first applications of these technologies in SRS waste tanks. DOE has been applying the lessons learned to the extent possible from these waste removal activities on subsequent waste removal efforts.

Relative to the deployment of mixing pumps in Tank 16, the state of technology at that time (1970s) must be taken in context. The deployment of slurry pumps in Tank 16 represented the initial field-testing of what was then a new pump design. Deployment of a single Bingham-Willamette pump (now referred to as Sulzer Pumps) in Tank 16 in December 1978, represented the first successful test run of a slurry mixer pump in an SRS waste tank. The single pump was operated for two removal campaigns to gain knowledge on this technology. Two additional pumps were then added totaling three pumps operating in Tank 16 for the remainder of the mechanical sludge removal campaigns. The purpose of operating three pumps was to serve as the first multi-pump test, which was being performed to extrapolate the number of pumps required to mix a waste tank adequately. From the test results it was determined that three or four pumps (i.e., SLPs) are needed. Therefore, Tank 16 initial mixer pump deployment was based on the testing strategy being deployed at the time. [V-ESR-G-00003]

The deployment of SMPs in Tank 5 and 6 was the first use of SMPs to perform heel removal activities in an SRS waste tank. Based on computational modeling and on-site full-scale mock-up testing, it was expected that two SMPs would adequately mix the waste tank; therefore, initial deployment in the waste tanks utilized two SMPs. Two SMPs were deployed in Tank 5 and two mechanical sludge removal campaigns were performed. To optimize sludge batch preparation for waste treatment operations in DWPF, the two SMPs from Tank 5 were moved to Tank 6 and a series of mechanical sludge removal campaigns were performed (i.e., eleven total campaigns) in Tank 6 until the two pumps reached a point of diminished effectiveness. Following these campaigns, the estimated solids volume in Tank 6 was below 10,000 gallons, which was the goal at that time in preparation for OA cleaning. After completion of the mechanical sludge removal campaigns in Tank 6, the campaigns in Tank 5 resumed. Based on the lessons learned from the Tank 6 campaigns regarding the limited mixing resulting from the use of two SMPs, Tank 5 removal was resumed with three SMPs installed for the remaining mechanical sludge removal campaigns. [SRR-CWDA-2012-00071]

After completion of the mechanical sludge removal campaigns in Tank 5 and Tank 6, DOE proceeded with chemical cleaning of the waste tanks. Chemical cleaning in Tanks 5 and 6 was occurring simultaneously and therefore, as noted in the *Basis* to this RAI, there was a limitation to the number of pumps per waste tank that could be operated. Electrical configuration in the area only allowed for four total SMPs to be operational without extensive rework of the electrical distribution system. Because of this, chemical cleaning in Tank 5 and Tank 6 was performed with two SMPs per waste tank. Since the primary role of mixing for the chemical cleaning campaigns was focused on surface area contact of the residual solids and the acid, and because experience in both Tanks 5 and 6 demonstrated that material was moved and mixed with two SMPs, the use of two pumps in this application had a technical basis.

At the conclusion of OA cleaning, DOE performed an alternative evaluation to identify technologies that could be deployed to remove additional material from Tank 5 and Tank 6. [SRR-CES-2009-00022] This evaluation took into consideration the lessons learned regarding SMP operation, among other things, during the mechanical sludge removal and chemical cleaning phases within the two tanks. Based on this evaluation, a decision was made to perform additional waste removal utilizing “feed-and-bleed” (i.e., continuous addition of liquid to the waste tank while simultaneously pumping out of the waste tank) with three SMPs operating. This “feed-and-bleed” mechanical technology was utilized in each waste tank until operations reached the point of diminished effectiveness. [SRR-CWDA-2012-00071] At the conclusion of “feed-and-bleed” mechanical cleaning, DOE analyzed the use of a fourth SMP and concluded that the lack of benefit associated with further removal of residuals from Tank 5 and Tank 6 did not justify the associated additional costs including, among other costs, the resulting delays in other risk-reducing activities in the Liquid Waste System. [SRR-CWDA-2012-00071]

Waste removal activities in Tank 12 provide an excellent example of DOE applying lessons learned and flexibility when determining the optimal pump deployment strategy. In Tank 12, four SLPs were utilized in all the waste removal campaigns and a new chemical cleaning technology, Low-Temperature Aluminum Dissolution, was deployed to optimize waste removal. The use of four SLPs eliminated the existence of “quiet zones” where heavier solids accumulated over time, as seen in Tanks 5 and 6, and allowed the pumps to continue mixing to lower liquid levels during transfers compared to SMPs. Further, as described in the response to RAI-MEP-2, Low-Temperature Aluminum Dissolution specifically targets aluminum compounds, a large fraction of the HTF waste, and appears to beneficially change the rheology of the residuals, increasing the overall mechanical removal efficiency of the operations. [SRR-WRC-2011-0004]

Similar to needing to evaluate the number and type of pumps on a waste tank-by-waste tank basis, the initial plan for the pump run programs (i.e., indexing angle, fixed versus oscillating) is also addressed on a waste tank-by-waste tank basis. Establishing the pump run program takes into account known waste tank conditions such as location of any waste mounds, location of cooling coils beneath risers, cooling coil “fences,” and knowledge of the waste characteristics (e.g., are the mounds believed to be previously undisturbed). Based on these types of input, an initial pump run strategy is developed. At the conclusion of each pump run campaign, an evaluation of the remaining waste (e.g., estimated volume, location, changes to waste accumulations) is performed and adjustment to the pump run strategies are made. These evaluations are done on a case-by-case basis utilizing the knowledge of the field personnel intimately involved in the waste removal process. As described in Appendix B of the Draft HTF 3116 Basis Document, an Operating Plan is developed for each waste tank undergoing waste removal. 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, OA cleaning, combination of technologies) and the methods of implementation, among other things. Each Operating Plan will describe the planned flowsheet, including mixing strategies, chemical cleaning strategies, and lessons learned from earlier waste removal efforts, as applicable. DOE has established procedures for the development of Operating Plans and implementation of the documentation approach provided in Appendix B of the Draft HTF 3116 Basis Document. [S4 Manual Procedure ENG.50, S4 Manual Procedure ADM.53] The procedures will ensure that decisions concerning implementation and optimization of chosen technologies, as well as lessons learned, are all documented in a consistent format. Utilizing the process outlined in the procedures will allow site personnel to become more consistent in the approach to implementing chosen technologies, optimizing the technology during operation, and applying lessons learned from previous waste removal activities.

Regarding the specific question on the mixing model under development at the DOE Hanford Site; DOE and its contractors at SRS continue to interface with its counterparts at the Hanford Site on a myriad of activities related to optimizing waste removal. One such Hanford Site activity is the effort to model their mixing of prep tanks for the Waste Treatment Plant as well as modeling for Hanford Site waste retrieval activities. To date, the mixing model activities do not appear to have specific application to the SRS waste tanks. SRS personnel will continue to interface with the Hanford Site, following technology development, waste retrieval successes, as well as any mixing modeling that may be applicable to SRS. DOE will provide updates on the interfaces during the annual technology briefing given to SCDHEC.

RAI-MEP-4

DOE's approach to optimization of technology through sampling and monitoring during cleaning should be documented.

Basis

Section 5.3 of the DOE/SRS-WD-2013-001, Rev. 0 states that "throughout the heel removal process, DOE continually evaluates the ongoing effectiveness of the technology being implemented and optimizes the existing technologies." DOE stated in SRR-STI-2013-00198 that there will be "Sampling and monitoring program in place to ensure operational efficiency..." during Tank 12H oxalic acid cleaning. These include visual inspections after each chemical strike and a volumetric examination of the tank wall after the third chemical strike. Visual inspections are described throughout the multiple cleaning phases of Tank 16H (SRR-CWDA-2011-00126, Rev. 0). Visual observations, transfer line radiation readings, and ratio of water additions to solids removed were used for Tank 18F and 19F (DOE/SRS-WD-2010-001, Rev. 0). During Tank 5F mechanical feed and bleed campaigns, radiological data was collected using electronic personnel dosimetry in a valve box on the transfer line between Tanks 5F and 6F transfer line readings (SRR-CWDA-2012-00071, Rev. 0). These are all examples of the types of sampling and monitoring that DOE has completed during the cleaning of specific tanks, but the general approach is not well documented. NRC staff acknowledges that each tank will be evaluated on a case-by-case basis and generic action thresholds for metrics are not practical. However, documentation of the general process including the types of sampling and monitoring as well as examples of metrics that may be used would be useful to ensure consistency in the approach for each tank.

Path Forward

Please clarify the process for determining the sampling and monitoring that will take place during future tank cleanings. Clarify what metrics are used throughout the process to determine effectiveness.

Response RAI-MEP-4

As described in Appendix B of the Draft HTF 3116 Basis Document, an Operating Plan will be developed for each waste tank undergoing waste removal. 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, OA cleaning, combination of technologies), the methods of implementation, and identify anticipated end states. The Operating Plans will also include the metrics to be used to track waste removal progress in order to determine whether the technology being implemented is continuing to be effective, can be cost effectively optimized, or has reached the point of diminished effectiveness. The specific metrics used to evaluate effectiveness are dependent on the technology undergoing implementation and the waste tank undergoing waste removal and can include, for example:

- Monitoring radiation levels on transfer lines (if supernate is not the liquid media used to slurry the waste);
- Waste removal equipment operating parameters (e.g., current drawn by a mixer pump, transfer rates);

- Monitoring density readings for a solution;
- Monitoring solids concentration being removed;
- Waste volume reduction achieved by comparing pictures, video and mapping results; and
- Effective cleaning radius of mixing devices.

Each Operating Plan will describe the planned flowsheet, including mixing strategies, chemical cleaning strategies, and lessons learned from earlier waste removal efforts, as applicable. Taking into consideration the expected flowsheet information, technology being implemented, equipment configuration (e.g., access to transfer lines, available sample ports), and lessons learned to date, site personnel will determine the metrics which can best be used to monitor, optimize, and evaluate waste removal effectiveness. As described in the *Basis* to this RAI, to date DOE has used various methods to evaluate the effectiveness of cleaning strategies being implemented and a predetermined set of specific metrics and criteria has not been established. It is anticipated that as cleaning of waste tanks progresses and more experience is gained that, as much as possible, a more standardized set of metrics can be established.

In order to help establish consistency in the approach for future cleaning of waste tanks, DOE has established procedures for the development of Operating Plans and implementation of the documentation approach provided in Appendix B of the Draft HTF 3116 Basis Document. [S4 Manual Procedure ENG.50, S4 Manual Procedure ADM.53] The procedures will ensure that decisions concerning selection of metrics, data to be collected during operations, evaluation of the data, and lessons learned, are all documented in a consistent format. Utilizing the process outlined in the procedures will allow site personnel to become more consistent in the approach to selecting and evaluating metrics necessary to make decisions regarding effectiveness of technologies during operations.

RAI-MEP-5

DOE should document its general approach for evaluating removal efficiency of cleaning technologies.

Basis

NUREG 1854, Section 3.3 advises NRC staff to verify that DOE's reported removal efficiencies are reasonably reliable. Specifically, if DOE bases its decision to terminate removal activities on declining removal efficiency, it is important to have confidence in the reported removal efficiencies of a specific technology.

As a comment on the Tanks 5F and 6F Closure Module, the NRC staff suggested that DOE should consider analyzing tank waste samples prior to chemical sludge removal to enable a more thorough and accurate analysis of key radionuclide removal (ML13081A051). DOE reported removal percentages for specific radionuclides in the Closure Module (page 53) based on liquid process samples taken during each chemical sludge removal campaign which were compared to a solid sample taken from Tank 5F during mechanical sludge removal campaigns in 2006 prior to chemical cleaning (SRR-CWDA-2012-00071, Rev. 0). Note that a solid process sample had also been taken prior to chemical cleaning between the mechanical sludge removal campaigns. The solid process sample taken during the mechanical sludge removal campaigns was taken after the second mechanical sludge campaign and before mechanical sludge removal campaigns 3-7. Given that mechanical sludge removal may not have removed all species proportionally (i.e., faster settling constituents may increase as mechanical sludge removal progresses), the sample taken may not represent the composition of the sludge directly prior to chemical cleaning. Therefore, the report utilized the liquid process samples taken during chemical sludge removal as opposed to the solid sample taken during mechanical sludge removal. The report recommended that directly prior to chemical cleaning future tanks DOE collect a sludge sample and have personnel from Savannah River National Laboratory analyze it for key contaminants. SRNL-STI-2009-00492, Rev. 0 stated this would provide a baseline for comparison, which would allow for better evaluation of the efficiency of future chemical cleaning activities. At the June 5, 2013 meeting (ML13183A410), DOE informed NRC that it was able to retrieve a sample from Tank 12H prior to chemical cleaning, because there happened to be an accumulation that was accessible directly beneath the riser. The NRC staff acknowledges that specific circumstances for each tank may not always be conducive to sampling at various stages of cleaning and that there are financial and worker dose considerations. However, it would be helpful for DOE to document general guidelines for conditions when it is appropriate to take a sample between cleaning phases. This will ensure good practices and consistency in the approach to evaluating removal efficiency for future tanks.

NRC staff also suggested in its comments on the Tanks 5F and 6F Closure Module (ML13081A051) that DOE perform a critical evaluation of the differences in oxalic acid delivery, waste agitation, waste transfer, and other factors that led to more successful cleaning of Tank 16H compared to Tank 5F and 6F. Such an evaluation could also compare the effectiveness of the upcoming Tank 12H chemical cleaning.

Path Forward

Clarify DOE's general approach to evaluating the effectiveness of previously implemented cleaning technologies. The clarification should include a specific example of how this generic approach will be implemented with the evaluation of oxalic acid effectiveness for Tank 12H. With the understanding that each tank is different, and technologies are not expected to be equally effective on all tanks, DOE should also describe how the approach compares the effectiveness for cleaning technologies between different tanks. Finally, DOE should include generic guidelines on when it is appropriate to sample the sludge prior to chemical cleaning.

Response RAI-MEP-5

In evaluating the effectiveness of cleaning technologies there are two different aspects of the residual waste material that must be considered, changes in waste volume and changes in radionuclide concentrations. To determine changes in the volume of material remaining in a waste tank, DOE has established a waste mapping methodology. This methodology is being utilized throughout waste tank cleaning activities to estimate the volume of residual waste material remaining in the waste tanks between different cleaning steps, as well as the final volume at the conclusion of waste removal activities. [SRR-LWE-2010-00240] Determination of radionuclide concentrations, for either material remaining in the waste tank, or being removed from the waste tank, requires challenging and expensive sampling and analysis of the actual waste material. As described in Appendix B of the Draft HTF 3116 Basis Document, once DOE, SCDHEC, and EPA agree to suspend waste removal activities and move into the sampling and analysis stage of the waste tank system operational closure process, DOE will develop and document a sampling plan that minimizes uncertainty through representative sampling of the residuals. As required by *Industrial Wastewater General Closure Plan for H-Area Waste Tank System*, SRR-CWDA-2011-00022, characterization of the residuals will be completed per the SRS *Liquid Waste Tank Residuals Sampling and Analysis Program Plan*, SRR-CWDA-2011-00050, and the *Liquid Waste Tank Residuals Sampling-Quality Assurance Program Plan*, SRR-CWDA-2011-00117. Final characterization, including a volume determination as well as sampling and analysis to determine radionuclide concentrations, are used to determine the final radionuclide inventory remaining in the waste tank at the conclusion of waste removal activities.

The established processes for determining final residual inventories provide a sound basis for the final inventory values to be used in determining waste tank cleaning technology effectiveness. In order to evaluate technology effectiveness overall or during various stages of waste tank cleaning activities, radionuclide inventories prior to and between cleaning stages may also need to be determined. In the waste tanks that have been cleaned to date in both FTF and HTF, DOE has approached the evaluation of cleaning effectiveness on a waste tank-by-waste tank basis. Because this is a first of a kind clean-up and each waste tank presents different challenges, DOE has not yet been able to identify a specific set of criteria for use in determining the optimal intermediate sampling program which would be applicable to all waste tanks. However, as additional experience is gained, DOE will analyze the benefits of intermediate samples and establish sampling criteria as appropriate. The cost of the sampling and associated analyses, in monetary costs, dose to workers, and risks of waste tank entries are balanced against the benefit of the information potentially gained in aiding further residual removal and associated future risk reduction.

During cleaning in FTF Tank 18 and Tank 19 the cleaning methodology used (i.e., Mantis) was not expected to preferentially remove specific radionuclides; therefore, intermediate samples of the waste were not taken for later comparison to final residual inventories for the purpose of determining cleaning effectiveness. Overall effectiveness of the Mantis technology was based on changes in residual waste volume.

In FTF Tank 5 and Tank 6, both mechanical mixer pumps and chemical cleaning with OA were utilized to clean the waste tanks. Effectiveness of the mixer pump technology was based on changes in solids volume determined by visual mapping performed prior to and between mixing stages. At the completion of each mechanical mixing campaign, intermediate mapping of the remaining residual waste was performed to determine effectiveness of the campaign. Although it is possible for mechanical mixing to result in preferential removal of radionuclides due to differences in settling velocities of particles, monitoring of overall solids removal does provide general indication of whether or not the existing waste content is continuing to be removed.

Determination of the effectiveness of OA cleaning is more complicated. It is possible that the formation of new waste tank solids (i.e., oxalates) could impact the volume of solids within the waste tank resulting in HRR removal not corresponding proportionately to volume reduction. Since HRRs may be removed without a corresponding change in volume, volume alone may not be a good indication of cleaning effectiveness during chemical cleaning. For Tank 5 and Tank 6, DOE took liquid samples (i.e., “process samples”) during each of the chemical cleaning campaigns and compared these samples to a solid sample taken previously during mechanical cleaning, to assess any projected effect chemical cleaning had on the sludge. [SRNL-STI-2009-00492, SRNL-STI-2009-00493] As discussed in the *Basis* for this RAI, the solids sample was taken early in the mechanical sludge removal campaigns, which raises some question about how well the sample represented changes only due to OA cleaning versus possible impacts from the remaining mixing campaigns performed during mechanical sludge removal.

DOE continues to make improvements in the ability to determine the effectiveness of OA cleaning in an actual waste tank. Prior to OA cleaning on Tank 12, a sludge sample was taken (from the waste tank), and is in the process of being analyzed. In addition, during Tank 12 OA cleaning, liquid samples (often referred to as “dip” samples) were taken at the completion of each OA addition campaign. [U-ESR-H-00103] The Tank 12 samples will be utilized, along with the final residual samples that will be taken at the conclusion of waste removal activities, to evaluate the overall effectiveness of OA cleaning in Tank 12.

As discussed in the response to RAI-MEP-1, DOE has commissioned a study which will evaluate OA cleaning against the downstream impacts and will use this study to gain information on the impacts of OA cleaning on the Liquid Waste System versus the benefits gained by utilizing the OA process. DOE intends to use information gained from the final residuals inventory for Tank 16, the sample analyses from the recent Tank 12 chemical cleaning campaigns, data from previous chemical cleaning campaigns in Tanks 16, 24, 5, and 6, and existing research information to evaluate the overall effectiveness of OA cleaning. The intent of the evaluation is to determine the role of OA cleaning in future heel removal efforts for both HTF and FTF.

RAI-MEP-6

Lessons learned from removal from Tanks 5F and 6F with regard to limitations due to low liquid levels should be incorporated into plans for future cleaning.

Basis

Section 2.3.2.1 of DOE/SRS-WD-2013-001, Rev. 0 states that "the SMPs [submersible mixing pumps] are required to be shut down as the liquid level approaches the elevation of the discharge nozzles to prevent waste spraying." Because the submersible mixing pumps could not be operated at lower liquid levels, ineffective mixing during acid strike 2 in Tanks 5F and 6F appears to have contributed to the formation of solids during chemical cleaning. The second chemical sludge removal campaign lasted 54 days for Tank 5F, and 46 days for Tank 6F. The third chemical sludge removal campaign for Tank 6F lasted 90 days. The long residence time of the oxalic acid may have contributed to the formation of oxalates in addition to the lack of mixing (SRR-CWDA-2012-00071, Rev. 0). DOE has indicated that a low volume mixing pump, which would be able to operate at lower liquid levels, has been evaluated to support chemical cleaning in its technology April, 2013 briefing to the state of South Carolina (SRR-LWE-2013-00077), but in the May 16, 2013 teleconference (ML13193A072) with NRC staff, DOE explained stated that the technology is not available at this time.

Path Forward

DOE should clarify the timeline of the low volume mixing pump technology and whether DOE anticipates it to be available for future cleaning of HTF tanks. DOE should also clarify the reasons for the long residence time of oxalic acid during chemical sludge removal strike 2 for Tanks 5F and 6F while there was no mixing and if the long residence time of oxalic acid was a contributing factor to the formation of oxalates. If the length of time the oxalic acid sat in the tank without mixing was a contributing factor to the buildup of oxalates, DOE should explain how similar circumstances will be managed in future cleaning efforts.

Response RAI-MEP-6

DOE is committed to sharing technologies and lessons learned for waste removal from across the United States and the world as a whole. The low volume mixing pump technology discussed during the April 2013 briefing to the state of South Carolina alluded to ongoing technology discussions between SRS personnel and personnel at the Sellafield Site in the United Kingdom. The Sellafield Site uses a technology called Pulse Jet Mixers to mix radioactive liquid waste in some of their vessels. The Pulse Jet Mixers have some notable advantages in a nuclear environment as they have no moving parts in contact with the waste and allow for mixing at relatively low tank levels. As discussed in the May 16, 2013 teleconference, the Pulse Jet Mixers are best suited for mixing in small vessels and, while effective on the smaller tanks at Sellafield, would not be effective on large diameter tanks such as the waste tanks at SRS. At this time, DOE is not pursuing development of a low-volume mixing pump for use in SRS waste tanks.

Relative to the OA cleaning flowsheet for Tanks 5 and 6, the planned in-tank residence time of OA during chemical sludge removal strike 2 was targeted at approximately seven days for each waste tank. [LWO-LWE-2007-00104] As noted in the *Basis* section of this RAI, the actual in-tank OA residence time during strike 2 was much longer than originally planned; approximately 54 days in Tank 5 and 46 days in Tank 6. [LWO-LWE-2008-00283, LWO-LWE-2008-00284] The extended OA residence time in each waste tank was the result of an upper mechanical seal failure in one of the standard long-shafted 150-hp slurry pumps in Tank 7 that delayed the Tank

5 to Tank 7 transfer and the Tank 6 to Tank 7 transfer. In Tank 7, four slurry pumps were required to be periodically operated to release hydrogen gas trapped in the slurried sludge, to ensure the bulk vapor space flammability was safely managed in accordance with Documented Safety Analysis requirements. The material in Tank 5 and Tank 6 was transferred to Tank 7 when this slurry pump was repaired and full waste tank mixing capability was restored in Tank 7.

The long residence time of OA during chemical sludge removal strike 2 in Tank 5 and Tank 6 contributed to an increase in solids volume due to the precipitation of metal oxalates as the pH increased over time. The precipitation of metal oxalates is a direct result of the pH maintained within the waste tank; as pH increases, the oxalate solubility decreases. The extended residence time of the OA within the waste tanks allowed time for the pH of the material in the waste tank to increase and led to additional oxalate precipitation. [SRR-LWE-2011-00068]

Use of OA to clean waste tanks has only been performed on a limited number of occasions and DOE continues to evaluate results, learn new lessons, and improve techniques (see response to RAI-MEP-1). For example, numerous lessons from OA cleaning experience in Tanks 5 and 6 were incorporated into the chemical sludge removal flowsheet for Tank 12, including ones specifically targeted at minimizing the precipitation of oxalates. To address the concerns with allowing the OA to remain in contact with the sludge without mixing, the liquid level in Tank 12 was increased during each of the three OA strikes to allow the mixer pumps to run at maximum speed to optimize mixing. In addition, the sludge heel in Tank 12 was extensively pre-washed prior to adding OA to reduce the sodium molarity, thereby reducing the formation of sodium oxalates. Four standard mixer pumps were run at maximum speed for longer periods to support full waste tank coverage to maximize OA-sludge contact and mobilize insoluble particles. Experience showed that increased pH (>2) reduces the solubility of metals, therefore precipitating oxalates. As a result, the Tank 12 chemical sludge removal flowsheet required that the pH be maintained less than 2 to minimize the potential for oxalate precipitation. In addition, the Tank 12 Operating Plan required the availability of the transfer receipt tank be confirmed prior to acid addition to Tank 12 to facilitate timely transfers. [U-ESR-H-00103, SRR-STI-2013-00198] The pH of the material in Tank 12 remained below 2 prior to being transferred from the waste tank for each of the three strikes. Preliminary review of video and photographs within Tank 12 following the OA campaign indicate that significant precipitation of the oxalates, as seen in Tanks 5 and 6, did not occur.

As discussed in the response to RAI-MEP-1, DOE is in the process of evaluating sample results taken during OA cleaning in Tank 12 and will be evaluating the results, along with other information, to determine the effectiveness and path forward for application of OA cleaning in future SRS waste tank cleaning operations.

RAI-MEP-7

The practicality of removing additional material from the Tank 16H annulus should be analyzed.

Basis

The 2008 samples from the Tank 16H annulus described in WSRC-STI-2008-00203, Rev. 0 showed a wide variation in the annulus material, and specified that the material inside the duct was more soluble than the material outside the duct. "The sample from outside the duct at IP-35 shows more water insoluble material than the sample from inside the duct. Interestingly, the sample from outside the duct at IP-118 contains much more water insoluble material than the sample from outside the duct at IP-35....the IP-118 sample also shows a small difference in composition from the top to the bottom of the sample. The bottom section of the sample appears to contain more water soluble material than the top based on the XRD [X-ray diffraction] data. This aspect of the sample again seems reasonable since the material at the bottom of the annulus would also be less accessible to the washing/waste removal conducted in the annulus. The samples from outside the dehumidification duct at two locations in the annulus show very different compositions and estimated solubility in water. This indicates the waste material in Tank 16H annulus may have a wide range of compositions at different locations." If some parts of the material in the annulus (and especially the duct) are soluble, it may be practical to remove additional amounts of it. The last cleaning effort was in 1977 and mixing was poor as exhibited by the variability in sample results. Prior DOE documentation assumed that it would be necessary to remove additional material from the annulus, but efforts to clean the annulus ceased because the mechanical technology that was being pursued was not mature enough to deploy (ML13183A410). Waste in the annulus and/or sand pads tends to be more risk significant, because it contains more soluble radionuclides and is located outside of the primary containment. NRC staff is concerned that the base case (and alternative cases) in the HTF Performance Assessment may underestimate annular contamination risk (see RAI-NF-12 and RAI-NF-13).

Path Forward

Please provide more detailed information on the practicality of removing additional waste from the Tank 16H annulus to the extent necessary to reduce the risk, taking into consideration the potential risk posed by preferential pathways and addressing the issues identified in RAI-NF-12 and RAI-NF-13. As a further consideration please describe the practicality of removing additional waste from the ventilation duct since it is more soluble than the annulus floor material.

Response RAI-MEP-7

Detailed documentation on the practicality of removing additional waste from the Tank 16 annulus will be provided in a Removal Report that will be issued after the completion of final residual characterization and subsequent cost-benefit analysis.

As discussed in Appendix B of the Draft HTF 3116 Basis Document, prior to ceasing waste removal on a given waste tank and proceeding into final sampling and analysis, DOE performs a qualitative analysis of the practicality of additional waste removal. If this qualitative review supports that implementation and execution of additional removal with a new technology is not practical, then DOE will document this analysis. As required by *Industrial Wastewater General Closure Plan for H-Area Waste Tank Systems*, SRR-CWDA-2011-00022, DOE will review this information with SCDHEC and the EPA. If the three agencies (DOE, SCDHEC, EPA) concur, DOE will suspend waste removal and move into final sampling and analysis phase.

Such a qualitative analysis was performed to determine if additional technologies should be deployed to remove additional waste from the Tank 16 annulus. The results of the evaluation are documented in *Tank 16H: Preliminary Evaluation of Cessation of Annulus Waste Removal Activities*, U-ESR-H-00107, which is being provided as part of this response submittal. The recommendation of the evaluation team was to cease waste removal activities in the Tank 16 annulus. The information supporting DOE's recommendation to suspend waste removal activities in Tank 16, both the primary tank and annulus, and move forward into the final sampling and analysis phase was presented by SRR to DOE, SCDHEC, and EPA in *Proposal to Cease Waste Removal Activities in Tank 16 and Enter Sampling and Analysis Phase*, SRR-CWDA-2013-00041. Based on the presentation and associated inter-agency discussions, DOE, SCDHEC, and EPA concurred to move forward into the final sampling and analysis phase for Tank 16. [DHEC_04-11-2013, EPA_04-10-2013] DOE is currently in the process of sampling Tank 16, both the primary tank and annulus, and performing final volume determinations. The sampling evolution involves the retrieval and subsequent analysis of numerous residual material samples from both inside and outside of the annulus ventilation duct. As part of this effort, final volumes are also determined utilizing the data gathered during the sample planning and collection evolutions. 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. Details of the Tank 16 sampling and characterization effort are provided in *Tank 16 Sampling and Analysis Plan*, SRR-LWE-2013-00057.

The information contained in *Tank 16H: Preliminary Evaluation of Cessation of Annulus Waste Removal Activities*, U-ESR-H-00107, and the tri-party decision to enter the sampling and analysis phase, does not represent a final decision to cease waste removal activities. Based on the determinations of final primary tank and annulus residual material volumes and concentrations, final waste tank inventories will be established. A cost-benefit analysis will be performed informed, in part, by the dose impact results and conclusions of the HTF PA with the final radionuclide inventory considered. The final residual inventories will be compared to the current HTF PA assigned inventory for Tank 16, both the primary tank and annulus, on a radionuclide-by-radionuclide and chemical-by-chemical basis. This information will be evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1.

Typically, new fate and transport modeling will be performed as part of the SA process, replacing the PA assigned inventories with the final residual characterization data. If other "new information" has been identified (e.g., updated K_d values developed through research experimentation), this new data would also be evaluated through inclusion in the SA. The results of the SA are then evaluated to determine if any of the new information impacts PA-based conclusions. While it typically would not be necessary to replicate all of the PA sensitivity and uncertainty analyses in an SA, it may be appropriate to include additional analyses to evaluate the new or unique waste tank-specific information.

As outlined in Appendix B of the Draft HTF 3116 Basis Document, 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 (grout) Tank 16. A recent example of a final removal report developed by DOE is *Documentation of Removal of Highly Radioactive Radionuclides in Waste Tanks 5 and 6, F-Area Tank Farm Savannah River Site*, SRR-CWDA-2012-00138, which supported DOE's Tier 2 Closure decision for Tanks 5 and 6 in FTF.

Performance Assessment (PA)

CC-PA-1

DOE should clarify its screening of features, events, and processes in SRR-CWDA-2012-00011, Rev. 0 to improve transparency and traceability, in particular the use of expert judgment in lieu of data collection, the membership of the FEPs screening team, and the documentation of subject matter expert's basis for judgment.

Response CC-PA-1

DOE is providing a copy of the referenced document, *Features, Events, and Processes for Liquid Waste Performance Assessments,* SRR-CWDA-2012-00011. This report provides a complete description of the FEP screening process and provides background information with respect to the membership of the FEP screening team. In addition to the report, the Excel file, SRS_LW_FEPs_Rev0.xlsx, is provided. This file contains intermediate data related to the FEP screening process, including specific worksheets that were prepared by each member of the FEP screening team during Phase I and Phase II of the FEP screening process, and thus demonstrates appropriate reliance on data to support expert judgment.

CC-PA-2

Several changes were made to biosphere parameters for HTF. It is not clear how these changes impact the relative risk of various highly radioactive radionuclides compared to earlier performance assessments. Provide groundwater pathway dose conversion factors for HTF to facilitate comparison with previous analyses.

Response CC-PA-2

In order to better evaluate how changes impact the relative risk of various HRRs compared to earlier PAs, Tables CC-PA-2.1 through CC-PA-2.4 were prepared to provide side-by-side comparisons of the FTF PA and HTF PA dose conversion factors for the HTF HRRs and other radionuclides included in the HTF and FTF PAs. These values were calculated using the GoldSim dose calculators for each PA (*FTF v3.0.gsm* and *HTF Transport Model v0.026.gsm*, respectively). For Tables CC-PA-2.1, CC-PA-2.2, and CC-PA-2.3, the values were developed using 1.0 pCi/L for each radionuclide to determine the dose conversion factors. Similarly, the values in Table CC-PA-2.4 were calculated using drill cuttings containing 1.0 curie of each radionuclide. Differences in the values (i.e., those used in the FTF PA versus those used in the HTF PA) are attributed to updates to the input data used, based upon more current or appropriate references. For example, the FTF PA used the human health exposure parameters described in *Baseline Parameter Update for Human Health Input and Transfer Factors for Radiological Performance Assessments at the Savannah River Site,* WSRC-STI-2007-00004, published in June 2008; whereas, the HTF PA used revised data from *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, published in August 2010.

Table CC-PA-2.1: Conversion Factors for Dose Ingestion Pathways

Radionuclide	**Ingestion Pathways Dose Conversion Factors (mrem/yr)/(pCi/L)**							
	Water		***Fish***		***Beef***		***Milk***	
	FTF	**HTF**	**FTF**	**HTF**	**FTF**	**HTF**	**FTF**	**HTF**
Ac-227	2.10E+00	1.51E+00	1.40E+00	1.01E+00	5.37E-03	1.68E-03	6.08E-04	2.15E-04
Ag-108m	(a)	2.87E-03	(a)	8.43E-03	(a)	2.39E-05	(a)	3.23E-05
Al-26	4.38E-03	4.38E-03	5.85E-02	5.97E-03	4.21E-05	1.83E-05	1.31E-05	6.43E-06
Am-241	2.49E-01	2.49E-01	2.00E-01	1.60E+00	6.38E-05	3.47E-04	5.42E-06	7.46E-07
Am-242m	2.50E-01	2.50E-01	2.00E-01	1.60E+00	6.39E-05	3.48E-04	5.43E-06	7.47E-07
Am-243	2.50E-01	2.50E-01	2.01E-01	1.61E+00	6.41E-05	3.49E-04	5.44E-06	7.49E-07
Bi-210m	(a)	1.87E-02	(a)	7.49E-03	(a)	4.12E-05	(a)	1.19E-04
C-14	7.25E-04	7.25E-04	5.81E-05	5.81E-05	2.32E-04	7.55E-05	1.96E-04	7.22E-05
Ca-41	(a)	2.37E-04	(a)	7.59E-05	(a)	3.43E-05	(a)	5.72E-05
Cf-249	4.38E-01	4.38E-01	2.93E-01	2.93E-01	1.12E-04	4.88E-05	9.52E-06	4.68E-06
Cf-251	(a)	4.48E-01	(a)	2.99E-01	(a)	4.99E-05	(a)	4.79E-06
Cl-36	1.16E-03	1.16E-03	1.55E-03	1.46E-03	9.32E-03	8.72E-05	1.62E-02	2.06E-04
Cm-243	1.87E-01	1.87E-01	1.50E-01	1.50E-01	4.78E-05	2.08E-05	5.42E-05	2.67E-05
Cm-244	1.50E-01	1.50E-01	1.20E-01	1.20E-01	3.82E-05	1.66E-05	4.33E-05	2.13E-05
Cm-245	2.62E-01	2.62E-01	2.10E-01	2.10E-01	6.70E-05	2.92E-05	7.59E-05	3.73E-05
Cm-246	(a)	2.62E-01	(a)	2.10E-01	(a)	2.92E-05	(a)	3.73E-05
Cm-247	2.37E-01	2.37E-01	1.90E-01	1.90E-01	6.07E-05	2.64E-05	6.87E-05	3.38E-05
Cm-248	9.60E-01	9.60E-01	7.70E-01	7.70E-01	2.46E-04	1.07E-04	2.78E-04	1.37E-04
Co-60	4.25E-03	4.25E-03	3.40E-02	8.62E-03	2.75E-04	5.32E-06	1.87E-05	3.45E-06
Cs-135	2.49E-03	2.49E-03	2.00E-01	2.00E-01	4.04E-03	1.54E-04	1.33E-03	8.24E-05
Cs-137	1.62E-02	1.62E-02	1.30E+00	1.30E+00	2.12E-02	1.00E-03	7.00E-03	5.35E-04
Eu-152	1.75E-03	1.75E-03	1.40E-03	6.06E-03	2.25E-07	9.91E-08	7.64E-07	3.79E-07
Eu-154	2.49E-03	2.49E-03	2.00E-03	8.66E-03	3.21E-07	1.41E-07	1.09E-06	5.39E-07
Eu-155	(a)	3.98E-04	(a)	1.38E-03	(a)	2.23E-08	(a)	8.54E-08
Gd-152	(a)	5.12E-02	(a)	4.10E-02	(a)	2.96E-06	(a)	1.13E-05
H-3	2.24E-05	2.24E-05	5.99E-07	5.99E-07	0.00E+00	0.00E+00	5.59E-05	2.57E-06
I-129	1.37E-01	1.37E-01	1.47E-01	1.10E-01	3.63E-02	2.56E-03	1.85E-02	5.28E-03
K-40	7.72E-03	7.72E-03	2.06E-01	6.60E-01	1.46E-03	5.13E-04	1.16E-03	4.57E-04
Lu-174	(a)	3.37E-04	(a)	2.25E-04	(a)	4.19E-06	(a)	4.92E-08
Mo-93	(a)	3.88E-03	(a)	1.97E-04	(a)	2.02E-05	(a)	5.14E-05
Nb-93m	1.50E-04	1.50E-04	1.20E-03	1.20E-03	2.81E-07	1.08E-10	7.01E-08	4.37E-10
Nb-94	2.12E-03	2.12E-03	1.70E-02	1.70E-02	4.02E-06	1.53E-09	1.00E-06	6.19E-09
Ni-59	7.85E-05	7.85E-05	2.10E-04	4.40E-05	2.65E-06	1.12E-06	1.91E-05	5.41E-07
Ni-63	1.87E-04	1.87E-04	5.00E-04	1.05E-04	6.27E-06	2.66E-06	4.52E-05	1.29E-06
Np-237	1.38E-01	1.38E-01	7.75E-02	7.75E-02	8.95E-04	3.85E-04	1.01E-05	4.93E-06
Pa-231	8.86E-01	8.86E-01	2.37E-01	2.37E-01	2.54E-03	1.10E-03	6.43E-05	3.16E-05
Pb-210	8.59E-01	2.36E+00	6.89E+00	1.58E+00	2.21E-03	4.76E-03	3.24E-03	3.28E-03
Pd-107	4.62E-05	4.62E-05	1.23E-05	1.23E-05	1.22E-06	5.21E-07	6.90E-06	3.32E-06
Pu-238	2.87E-01	2.87E-01	2.30E-01	2.30E-01	1.84E-05	8.77E-07	4.57E-06	2.04E-05
Pu-239	3.12E-01	3.12E-01	2.50E-01	2.50E-01	2.00E-05	9.54E-07	4.97E-06	2.22E-05
Pu-240	3.12E-01	3.12E-01	2.50E-01	2.50E-01	2.00E-05	9.54E-07	4.97E-06	2.22E-05
Pu-241	6.00E-03	6.00E-03	4.81E-03	4.81E-03	3.83E-07	1.83E-08	9.55E-08	4.27E-07
Pu-242	2.99E-01	2.99E-01	2.40E-01	2.40E-01	1.92E-05	9.16E-07	4.77E-06	2.13E-05
Pu-244	3.01E-01	3.01E-01	2.41E-01	2.41E-01	1.93E-05	9.20E-07	4.80E-06	2.14E-05
Ra-226	3.51E-01	3.51E-01	4.68E-01	3.74E-02	2.06E-03	1.73E-03	6.72E-03	9.80E-04
Ra-228	1.04E+00	1.04E+00	1.39E+00	1.11E-01	5.99E-03	5.00E-03	1.96E-02	2.85E-03
Rn-222	3.12E-04	(b)	0.00E+00	(b)	0.00E+00	(b)	0.00E+00	(b)
Se-79	3.61E-03	3.61E-03	1.64E-02	5.78E-01	4.26E-04	1.79E-04	2.53E-04	1.18E-04
Sm-147	(a)	6.10E-02	(a)	4.89E-02	(a)	5.57E-05	(a)	1.34E-05
Sm-151	1.22E-04	1.22E-04	9.80E-05	9.80E-05	2.51E-07	1.11E-07	5.39E-08	2.69E-08
Sn-126	6.34E-03	6.34E-03	5.08E-01	5.08E-01	3.26E-03	1.44E-03	9.22E-05	4.60E-05

Table CC-PA-2.1: Conversion Factors for Dose Ingestion Pathways (Continued)

Radionuclide	Ingestion Pathways Dose Conversion Factors (mrem/yr)/(pCi/L)							
	Water		*Fish*		*Beef*		*Milk*	
	FTF	**HTF**	**FTF**	**HTF**	**FTF**	**HTF**	**FTF**	**HTF**
Sr-90	3.84E-02	3.84E-02	6.16E-02	2.98E-03	2.62E-03	1.51E-04	2.02E-03	3.81E-04
Tc-99	7.99E-04	7.99E-04	4.27E-04	4.27E-04	3.91E-05	4.15E-05	2.58E-05	2.72E-05
Th-228	8.96E-02	(b)	2.39E-01	(b)	2.26E-05	(b)	6.41E-06	(b)
Th-229	7.64E-01	7.65E-01	2.04E+00	1.23E-01	1.96E-04	4.91E-04	5.53E-05	2.73E-05
Th-230	2.62E-01	2.62E-01	6.99E-01	4.20E-02	6.70E-05	1.68E-04	1.90E-05	9.35E-06
Th-232	2.87E-01	2.87E-01	7.66E-01	4.60E-02	7.34E-05	1.84E-04	2.08E-05	1.02E-05
U-232	4.11E-01	5.90E-01	1.10E-01	1.51E-02	7.96E-04	6.75E-04	2.40E-03	7.89E-03
U-233	6.37E-02	6.37E-02	1.70E-02	1.63E-03	1.24E-04	7.34E-05	3.73E-04	8.57E-04
U-234	6.10E-02	6.10E-02	1.63E-02	1.56E-03	1.18E-04	7.03E-05	3.57E-04	8.21E-04
U-235	5.91E-02	5.90E-02	1.58E-02	1.51E-03	1.15E-04	6.79E-05	3.46E-04	7.93E-04
U-236	5.86E-02	5.86E-02	1.57E-02	1.50E-03	1.14E-04	6.76E-05	3.43E-04	7.89E-04
U-238	6.05E-02	6.03E-02	1.62E-02	1.55E-03	1.17E-04	6.95E-05	3.54E-04	8.12E-04
Zr-93	1.37E-03	1.37E-03	1.10E-02	8.06E-04	1.62E-06	4.61E-09	1.09E-08	3.54E-08

Radionuclide	Ingestion Pathways Dose Conversion Factors (mrem/yr)/(pCi/L)							
	Vegetable		*Soil*		*Poultry*		*Egg*	
	FTF	**HTF**	**FTF**	**HTF**	**FTF**	**HTF**	**FTF**	**HTF**
Ac-227	9.97E-01	2.28E-01	2.27E-04	3.00E-03	(a)	9.44E-05	(a)	4.78E-05
Ag-108m	(a)	4.34E-04	(a)	1.24E-06	(a)	5.99E-05	(a)	1.14E-05
Al-26	2.08E-03	6.64E-04	4.75E-07	1.27E-05	(a)	0.00E+00	(a)	0.00E+00
Am-241	1.19E-01	3.78E-02	2.70E-05	7.03E-04	(a)	1.56E-05	(a)	5.94E-06
Am-242m	1.19E-01	3.78E-02	2.70E-05	6.76E-04	(a)	1.57E-05	(a)	5.95E-06
Am-243	1.19E-01	3.79E-02	2.71E-05	7.19E-04	(a)	1.57E-05	(a)	5.97E-06
Bi-210m	(a)	7.40E-03	(a)	5.38E-05	(a)	2.99E-05	(a)	6.03E-05
C-14	6.01E-04	1.47E-04	7.85E-08	3.15E-07	(a)	0.00E+00	(a)	0.00E+00
Ca-41	(a)	2.13E-04	(a)	5.22E-08	(a)	2.97E-07	(a)	2.25E-06
Cf-249	2.08E-01	6.63E-02	4.75E-05	1.23E-03	(a)	2.75E-05	(a)	1.39E-05
Cf-251	(a)	6.79E-02	(a)	1.28E-03	(a)	2.81E-05	(a)	1.42E-05
Cl-36	4.17E-02	3.46E-04	1.26E-07	5.61E-08	(a)	4.86E-07	(a)	3.33E-05
Cm-243	8.88E-02	2.83E-02	2.03E-05	4.06E-04	(a)	1.17E-05	(a)	5.94E-06
Cm-244	7.10E-02	2.26E-02	1.62E-05	2.78E-04	(a)	9.38E-06	(a)	4.75E-06
Cm-245	1.25E-01	3.97E-02	2.84E-05	7.52E-04	(a)	1.64E-05	(a)	8.32E-06
Cm-246	(a)	3.97E-02	(a)	7.51E-04	(a)	1.64E-05	(a)	8.32E-06
Cm-247	1.13E-01	3.59E-02	2.57E-05	6.81E-04	(a)	1.49E-05	(a)	7.53E-06
Cm-248	4.57E-01	1.46E-01	1.04E-04	2.76E-03	(a)	6.02E-05	(a)	3.05E-05
Co-60	2.05E-03	6.92E-04	4.60E-07	2.50E-06	(a)	4.42E-05	(a)	1.14E-06
Cs-135	6.99E-03	3.84E-04	2.70E-07	1.08E-06	(a)	7.08E-05	(a)	7.97E-06
Cs-137	3.64E-02	2.49E-03	1.76E-06	6.50E-06	(a)	4.60E-04	(a)	5.17E-05
Eu-152	8.38E-04	2.73E-04	1.89E-07	2.85E-06	(a)	3.69E-08	(a)	5.61E-10
Eu-154	1.19E-03	3.86E-04	2.70E-07	3.11E-06	(a)	5.25E-08	(a)	7.99E-10
Eu-155	(a)	6.08E-05	(a)	3.10E-07	(a)	8.34E-09	(a)	1.27E-10
Gd-152	(a)	8.25E-03	(a)	1.47E-04	(a)	1.09E-06	(a)	1.66E-08
H-3	1.60E-04	3.89E-06	2.43E-09	1.22E-10	(a)	0.00E+00	(a)	0.00E+00
I-129	6.80E-02	2.08E-02	1.49E-05	2.51E-06	(a)	1.25E-05	(a)	2.61E-03
K-40	5.81E-03	1.54E-03	8.36E-07	1.70E-06	(a)	3.59E-05	(a)	6.81E-05
Lu-174	(a)	5.05E-05	(a)	1.88E-07	(a)	0.00E+00	(a)	0.00E+00
Mo-93	(a)	1.43E-03	(a)	1.11E-05	(a)	1.09E-05	(a)	2.95E-05
Nb-93m	7.21E-05	2.26E-05	1.62E-08	8.11E-10	(a)	4.69E-10	(a)	1.19E-09
Nb-94	1.03E-03	3.21E-04	2.30E-07	1.15E-08	(a)	6.64E-09	(a)	1.68E-08
Ni-59	3.97E-05	1.23E-05	8.51E-09	2.40E-08	(a)	8.31E-10	(a)	6.32E-08
Ni-63	9.41E-05	2.94E-05	2.03E-08	5.62E-08	(a)	1.98E-09	(a)	1.50E-07

Table CC-PA-2.1: Conversion Factors for Dose Ingestion Pathways (Continued)

Radionuclide	**Ingestion Pathways Dose Conversion Factors (mrem/yr)/(pCi/L)**							
	Vegetable		***Soil***		***Poultry***		***Egg***	
	FTF	**HTF**	**FTF**	**HTF**	**FTF**	**HTF**	**FTF**	**HTF**
Np-237	6.66E-02	2.10E-02	1.50E-05	1.86E-05	(a)	8.67E-06	(a)	4.39E-06
Pa-231	4.22E-01	1.34E-01	9.60E-05	1.19E-04	(a)	5.56E-05	(a)	2.81E-05
Pb-210	4.10E-01	3.78E-01	9.31E-05	4.79E-03	(a)	2.01E-02	(a)	1.91E-02
Pd-107	2.29E-05	7.14E-06	5.00E-09	1.41E-08	(a)	1.46E-10	(a)	1.48E-09
Pu-238	1.36E-01	4.34E-02	3.11E-05	7.33E-04	(a)	8.99E-06	(a)	2.73E-06
Pu-239	1.48E-01	4.72E-02	3.38E-05	8.77E-04	(a)	9.77E-06	(a)	2.97E-06
Pu-240	1.48E-01	4.72E-02	3.38E-05	8.76E-04	(a)	9.77E-06	(a)	2.97E-06
Pu-241	2.85E-03	9.05E-04	6.50E-07	9.89E-06	(a)	1.88E-07	(a)	5.71E-08
Pu-242	1.42E-01	4.53E-02	3.24E-05	8.42E-04	(a)	9.38E-06	(a)	2.85E-06
Pu-244	1.43E-01	4.55E-02	3.26E-05	8.46E-04	(a)	9.42E-06	(a)	2.86E-06
Ra-226	1.71E-01	5.67E-02	3.80E-05	3.53E-04	(a)	1.13E-04	(a)	8.83E-04
Ra-228	4.95E-01	1.62E-01	1.13E-04	5.33E-04	(a)	3.29E-04	(a)	2.58E-03
Rn-222	3.29E-05	(b)	3.38E-08	(b)	(a)	(b)	(a)	(b)
Se-79	2.19E-03	7.16E-04	3.91E-07	1.03E-05	(a)	4.05E-04	(a)	5.08E-05
Sm-147	(a)	9.82E-03	(a)	1.75E-04	(a)	1.30E-06	(a)	1.98E-08
Sm-151	5.93E-05	1.96E-05	1.33E-08	3.20E-07	(a)	2.61E-09	(a)	3.96E-11
Sn-126	3.03E-03	9.95E-04	6.86E-07	1.85E-05	(a)	5.36E-05	(a)	5.10E-05
Sr-90	2.55E-02	6.67E-03	4.16E-06	8.10E-06	(a)	8.44E-06	(a)	1.12E-04
Tc-99	4.76E-04	5.10E-04	8.65E-08	2.49E-08	(a)	5.31E-07	(a)	4.03E-05
Th-228	4.19E-02	(b)	9.71E-06	(b)	(a)	(b)	(a)	(b)
Th-229	3.63E-01	1.16E-01	8.27E-05	2.18E-03	(a)	4.80E-05	(a)	2.43E-05
Th-230	1.25E-01	3.98E-02	2.84E-05	7.47E-04	(a)	1.64E-05	(a)	8.33E-06
Th-232	1.36E-01	4.36E-02	3.11E-05	8.19E-04	(a)	1.80E-05	(a)	9.12E-06
U-232	1.98E-01	9.73E-02	4.45E-05	1.39E-03	(a)	4.77E-03	(a)	5.31E-03
U-233	3.07E-02	1.06E-02	6.90E-06	1.69E-04	(a)	5.17E-04	(a)	5.76E-04
U-234	2.94E-02	1.02E-02	6.61E-06	1.62E-04	(a)	4.95E-04	(a)	5.52E-04
U-235	2.84E-02	9.83E-03	6.40E-06	1.56E-04	(a)	4.78E-04	(a)	5.33E-04
U-236	2.82E-02	9.78E-03	6.35E-06	1.55E-04	(a)	4.76E-04	(a)	5.30E-04
U-238	2.91E-02	1.01E-02	6.56E-06	1.60E-04	(a)	4.89E-04	(a)	5.46E-04
Zr-93	6.53E-04	2.10E-04	1.49E-07	3.92E-06	(a)	8.63E-10	(a)	2.19E-09

(a) Not included in the FTF PA Dose Calculator.
(b) Not included in the HTF PA Dose Calculator.

Table CC-PA-2.2: Conversion Factors for Dose Exposure Pathways

Radionuclide	**Exposure Pathways Dose Conversion Factors (mrem/yr)/(pCi/L)**					
	Irrigated Soil (Shine)		**Swimming**		**Boating/Fishing**	
	FTF	**HTF**	**FTF**	**HTF**	**FTF**	**HTF**
Ac-227	2.35E-03	3.58E-04	4.57E-06	3.79E-06	1.28E-06	5.95E-06
Ag-108m	(a)	3.54E-04	(a)	1.57E-05	(a)	2.47E-05
Al-26	1.98E-02	3.97E-03	3.48E-05	2.74E-05	1.08E-05	4.30E-05
Am-241	5.99E-05	1.17E-05	2.23E-07	1.76E-07	3.27E-08	2.76E-07
Am-242m	8.91E-05	1.69E-05	2.05E-07	1.62E-07	4.87E-08	2.55E-07
Am-243	1.19E-03	2.38E-04	2.61E-06	2.05E-06	6.53E-07	3.23E-06
Bi-210m	(a)	3.56E-04	(a)	2.51E-06	(a)	3.94E-06
C-14	1.84E-08	5.56E-10	5.21E-11	4.10E-11	1.01E-11	6.44E-11
Ca-41	(a)	0.00E+00	(a)	0.00E+00	(a)	0.00E+00
Cf-249	2.35E-03	4.57E-04	4.09E-06	3.22E-06	1.28E-06	5.06E-06
Cf-251	(a)	1.40E-04	(a)	1.16E-06	(a)	1.82E-06
Cl-36	3.12E-06	1.05E-08	5.31E-09	4.18E-09	1.71E-09	6.56E-09
Cm-243	7.74E-04	1.17E-04	1.54E-06	1.21E-06	4.23E-07	1.91E-06
Cm-244	1.73E-07	2.23E-08	1.36E-09	1.07E-09	9.43E-11	1.68E-09

Table CC-PA-2.2: Conversion Factors for Dose Exposure Pathways (Continued)

Radionuclide	Exposure Pathways Dose Conversion Factors (mrem/yr)/(pCi/L)					
	Irrigated Soil (Shine)		Swimming		Boating/Fishing	
	FTF	HTF	FTF	HTF	FTF	HTF
Cm-245	4.60E-04	9.19E-05	1.05E-06	8.23E-07	2.52E-07	1.29E-06
Cm-246	(a)	3.17E-08	(a)	9.82E-10	(a)	1.54E-09
Cm-247	2.37E-03	4.73E-04	4.15E-06	3.27E-06	1.29E-06	5.13E-06
Cm-248	1.20E-07	2.40E-08	9.44E-10	7.43E-10	6.58E-11	1.17E-09
Co-60	1.86E-02	7.59E-04	3.25E-05	2.56E-05	1.02E-05	4.02E-05
Cs-135	5.26E-08	1.58E-09	1.30E-10	1.02E-10	2.88E-11	1.61E-10
Cs-137	4.14E-03	1.15E-04	7.03E-06	5.53E-06	2.26E-06	8.68E-06
Eu-152	8.24E-03	9.34E-04	1.46E-05	1.15E-05	4.50E-06	1.81E-05
Eu-154	9.01E-03	7.82E-04	1.57E-05	1.24E-05	4.92E-06	1.95E-05
Eu-155	(a)	1.36E-05	(a)	5.23E-07	(a)	8.22E-07
Gd-152	(a)	0.00E+00	(a)	0.00E+00	(a)	0.00E+00
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-129	1.78E-05	2.26E-08	1.06E-07	8.31E-08	9.70E-09	1.31E-07
K-40	1.17E-03	1.79E-05	2.06E-06	1.62E-06	6.39E-07	2.55E-06
Lu-174	(a)	2.62E-05	(a)	1.12E-06	(a)	1.76E-06
Mo-93	(a)	1.61E-07	(a)	5.52E-09	(a)	8.67E-09
Nb-93m	1.42E-07	5.37E-11	1.23E-09	9.66E-10	7.79E-11	1.52E-09
Nb-94	1.16E-02	4.38E-06	1.98E-05	1.56E-05	6.34E-06	2.45E-05
Ni-59	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Np-237	1.43E-03	1.33E-05	2.70E-06	2.12E-06	7.81E-07	3.34E-06
Pa-231	2.46E-04	2.29E-06	4.49E-07	3.52E-07	1.35E-07	5.53E-07
Pb-210	3.35E-06	1.15E-06	1.55E-08	1.82E-08	1.83E-09	2.86E-08
Pd-107	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pu-238	2.07E-07	3.67E-08	1.35E-09	1.06E-09	1.13E-10	1.67E-09
Pu-239	3.89E-07	7.58E-08	1.14E-09	8.94E-10	2.13E-10	1.41E-09
Pu-240	2.01E-07	3.92E-08	1.32E-09	1.04E-09	1.10E-10	1.63E-09
Pu-241	8.06E-09	9.24E-10	1.92E-11	1.51E-11	4.41E-12	2.37E-11
Pu-242	1.75E-07	3.43E-08	1.11E-09	8.70E-10	9.58E-11	1.37E-09
Pu-244	2.45E-03	4.76E-04	4.19E-06	3.28E-06	1.34E-06	5.16E-06
Ra-226	4.22E-05	9.03E-04	8.24E-08	1.80E-05	2.31E-08	2.82E-05
Ra-228	1.83E-02	6.55E-04	3.30E-05	2.60E-05	1.00E-05	4.08E-05
Rn-222	1.29E-02	(b)	2.28E-05	(b)	7.04E-06	(b)
Se-79	2.55E-08	5.06E-09	7.04E-11	5.53E-11	1.39E-11	8.68E-11
Sm-147	(a)	0.00E+00	(a)	0.00E+00	(a)	0.00E+00
Sm-151	1.35E-09	2.45E-10	1.01E-11	7.92E-12	7.37E-13	1.25E-11
Sn-126	1.45E-02	2.94E-03	2.48E-05	1.96E-05	7.92E-06	3.07E-05
Sr-90	3.16E-05	4.63E-07	4.48E-08	3.52E-08	1.73E-08	5.53E-08
Tc-99	1.71E-07	3.72E-10	3.73E-10	2.93E-10	9.37E-11	4.61E-10
Th-228	1.07E-05	(b)	2.43E-08	(b)	5.83E-09	(b)
Th-229	2.02E-03	4.00E-04	3.88E-06	3.06E-06	1.11E-06	4.81E-06
Th-230	1.64E-06	3.24E-07	4.67E-09	3.67E-09	8.94E-10	5.77E-09
Th-232	7.12E-07	1.41E-07	2.36E-09	1.85E-09	3.89E-10	2.91E-09
U-232	1.22E-06	1.85E-03	3.82E-09	1.63E-05	6.67E-10	2.56E-05
U-233	1.85E-06	3.41E-07	4.32E-09	3.39E-09	1.01E-09	5.33E-09
U-234	5.48E-07	1.01E-07	2.07E-09	1.63E-09	3.00E-10	2.56E-09
U-235	1.01E-03	1.86E-04	2.03E-06	1.59E-06	5.52E-07	2.50E-06
U-236	2.92E-07	5.37E-08	1.37E-09	1.08E-09	1.60E-10	1.69E-09
U-238	1.63E-04	3.42E-05	3.11E-07	2.76E-07	8.89E-08	4.33E-07
Zr-93	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

(a) Not included in the FTF PA Dose Calculator.

(b) Not included in the HTF PA Dose Calculator.

Table CC-PA-2.3: Conversion Factors for Dose Inhalation Pathways

Radionuclide	Inhalation Pathways Dose Conversion Factors (mrem/yr)/(pCi/L)							
	Irrigation (Water)		Irrigation (Dust)		Shower		Swimming	
	FTF	HTF	FTF	HTF	FTF	HTF	FTF	HTF
Ac-227	1.29E-04	1.17E-04	3.50E-04	2.14E-04	3.68E-04	3.32E-04	4.41E-05	9.30E-06
Ag-108m	(a)	1.52E-09	(a)	6.06E-10	(a)	4.33E-09	(a)	1.21E-10
Al-26	4.11E-09	4.11E-09	1.11E-08	1.10E-08	1.17E-08	1.17E-08	1.40E-09	3.28E-10
Am-241	8.60E-06	8.60E-06	2.33E-05	2.24E-05	2.45E-05	2.45E-05	2.93E-06	6.87E-07
Am-242m	8.48E-06	8.49E-06	2.30E-05	2.12E-05	2.42E-05	2.42E-05	2.89E-06	6.78E-07
Am-243	8.43E-06	8.43E-06	2.28E-05	2.24E-05	2.40E-05	2.40E-05	2.88E-06	6.73E-07
Bi-210m	(a)	6.99E-07	(a)	1.86E-06	(a)	1.99E-06	(a)	5.58E-08
C-14	4.11E-10	4.11E-10	1.11E-09	1.65E-10	1.17E-09	1.17E-09	1.40E-10	3.28E-11
Ca-41	(a)	1.95E-11	(a)	3.97E-12	(a)	5.56E-11	(a)	1.56E-12
Cf-249	1.44E-05	1.44E-05	3.89E-05	3.72E-05	4.09E-05	4.09E-05	4.90E-06	1.15E-06
Cf-251	(a)	1.46E-05	(a)	3.84E-05	(a)	4.16E-05	(a)	1.17E-06
Cl-36	1.50E-09	1.50E-09	4.05E-09	6.69E-11	4.27E-09	4.27E-09	5.11E-10	1.20E-10
Cm-243	6.38E-06	6.38E-06	1.73E-05	1.28E-05	1.82E-05	1.82E-05	2.18E-06	5.10E-07
Cm-244	5.54E-06	5.54E-06	1.50E-05	9.51E-06	1.58E-05	1.58E-05	1.89E-06	4.43E-07
Cm-245	8.60E-06	8.60E-06	2.33E-05	2.28E-05	2.45E-05	2.45E-05	2.93E-06	6.87E-07
Cm-246	(a)	8.60E-06	(a)	2.28E-05	(a)	2.45E-05	(a)	6.87E-07
Cm-247	7.99E-06	7.99E-06	2.16E-05	2.12E-05	2.28E-05	2.28E-05	2.73E-06	6.38E-07
Cm-248	3.08E-05	3.08E-05	8.33E-05	8.17E-05	8.77E-05	8.77E-05	1.05E-05	2.46E-06
Co-60	2.05E-09	2.05E-09	5.55E-09	1.12E-09	5.85E-09	5.85E-09	7.00E-10	1.64E-10
Cs-135	1.42E-10	1.42E-10	3.83E-10	5.68E-11	4.03E-10	4.03E-10	4.83E-11	1.13E-11
Cs-137	9.43E-10	9.43E-10	2.55E-09	3.49E-10	2.69E-09	2.69E-09	3.22E-10	7.53E-11
Eu-152	8.60E-09	8.60E-09	2.33E-08	1.29E-08	2.45E-08	2.45E-08	2.93E-09	6.87E-10
Eu-154	1.09E-08	1.09E-08	2.94E-08	1.25E-08	3.10E-08	3.10E-08	3.71E-09	8.68E-10
Eu-155	(a)	1.42E-09	(a)	1.02E-09	(a)	4.03E-09	(a)	1.13E-10
Gd-152	(a)	3.90E-06	(a)	1.04E-05	(a)	1.11E-05	(a)	3.12E-07
H-3	9.27E-12	9.27E-12	2.51E-11	4.63E-14	2.64E-11	2.64E-11	3.16E-12	7.40E-13
I-129	7.38E-09	7.38E-09	2.00E-08	1.25E-10	2.10E-08	2.10E-08	2.52E-09	5.89E-10
K-40	4.31E-10	4.31E-10	1.17E-09	8.76E-11	1.23E-09	1.23E-09	1.47E-10	3.44E-11
Lu-174	(a)	8.60E-10	(a)	4.43E-10	(a)	2.45E-09	(a)	6.87E-11
Mo-93	(a)	1.21E-10	(a)	3.19E-10	(a)	3.44E-10	(a)	9.66E-12
Nb-93m	1.05E-10	1.05E-10	2.84E-10	5.25E-13	2.99E-10	2.99E-10	3.58E-11	8.37E-12
Nb-94	2.26E-09	2.26E-09	6.11E-09	1.13E-11	6.43E-09	6.43E-09	7.70E-10	1.80E-10
Ni-59	2.67E-11	2.67E-11	7.22E-11	7.54E-12	7.60E-11	7.60E-11	9.10E-12	2.13E-12
Ni-63	9.88E-11	9.88E-11	2.67E-10	2.74E-11	2.81E-10	2.81E-10	3.37E-11	7.89E-12
Np-237	4.72E-06	4.72E-06	1.28E-05	5.85E-07	1.35E-05	1.34E-05	1.61E-06	3.77E-07
Pa-231	2.87E-05	2.87E-05	7.78E-05	3.56E-06	8.18E-05	8.18E-05	9.80E-06	2.30E-06
Pb-210	2.26E-07	9.21E-07	6.11E-07	1.73E-06	6.43E-07	2.62E-06	7.70E-08	7.36E-08
Pd-107	1.21E-10	1.21E-10	3.27E-10	3.42E-11	3.44E-10	3.44E-10	4.13E-11	9.66E-12
Pu-238	9.43E-06	9.43E-06	2.55E-05	2.23E-05	2.69E-05	2.69E-05	3.22E-06	7.53E-07
Pu-239	1.03E-05	1.03E-05	2.78E-05	2.67E-05	2.92E-05	2.92E-05	3.50E-06	8.20E-07
Pu-240	1.03E-05	1.03E-05	2.78E-05	2.66E-05	2.92E-05	2.92E-05	3.50E-06	8.20E-07
Pu-241	1.85E-07	1.85E-07	5.00E-07	2.81E-07	5.26E-07	5.26E-07	6.30E-08	1.48E-08
Pu-242	9.88E-06	9.88E-06	2.67E-05	2.57E-05	2.81E-05	2.81E-05	3.37E-06	7.89E-07
Pu-244	9.65E-06	9.65E-06	2.61E-05	2.51E-05	2.75E-05	2.75E-05	3.29E-06	7.71E-07
Ra-226	7.21E-07	7.27E-07	1.95E-06	6.76E-07	2.05E-06	2.07E-06	2.46E-07	5.80E-08
Ra-228	9.41E-06	9.43E-06	2.55E-05	4.47E-06	2.68E-05	2.69E-05	3.21E-06	7.53E-07
Rn-222	5.75E-09	(b)	1.56E-08	(b)	1.64E-08	(b)	1.96E-09	(b)
Se-79	2.26E-10	2.26E-10	6.11E-10	5.97E-10	6.43E-10	6.43E-10	7.70E-11	1.80E-11
Sm-147	(a)	1.97E-06	(a)	5.23E-06	(a)	5.61E-06	(a)	1.57E-07
Sm-151	8.21E-10	8.21E-10	2.22E-09	1.98E-09	2.34E-09	2.34E-09	2.80E-10	6.56E-11
Sn-126	5.88E-09	5.83E-09	1.59E-08	1.57E-08	1.67E-08	1.66E-08	2.01E-09	4.65E-10

Table CC-PA-2.3: Conversion Factors for Dose Inhalation Pathways (Continued)

Radionuclide	Inhalation Pathways Dose Conversion Factors (mrem/yr)/(pCi/L)							
	Irrigation (Water)		Irrigation (Dust)		Shower		Swimming	
	FTF	HTF	FTF	HTF	FTF	HTF	FTF	HTF
Sr-90	7.71E-09	7.71E-09	2.09E-08	1.50E-09	2.20E-08	2.20E-08	2.63E-09	6.16E-10
Tc-99	8.21E-10	8.21E-10	2.22E-09	2.37E-11	2.34E-09	2.34E-09	2.80E-10	6.56E-11
Th-228	8.21E-06	(b)	2.22E-05	(b)	2.34E-05	(b)	2.80E-06	(b)
Th-229	1.76E-05	1.76E-05	4.77E-05	4.63E-05	5.02E-05	5.01E-05	6.02E-06	1.40E-06
Th-230	2.87E-06	2.87E-06	7.78E-06	7.57E-06	8.18E-06	8.18E-06	9.80E-07	2.30E-07
Th-232	5.13E-06	5.13E-06	1.39E-05	1.35E-05	1.46E-05	1.46E-05	1.75E-06	4.10E-07
U-232	1.60E-06	1.05E-05	4.34E-06	2.28E-05	4.57E-06	2.99E-05	5.47E-07	8.37E-07
U-233	7.38E-07	7.38E-07	2.00E-06	1.80E-06	2.10E-06	2.10E-06	2.52E-07	5.89E-08
U-234	7.21E-07	7.21E-07	1.95E-06	1.76E-06	2.05E-06	2.05E-06	2.46E-07	5.76E-08
U-235	6.38E-07	6.38E-07	1.73E-06	1.56E-06	1.82E-06	1.82E-06	2.18E-07	5.10E-08
U-236	6.55E-07	6.55E-07	1.77E-06	1.60E-06	1.86E-06	1.86E-06	2.23E-07	5.23E-08
U-238	5.95E-07	5.99E-07	1.61E-06	1.47E-06	1.70E-06	1.71E-06	2.03E-07	4.79E-08
Zr-93	2.05E-09	2.05E-09	5.55E-09	5.41E-09	5.85E-09	5.85E-09	7.00E-10	1.64E-10

(a) Not included in the FTF PA Dose Calculator.
(b) Not included in the HTF PA Dose Calculator.

Table CC-PA-2.4: Conversion Factors for Acute Dose Pathways

Radionuclide	Acute Exposure Pathways Dose Conversion Factors (m^3-mrem/Ci-yr)					
	Acute Ingestion		Acute Exposure		Acute Inhalation	
	FTF	HTF	FTF	HTF	FTF	HTF
Ac-227	3.23E+02	2.27E+02	2.46E+03	2.71E+03	1.84E+03	1.62E+03
Ag-108m	(a)	4.33E-01	(a)	1.24E+04	(a)	2.12E-02
Al-26	6.74E-01	6.61E-01	2.08E+04	2.08E+04	5.83E-02	5.72E-02
Am-241	3.84E+01	3.77E+01	6.29E+01	6.28E+01	1.22E+02	1.20E+02
Am-242m	3.84E+01	3.77E+01	9.35E+01	9.43E+01	1.20E+02	1.18E+02
Am-243	3.85E+01	3.78E+01	1.25E+03	1.25E+03	1.20E+02	1.18E+02
Bi-210m	(a)	2.82E+00	(a)	1.87E+03	(a)	9.74E+00
C-14	1.11E-01	1.09E-01	1.93E-02	1.93E-02	5.83E-03	5.72E-03
Ca-41	(a)	3.58E-02	(a)	0.00E+00	(a)	2.72E-04
Cf-249	6.74E+01	6.61E+01	2.46E+03	2.46E+03	2.04E+02	2.00E+02
Cf-251	(a)	6.77E+01	(a)	7.41E+02	(a)	2.03E+02
Cl-36	1.78E-01	1.75E-01	3.28E+00	3.27E+00	2.13E-02	2.09E-02
Cm-243	2.88E+01	2.82E+01	8.12E+02	8.12E+02	9.06E+01	8.89E+01
Cm-244	2.30E+01	2.26E+01	1.81E-01	1.81E-01	7.87E+01	7.73E+01
Cm-245	4.03E+01	3.95E+01	4.83E+02	4.83E+02	1.22E+02	1.20E+02
Cm-246	(a)	3.95E+01	(a)	1.67E-01	(a)	1.20E+02
Cm-247	3.65E+01	3.58E+01	2.48E+03	2.48E+03	1.13E+02	1.11E+02
Cm-248	1.48E+02	1.45E+02	1.26E-01	1.26E-01	4.37E+02	4.29E+02
Co-60	6.53E-01	6.41E-01	1.95E+04	1.95E+04	2.91E-02	2.86E-02
Cs-135	3.84E-01	3.77E-01	5.52E-02	5.50E-02	2.01E-03	1.97E-03
Cs-137	2.49E+00	2.45E+00	4.35E+03	4.35E+03	1.34E-02	1.32E-02
Eu-152	2.68E-01	2.64E-01	8.65E+03	8.65E+03	1.22E-01	1.20E-01
Eu-154	3.84E-01	3.77E-01	9.45E+03	9.45E+03	1.54E-01	1.52E-01
Eu-155	(a)	6.00E-02	(a)	2.62E+02	(a)	1.97E-02
Gd-152	(a)	7.73E+00	(a)	0.00E+00	(a)	5.44E+01
H-3	3.45E-03	3.39E-03	0.00E+00	0.00E+00	1.32E-04	1.29E-04
I-129	2.11E+01	2.07E+01	1.86E+01	1.86E+01	1.05E-01	1.03E-01
K-40	1.19E+00	1.17E+00	1.23E+03	1.23E+03	6.12E-03	6.01E-03
Lu-174	(a)	5.08E-02	(a)	7.11E+02	(a)	1.20E-02
Mo-93	(a)	5.85E-01	(a)	8.49E-01	(a)	1.69E-03
Nb-93m	2.30E-02	2.26E-02	1.50E-01	1.50E-01	1.49E-03	1.46E-03

Table CC-PA-2.4: Conversion Factors for Acute Dose Pathways (Continued)

Radionuclide	Acute Exposure Pathways Dose Conversion Factors (m^3-mrem/Ci-yr)					
	Acute Ingestion		Acute Exposure		Acute Inhalation	
	FTF	HTF	FTF	HTF	FTF	HTF
Nb-94	3.26E-01	3.20E-01	1.22E+04	1.22E+04	3.21E-02	3.15E-02
Ni-59	1.21E-02	1.19E-02	0.00E+00	0.00E+00	3.79E-04	3.72E-04
Ni-63	2.88E-02	2.82E-02	0.00E+00	0.00E+00	1.40E-03	1.38E-03
Np-237	2.13E+01	2.09E+01	1.50E+03	1.50E+03	6.70E+01	6.58E+01
Pa-231	1.36E+02	1.34E+02	2.58E+02	2.58E+02	4.08E+02	4.01E+02
Pb-210	1.32E+02	3.56E+02	3.52E+00	8.58E+00	3.21E+00	1.28E+01
Pd-107	7.10E-03	6.97E-03	0.00E+00	0.00E+00	1.72E-03	1.69E-03
Pu-238	4.41E+01	4.33E+01	2.17E-01	2.17E-01	1.34E+02	1.32E+02
Pu-239	4.79E+01	4.71E+01	4.08E-01	4.07E-01	1.46E+02	1.43E+02
Pu-240	4.79E+01	4.71E+01	2.11E-01	2.11E-01	1.46E+02	1.43E+02
Pu-241	9.22E-01	9.06E-01	8.46E-03	8.46E-03	2.62E+00	2.58E+00
Pu-242	4.60E+01	4.52E+01	1.84E-01	1.84E-01	1.40E+02	1.38E+02
Pu-244	4.62E+01	4.54E+01	2.57E+03	2.55E+03	1.37E+02	1.35E+02
Ra-226	5.39E+01	5.29E+01	4.43E+01	1.36E+04	1.02E+01	1.01E+01
Ra-228	1.60E+02	1.57E+02	1.93E+04	1.93E+04	1.34E+02	1.32E+02
Rn-222	4.79E-02	(b)	1.35E+04	(b)	8.16E-02	(b)
Se-79	5.55E-01	5.44E-01	2.68E-02	2.67E-02	3.21E-03	3.15E-03
Sm-147	(a)	9.21E+00	(a)	0.00E+00	(a)	2.75E+01
Sm-151	1.88E-02	1.85E-02	1.42E-03	1.42E-03	1.17E-02	1.15E-02
Sn-126	9.74E-01	9.57E-01	1.52E+04	1.52E+04	8.35E-02	8.12E-02
Sr-90	5.91E+00	5.80E+00	3.32E+01	3.31E+01	1.10E-01	1.08E-01
Tc-99	1.23E-01	1.21E-01	1.80E-01	1.80E-01	1.17E-02	1.15E-02
Th-228	1.38E+01	(b)	1.12E+01	(b)	1.17E+02	(b)
Th-229	1.17E+02	1.16E+02	2.12E+03	2.12E+03	2.50E+02	2.45E+02
Th-230	4.03E+01	3.95E+01	1.72E+00	1.72E+00	4.08E+01	4.01E+01
Th-232	4.41E+01	4.33E+01	7.47E-01	7.48E-01	7.29E+01	7.15E+01
U-232	6.32E+01	8.90E+01	1.28E+00	1.18E+04	2.28E+01	1.46E+02
U-233	9.79E+00	9.62E+00	1.95E+00	1.94E+00	1.05E+01	1.03E+01
U-234	9.38E+00	9.21E+00	5.75E-01	5.75E-01	1.02E+01	1.01E+01
U-235	9.08E+00	8.90E+00	1.06E+03	1.06E+03	9.06E+00	8.89E+00
U-236	9.02E+00	8.85E+00	3.06E-01	3.06E-01	9.30E+00	9.13E+00
U-238	9.31E+00	9.11E+00	1.71E+02	1.95E+02	8.45E+00	8.35E+00
Zr-93	2.11E-01	2.07E-01	0.00E+00	0.00E+00	2.91E-02	2.86E-02

(a) Not included in the FTF PA Dose Calculator.
(b) Not included in the HTF PA Dose Calculator.

Inventory (INV)

RAI-INV-1

Annular volume uncertainty could be quantified.

Basis

NUREG-1854 states that the reviewer should evaluate the methods used for estimating waste volumes, and confirm that uncertainty has been considered and propagated into the inventory estimate. On page 40 of SRR-CWDA-2011-00126, Rev. 0, DOE estimates that approximately 18 m^3 (4,700 gal) is remaining in the Tank 16 Annulus using a depth profile of the waste in the annulus. Figure 3.6-4 of SRR-CWDA-2011-00126, Rev. 0 shows nine depth values, although it is unclear if these depth values were determined visually or measured. DOE later estimated the remaining volume as about 12.5 m^3 (3,300 gal) (HTF Performance Assessment and SRR-LWE-2012-00039, Rev. 0) through the use of camera views and interior landmarks (i.e., duct diameter, annulus wall radius) as well as a depth measurement under each of the four risers. DOE separates the residual annulus material into: (1) material at the bottom of the annulus outside the ventilation duct (estimated at 8 m^3 (2,100 gal)), and (2) material inside the ventilation duct (estimated at 4.5 m^3 (1,200 gal)). There are a total of 13 inspection ports in the Tank 16H annulus, and DOE visually estimated material depths from multiple locations in the annulus outside the duct and from five locations inside the duct. These depth estimates range from 5 cm (2 in) to 40 cm (16 in). Still, there are many areas of the annulus (and inside the duct) where visual determination of the waste level was not possible, so there is some degree of uncertainty associated with the estimate. In those areas, DOE extrapolated the waste level using the data from surrounding areas. DOE clarified that the more recent estimate is the correct one, but it is unclear why this value is superior to the first one and also the reasons for the differences between the two values assuming they were both made using estimated profiles of the waste.

Also, in Section 9.0 of SRR-CWDA-2010-00023, Rev. 3, DOE describes how multipliers are applied to the tank inventories to account for uncertainty in the HTF Performance Assessment. However, similar inventory multipliers are not applied for the annuli of the tanks. The HTF Performance Assessment states that 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 9H, 10H and 14H annuli contain about 3,300, 1,100, and 5,600 gal of material respectively, although 3,300 gal is assumed for all tanks with residual material in the annulus in the HTF Performance Assessment (ML13183A410).

Path Forward

Provide estimates of the uncertainty of remaining volume in Tank 16H annulus and describe how this uncertainty is related to the uncertainty of the material in the annuli or other tanks. Clarify why the 12.5 m^3 (3,300 gal) estimate is more accurate than the 18 m^3 (4,700 gal) estimate and the reason for the different values.

Response RAI-INV-1

Since a final residuals inventory determination has not been completed for Tank 16, quantification of the volume and concentration uncertainties surrounding the Tank 16 inventory have not been finalized. The waste tank inventories considered in the HTF PA are inventory assignments (i.e., risk informed inventory projections with integrated conservatism) that were developed using all of the current information available. The specific purpose of the assignment was to evaluate if there is reasonable assurance that the specific requirements of DOE Manual 435.1-1 and NDAA Section 3116 will be met. As future waste tank cleaning operations proceed; and waste heel removal, sampling, and analyses are completed, final waste tank residual inventories will be available. These final residual inventories will be compared to the current HTF PA assigned inventories for that specific waste tank on a radionuclide-by-radionuclide basis. This information will be evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1.

The 4,700-gallon estimate for the Tank 16 annulus from *Tank 16 History of Waste Removal 1959 Through 2010*, SRR-CWDA-2011-00126, was a preliminary estimate, not a final volume determination. At the time of the inventory estimate development, the 3,300-gallon estimate for annulus residual volume from the HTF PA was considered a conservative and bounding estimate and any uncertainty would reflect a decrease in the inventory estimate. Current Tank 16 annulus sampling evolutions are confirming that the 3,300-gallon estimate is conservative.

The conservative and bounding expectation for remaining volume in the annulus of Tanks 9, 10, and 14 was based on future cleaning plans and assumed waste characteristics for these waste tanks. It is highly likely that a significant fraction of the salt waste in the annulus of Tanks 9, 10, and 14 will be removed prior to grouting. This is based on the leakage history of these three waste tanks, the likely characteristics of the salt waste currently contained in their respective annulus, and the physical and chemical differences when compared to the waste in the Tank 16 annulus.

It was understood that by only utilizing a reasonably conservative base case inventory value and not including an annulus inventory uncertainty, the probabilistic analysis would be conservative with respect to annulus inventories, since this approach would not consider that the majority of expected inventory outcomes would tend to fall on the low side of the inventory distribution. At closure, a volumetric uncertainty will be appropriately incorporated into the final inventory determination and utilized in the waste tank-specific SA.

RAI-INV-2

The representativeness of Tank 16H annulus samples for Tanks 9H, 10H, and 14H is not clear.

Basis

Section 2.5.1.2 of SRR-CWDA-2010-00023, Rev. 3 describes how, for those radionuclides analyzed in the four Tank 16H annulus samples, DOE applies the concentrations of the Tank 16H annulus sample results to other tanks with annulus material (Tanks 9H, 10H and 14H). DOE also states that the material in the Tank 16H annulus is expected to be chemically different than the other tanks due to the unique circumstances of the leak in Tank 16H (i.e., rapid leak) and the history of Tank 16H annulus (i.e., use of sand in determination of leak sites). The NRC staff recognizes that the data on annulus material was limited at the time of the development of the draft Basis for Waste Determination, and that the only annulus samples available currently are those taken from Tank 16H annulus. However, the impact on the expected differences between Tank 16H annulus material and that of the other tanks on the inventory estimate could be further explained.

Path Forward

Please clarify the technical basis for using Tank 16H annulus concentrations in the annuli of Tanks 9H, 10H, and 14H given the expected chemical differences between Tank 16H annulus material and that of the other tanks. Please describe the impact of the anticipated differences on the inventory estimates in terms of radionuclides that could be under- or over-estimated as a result of assuming the Tank 16H annulus concentrations.

Response RAI-INV-2

The waste tank inventories considered in the HTF PA are inventory assignments (i.e., risk informed inventory projections with integrated conservatism) that were developed using all of the current information available. The specific purpose of the assignment was to evaluate if there is reasonable assurance that the specific requirements of DOE Manual 435.1-1 and NDAA Section 3116 will be met, and not to predict final conditions. As future waste tank cleaning operations proceed; and waste heel removal, sampling, and analyses are completed, final waste tank residual inventories will be available. These final residual inventories will be compared to the current PA assigned inventories for that specific waste tank on a radionuclide-by-radionuclide basis. This information will be evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1.

The goal during the development of the annulus inventory estimates was to ensure conservative estimates were assigned, not necessarily representative estimates. The annulus inventory estimates in Tanks 9, 10, and 14 were based on estimates of both residual concentration and volume, both of which provided conservatisms.

The concentration assignments were based on knowledge gained from samples taken from the Tank 16 annulus. The sample analysis from Tank 16 was utilized since it provided sample characterization data for actual annulus residual material. Since samples of the waste contained in the annulus of Tanks 9, 10 and 14 have not been taken, assumptions of the waste characteristics are required. The concentration assignments were considered conservative due to the expected solubility behavior of the specific radionuclides, as discussed next.

The conservatism, with respect to the soluble species concentrations, is based on the expectation that cleaning activities will add liquid to dissolve the material in each annulus. This expectation results from an understanding of the leakage history of these waste tanks that suggests a majority of the material is salt, formed over time from evaporation of supernate that has weeped from the primary tank into the annulus. This supernate should not contain significant quantities of insoluble solids. Unlike Tank 16, sand has not been introduced into any other annulus. Given these facts, the material present in the annulus of Tanks 9, 10, and 14 should be highly soluble in water. The cleaning activity will dissolve soluble species, thereby reducing their concentrations in the final material and providing conservatism in the soluble species concentrations.

The conservatism with respect to the insoluble species concentrations is based on the nature of the Tank 16 primary liner leaks. Since the Tank 16 annulus material was due to a rapid leak, this annulus material is expected to be higher in insoluble species than other material formed from a supernate only leak at a slower leak rate. Therefore, the insoluble species concentrations are expected to be lower in Tanks 9, 10, and 14 than those measured in the Tank 16 material, since the Tank 16 rapid leaks contained higher concentrations of insoluble species.

The final part of the annulus inventory estimate’s conservatism is based on the annulus residual volume estimate. The annulus residual volume estimates for Tanks 9, 10, and 14 were based on the current Tank 16 annulus volume estimate. Cleaning of the annulus material associated with Tanks 9, 10, and 14 is planned. These cleaning activities are expected to remove a significant portion of the annulus material, therefore, the annulus residual volume estimate of 3,300 gallons is believed to be conservative.

RAI-INV-3

There are differences in processes between HTF and FTF which are not reflected in projected inventories.

Basis

NUREG 1854 advises the reviewers to determine whether there is an adequate technical basis for the accuracy of inventory estimates based on historical or process knowledge. HTF and FTF processed different waste streams, which would lead to expected differences in the inventories. SRNL-STI-2012-00479 states that there is expected to be approximately 2000 times more U-234 at HTF than at FTF because HTF mainly processed enriched uranium while FTF processed irradiated depleted uranium targets. There is also expected to be about 100 times more U-233 and 70 times more U-236 at HTF.

The Waste Characterization System did not provide values for U-234, or U-236, so DOE estimated these inventories using other methods. SRR-CWDA-2010-00023, Rev. 3 states, "...there were other constituents for which WCS did not estimate an inventory for all HTF tanks. Therefore, these radionuclide concentrations were estimated only for waste tanks where no values were present in WCS. The isotopes requiring additional estimates were 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." Table 2.2-1 in SRR-CWDA-2010-00023, Rev. 3 describes the methods used to estimate Ba-137m, C-14, H-3, Y-90, B, and Mo. However, SRR-CWDA-2010-00023, Rev. 3 does not provide details on how the other radionuclides not included in the Waste Characterization System were estimated. Details should be provided as to the methods for estimation and whether inventories were estimated based on the values of other radionuclides for the same tank or for other tanks.

The estimates for some radionuclides (e.g., U-234) do not appear to agree with expectations provided in SRNL-STI-2012-00479, Rev. 3. DOE estimates 24.42 GBq (0.66 Ci) of U-234 in the Type III/IIIA tanks at HTF. Assuming 15m^3 (4,000 gal) of waste per tank and 1.2 g/cm^3 bulk (dry) density of waste, results in approximately 6 mg U-234 per kg of waste. This concentration is similar to the average U-234 in FTF waste of 7-8 mg/kg reported in WSRC-STI-2007-00192, Rev. 1. It might be coincidental that the projected U-234 concentrations for HTF happen to be similar to the FTF data. However, if some constituents that were not reported in the Waste Characterization System for HTF were estimated based on data from FTF, it is possible that certain radionuclides could have been underestimated since differences are expected in the waste streams between HTF and FTF. Examples of other potentially important radionuclides that do not follow the expectations from SRNL-STI-2012-00479, Rev. 3 are U-233, U-236. In addition to explaining the methods for estimating those radionuclides that are not listed in the Waste Characterization System, it would be helpful if DOE could explain why inventory projection ratios may not follow the expectations in SRNL-STI-2012-00479, Rev. 3.

Path Forward

Please clarify how DOE estimated the inventories for radionuclides not included in the Waste Characterization System and not explained in Table 2.2-1 of SRR-CWDA-2010-00023, Rev. 3. Please also explain the differences in inventory projections from the expectations that are described in SRNL-STI-2012-00479, Rev. 3.

Response RAI-INV-3

The waste tank inventories considered in the HTF PA are inventory assignments (i.e., risk informed inventory projections with integrated conservatism) that were developed using all of the current information available. The specific purpose of the assignment was to evaluate if there is reasonable assurance that the specific requirements of DOE Manual 435.1-1 and NDAA Section 3116 will be met. As future waste tank cleaning operations proceed; and waste heel removal, sampling, and analyses are completed, final waste tank residual inventories will be available. These final residual inventories will be compared to the current PA-assigned inventories for that specific waste tank on a radionuclide-by-radionuclide basis. This information will be evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1.

Development of the initial inventory assignments was based primarily on the WCS via the report *Information on the Radiological and Chemical Characterization of the Savannah River Site Tank Waste As of July 5, 2011*, SRR-LWE-2011-00201, that summarized the waste tank inventories in WCS as well as other technical information used to estimate waste tank inventories. The inventories and volumes from the July 2011 waste characterization report were used to estimate each waste tank's constituent specific concentration. This specific concentration was applied to each waste tank's residual volume assignment to calculate the initial residual inventory.

The report, *Chemical Differences Between Sludge Solids at the F and H Area Tank Farms*, SRNL-STI-2012-00479, was commissioned specifically to inform the HTF inventory assignments. While inventory assignments for the HTF PA considered this general comparison information, individual waste tank inventories did not necessarily align with this general information. It should be noted that the chemical differences report used the same basis as the HTF PA inventory assignments. The concern raised in the basis to this RAI (constituents not reported in WCS) is illustrated within the chemical differences report. For example, the U-234 estimated curies in Tank 5, Table 3-3 of the Chemical Differences report, lists no inventory. Since Tank 5 (as well as other waste tanks listed with zero curies) is known to contain U-234, an estimating method to account for these inventories is needed (the estimating method is discussed below). The basis for expecting 2,000 times more U-234 in HTF than FTF comes from the totals in Table 3-3 of the Chemical Differences report. Since Table 3-3 of the report lists no inventory for several waste tanks in FTF, the ratio comparing the tank farm inventories was distorted.

Since one of the inventory adjustment steps applies the maximum waste tank inventory within a grouping to the other waste tanks within that grouping, an individual waste tank might not follow the general expectations. This grouping technique can lead to the inventory assignments for numerous waste tanks inventories being different from expectations, although these estimates would be purposefully biased conservative relative to expectations. As an example, for U-233, the Type III and IIIA tanks inventory estimate is 1.3E+00 curies. This inventory estimate is based on the Tank 42 inventory estimate. The other Type III and IIIA tanks' U-233 "expected" inventories range from 1.2E-01 to 8.6E-01 curies but a value of 1.3E+00 curies is assigned to all waste tanks in the group.

For constituents not included in WCS, using the average concentration of similar tank types and materials is typically used as the assignment method. The impact of this "averaging" method is marginalized due to the adjustment step that applies the maximum inventory within the waste tank's grouping to each waste tank within the grouping. This step (maximum inventory within waste tank groupings) helps to safeguard against any underestimates caused by differences between FTF and HTF wastes.

RAI-INV-4

The representativeness of 2010 Tank 16H annulus samples for the concentration of the material in the Tank 16 annulus is not clear.

Basis

DOE uses the concentration of the four samples taken in 2010 (all from outside of the ventilation duct) to represent all of the Tank 16H 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 (HTF Performance Assessment) has higher concentrations of several highly radioactive radionuclides based on comparison of the 2006 samples (WSRC-STI-2008-00203, Rev. 0) to the 2010 samples. Note that the two samples taken from outside the ventilation duct in 2006 also have higher concentrations than the 2010 samples. DOE has not adequately explained why the 2006 samples were not used in the inventory calculation for Tank 16H annulus. DOE has also not provided adequate justification for why the concentration of the samples taken from outside the duct in 2010 adequately represents the concentrations of material inside the duct for the purposes of the HTF Performance Assessment. NRC staff acknowledges that the Tank 16H annulus will be sampled again for purposes of final characterization and that the final inventories will be evaluated in a special analysis. However, it is not clear why results from the samples taken in 2006 were not used to help inform the projected inventory calculations in the HTF Performance Assessment, especially if the concentrations were higher for certain highly radioactive radionuclides.

Table 4. Results of Annulus Samples Collected in 2006

	HTF-16-06-104 (IP-35 Inside Duct)	HTF-16-06-105 (IP-35 Outside Duct)	HTF-16-06-106 (IP-II8 Outside Duct)
U total (IM) (mg/kg)	6.64E+02	5.49E+02	7.03E+01
Pu-238 (SA) (dpm/kg)	1.84E+10	1.15E+10	1.24E+09
Pu-239 (SA) (dpm/kg)	3.46E+09	2.08E+09	2.26E+08
Np-237 (IM) (dpm/kg)	1.39E+07	1.13E+07	1.65E+06
Sr-90 (BS) (dpm/g)*	1.21E+10		
Cs-137 (GS) (dpm/g)*	7.84E+08		

* results of the Cs-137 and Sr-90 analysis of an archived portion of sample (HTF-16-06-104) (SRNL-STI-2012-00178)
IM: Inductively Coupled Plasma-Mass Spectrometry
SA: Separation/Alpha Spectroscopy
GS: Gamma Spectroscopy
BS: Beta Spectroscopy

Table 5. Results of Annulus Samples Collected in 2010

	N	S	E	W	Average
U total (mg/kg)	5.28E+01	1.25E+02	2.59E+02	2.58E+02	1.74E+02
Pu-238 (SA) (dpm/kg)	1.18E+06	2.72E+06	4.22E+06	5.81E+06	3.48E+06
Pu-239 (SA) (dpm/kg)	1.71E+05	4.50E+05	7.31E+05	1.09E+06	6.11E+05
Np-237 (IM) (dpm/kg)	1.50E+03	2.67E+03	4.05E+03	4.15E+03	3.09E+03
Sr-90 (BS) (dpm/g)	3.65E+08	1.55E+09	1.54E+09	2.58E+09	1.51E+09
Cs-137 (GS) (dpm/g)	2.33E+09	1.63E+09	2.09E+09	3.15E+09	2.30E+09

Path Forward

DOE should clarify why the 2006 annulus samples were not used (in addition to the 2010 annulus samples) to help inform the inventory calculation for Tank 16 annulus. DOE should also provide justification for why the concentration of the samples taken from outside the ventilation duct in 2010 adequately represents the concentration of material inside the duct for the purposes of the HTF Performance Assessment.

Response RAI-INV-4

The waste tank annulus inventories considered in the HTF PA are inventory assignments (i.e., risk-informed inventory projections with integrated conservatism) that were developed using all of the current information available. The specific purpose of the assignment was to evaluate if there is reasonable assurance that the specific requirements of DOE Manual 435.1-1 and NDAA Section 3116 will be met, and not to predict final conditions. As future waste tank cleaning operations proceed; and waste heel removal, sampling, and analyses are completed, final waste tank residual inventories will be available. These final residual inventories will be compared to the current PA-assigned inventories for that specific waste tank on a radionuclide-by-radionuclide basis. This information will be evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1.

The goal during the development of the annulus inventory estimates was to ensure conservative estimates were assigned, not necessarily representative estimates. The annulus inventory estimate for Tank 16 relied predominately on the 2011 samples and disregarded the 2006 samples because of the known conditions under which the 2011 samples were taken and the then planned cleaning process.

The 2011 samples were collected under a sample plan that was developed primarily to obtain process information, but also included in the sample plan was the idea that the samples could also be used to inform closure projections. The 2011 sample design represented more area than the 2006 sample design by collecting four samples as compared to two samples (not including the duct sample that is discussed later). There was more familiarity with the sample plan, quality assurance, and sample design, which provided more confidence in using the results for closure assignment purposes.

At the time the inventory estimate was developed, the material outside the duct was expected to make up the bulk of the residual material at the time of closure due to an earlier expectation that a large portion of the duct material would be removed as part of an annulus cleaning process. The material in the duct was expected to be more soluble, which led to the assumption that less material would remain following the cleaning process. For this reason, the samples of the material outside the duct were considered more applicable to the residual inventory estimate as modeled in the HTF PA. The current status of Tank 16 waste removal activities is discussed in the response to RAI-MEP-7.

A comparison of the 2006 and 2011 sample analyses (Table RAI-INV-4.1), shows, for the radionuclides listed, the results were closer than listed in the tables in the RAI-INV-4 *Basis* (Table 4 and 5). Converting the sample results to the same units in Table RAI-INV-4.1 illustrates that the 2006 and 2011 sample analysis are more similar that previously recognized.

Table RAI-INV-4.1: Comparison of 2006 and 2011 Tank 16 Annulus Sample Analysis Results

	2006 Samples (WSRC-STI-2008-00203)			2011 Samples (SRNL-STI-2012-00178)			
Radionuclide	**HTF-16-06-104[a] mCi/kg**	**HTF-16-06-105[b] mCi/kg**	**HTF-16-06-106[c] mCi/kg**	**HTF-16-N-1 mCi/kg**	**HTF-16-S-2 mCi/kg**	**HTF-16-E-2 mCi/kg**	**HTF-16-W-1 mCi/kg**
Sr-90	5.50E+03	NA	NA	1.64E+02	6.98E+02	6.94E+02	1.16E+03
Tc-99	5.44E-01	1.19E+00	4.28E-01	1.31E-01	2.46E-01	2.09E-01	4.50E-01
Cs-137	3.56E+02	NA	NA	1.05E+03	7.34E+02	9.41E+02	1.42E+03
Pu-238	8.28E+00	5.16E+00	5.60E-01	5.32E-01	1.23E+00	1.90E+00	2.62E+00
Pu-239/240	1.56E+00	9.35E-01	1.02E-01	7.70E-02	2.03E-01	3.29E-01	4.91E-01
Np-237	6.26E-03	5.08E-03	7.42E-04	6.76E-04	1.20E-03	1.82E-03	1.87E-03

a: IP-35 Inside Duct
b: IP-35 Outside Duct
c: IP-118 Outside Duct

Table RAI-INV-4.2 shows estimates of the inventories for the radionuclides listed in Table RAI-INV-4.1 assuming that the 2006 sample results were included in the inventory assignment process. In calculating the Table RAI-INV-4.2 values, the 2006 sample analysis was used exclusively for the inside duct inventory estimate. The average of the 2006 (two samples) and 2011 (four samples) analyses was used for the outside duct inventory estimate. A volume of 1,200 gallons was used for inside the duct and 2,100 gallon was used for outside the duct. [SRR-LWE-2013-00010] A density of 1.5 g/mL was used in the Table RAI-INV-4.2 estimates.

Table RAI-INV-4.2: Inventory Impact of Incorporating 2006 Tank 16 Annulus Sample Analysis Results

Radionuclide	Inside Duct Concentration (mCi/kg)	Inside Duct Inventory Estimate (Ci)	Outside Duct Concentration (mCi/kg)	Outside Duct Inventory Estimate (Ci)	Total Inventory Estimate (Ci)	PA Inventory (Ci)	Ratio to PA Inventory
Sr-90	5.5E+03	3.7E+04	6.8E+02	8.1E+03	4.6E+04	7.8E+03	5.8
Tc-99	5.4E-01	3.7E+00	4.4E-01	5.3E+00	9.0E+00	4.9E+00	1.8
Cs-137	3.6E+02	2.4E+03	1.0E+03	1.2E+04	1.5E+04	1.7E+04	0.9
Np-237	6.3E-03	4.3E-02	1.9E-03	2.3E-02	6.5E-02	2.6E-02	2.5
Pu-238	8.3E+00	5.6E+01	2.0E+00	2.4E+01	8.0E+01	2.5E+01	3.2
Pu-239/240	1.6E+00	1.1E+01	3.6E-01	4.2E+00	1.5E+01	7.8E+00	1.9

As shown in Table RAI-INV-4.2, incorporating the 2006 Tank 16 annulus sample analysis results into the HTF PA inventory would have a relatively minor (i.e., less than a factor of 6 in the worst case) impact on the annulus inventory. As discussed in the response to RAI-NF-12, annulus inventories are not expected to contribute significantly to risk. The radionuclide contributions to the MOP dose from only the annular inventories were evaluated using modeling assumptions in the response to RAI-NF-12. This model releases the majority of the contaminants at the start of the simulation and applies assumptions to accelerate flow. In this modeling case, 50 % of the Tc-99 inventories associated with the annulus are released within the first 500 years, and more than 99 % of the Tc-99 inventories is released in the first 3,000 years. In this modeling scenario, the total dose peaks at about 13 mrem/yr (around 400 years after final facility closure) assuming releases from the annular inventories of all the HTF waste tanks. This sensitivity analysis compares to the Base Case (Case A) modeling approach, which peaked at less than 4 mrem/yr (around 9,800 years after final facility closure). Increasing the peak dose driver (Tc-99) inventory for all waste tanks by 1.8 times (as a consequence of incorporating the 2006 Tank 16 annulus sample analysis results into the HTF PA inventory for all waste tanks), would not increase the peak dose results from the RAI-NF-12 response above

23 mrem/yr (1.8 times 13 mrem/yr). The other radionuclides listed in Table RAI-INV-4.2 (e.g., Sr-90) did not contribute enough to the 13 mrem/yr peak dose shown in the RAI-NF-12 response (i.e., all contribute less than 1.0 mrem/yr) to contribute significantly even considering the inventory ratios from Table RAI-INV-4.2.

Finally, sampling of the Tank 16 annulus is ongoing. These sampling evolutions, in general, are showing that the current volume estimate of 3,300 gallons in the annulus is likely conservative. This trend is especially true where the annulus duct samples have been taken. If the final residual volume in the annulus does decrease from the current estimate, the impact of the annulus residuals to the overall HTF dose becomes even less significant.

RAI-INV-5

Cs, Sr, and Zr inventory adjustments require further justification.

Basis

During the May 16, 2013 clarification teleconference, NRC staff asked DOE to clarify the assumptions regarding Cs-137, Sr-90, and Zr-93 inventories (ML13193A072). 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 Waste Characterization System and 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. SRR-CWDA-2013-00086, Rev. 0 reports initial concentration in Table A3-5 which is adapted in Table 6.

Table 6. Initial Concentrations of Cs-137, Sr-90, and Zr-93 for Tanks 9H-15H. (Adapted from Table A3-5 of SRR-CWDA-2013-00086, Rev. 0.)

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

*To convert Ci to Bq multiply by 3.7×10^{10}

DOE calculates the total activity (in Curies) by assuming 4,000 gallons per tank and decays it to the time of 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 5F. SRR-CWDA-2013-00086, Rev. 0 reports Tank 5F process sample concentrations before and after chemical cleaning in Table A3-9 which is reproduced in Table 7.

The results in Table 7 were measured from a single process sample that was taken during mechanical sludge removal campaigns and a single process sample taken after chemical cleaning, and therefore do not 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 Tanks 5F and 6F (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. Tanks 18F and 19F (Type IV) have also been fully characterized, but they were not chemically cleaned due to the zeolite resin in these tanks. Two of the Type IV tanks in HTF (Tank 23H and 24H) contain or have processed zeolites, as well as several of the Type III/IIIA HTF Tanks.

Table 7. Tank 5F Concentrations for Cs-137, Sr-90, Elemental Sr and Zr Before and After Chemical Cleaning. (Adapted from Table A3-9 of SRR-CWDA-2013-00086, Rev. 0.)

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

*To convert Ci to Bq multiply by 3.7x10^{10}

(Tanks 32H, 38H, 40H, 42H, and 51H). Chemical cleaning of tanks with zeolite may not be practical, and therefore adjusting the inventory of these tanks may not be appropriate. As can be seen in Table 8, the majority of the projected inventories for these the tanks that have already been cleaned were not overestimated by an order of magnitude (i.e., 10x). All inventories were estimated within one order of magnitude and many of the projected inventories were underestimated, especially with respect to Zr-93. Also, the final ratio of Sr-90 to Zr-93 is substantially less than the 58,000:1 for Tanks 5F-6F and Tank 19F.

Table 8. 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 5F/6F (Type I)		Tanks 18F/19F (Type IV)		HTF Projections	
	Projected[1] (Ci)	Measured[2] (Ci)	Projected[1] (Ci)	Measured[2] (Ci)	Type I[3] (Ci)	Type IV[3] (Ci)
Cs-137	**5/6:** 9.2E3	**5:** 3.5E3 **6:** 6.7E3	**18:** 9.7E3 **19:** 6.5E3	**18:** 9.2E3 **19:** 4.2E3	7.9E-2	2.4E3
Sr-90	**5/6:** 1.3E5	**5:** 9.7E4 **6:** 2.0E5	**18:** 1.1E3 **19:** 5.2E0	**18:** 2.5E3 **19:** 6.9E0	1.4E4	3.1E2
Zr-93	**5/6:** 1.0E-3	**5:** 3.0E1 **6:** 2.2E1	**18:** 1.0E-3 **19:** 1.0E-3	**18:** 8.6E-2 **19:** 1.8E-2	4.0E-1	8.8E-3
Sr-90:Zr-93		**5:** 3E3 **6:** 9E3		**18:** 3E4 **19:** 4E2	3.5E4	3.5E4

Notes: [1] SRS-REG-2007-00002, Rev. 1
[2] SRR-CWDA-2012-00071, Rev. 0
[3] SRR-CWDA-2010-00128, Rev. 1
*To convert Ci to Bq multiply by 3.7x10^{10}

Path Forward

DOE should provide additional justification for estimating the zirconium inventory based on an assumed ratio to the Sr-90 concentration given that the measured results for the tanks which have been cleaned do not reflect a 58,000:1 Sr-90 to Zr-93 ratio. DOE should also clarify the basis for reducing cesium, strontium, and zirconium by a one order of magnitude for all tank types given that the experience with cleaning the tanks thus far does not reflect an overestimation, but rather an underestimation in many cases. Finally, DOE should clarify if the tanks containing zeolite in HTF are intended to be cleaned using chemical cleaning and how this impacts the assumed cesium, strontium, and zirconium inventory for those tanks.

Response RAI-INV-5

The waste tank inventories considered in the HTF PA are inventory assignments (i.e., risk informed inventory projections with integrated conservatism) that were developed using the current information available at the time the HTF PA was being developed. The specific purpose of the assignment was to evaluate if there is reasonable assurance that the specific requirements of DOE Manual 435.1-1 and NDAA Section 3116 will be met. As future waste tank cleaning operations proceed; and waste heel removal, sampling, and analyses are completed, final waste tank residual inventories will be available. These final residual inventories will be compared to the current PA assigned inventories for that specific waste tank on a radionuclide-by-radionuclide basis. This information will be evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1.

At the time that the HTF inventory assignments were developed, the Sr-90 to Zr-93 58,000:1 ratio was based on sludge batch analyses, which incorporate material from many different waste tanks, particularly HTF material. [WSRC-RP-99-00436, WSRC-TR-2005-00157, WSRC-STI-2008-00142, SRNL-STI-2009-00821] This ratio was believed to be a better indication of the overall ratio for HTF than a ratio based on results from a small number of waste tanks in FTF. As updates to the inventory assignments are made, the measured Sr-90 to Zr-93 ratios for Tanks 5, 6, 18, and 19 will be considered again, as well as any other new information (especially new information from HTF sources).

The reduction of one order of magnitude in the cesium, strontium, and zirconium concentrations was based on a change in concentration observed during the Tank 5 chemical cleaning process. While there was uncertainty around the single process sample collected prior to the chemical cleaning process, comparison of the final measurements to those taken prior to chemical cleaning demonstrate good agreement between constituents not affected by the chemical cleaning process. Table RAI-INV-5.1 shows several constituents' concentrations prior to and following chemical cleaning. Since there is good agreement between concentrations of other constituents, there is confidence that the process sample could be used as an indication of the concentrations prior to chemical cleaning and, by extension, in the concentration changes of cesium, strontium, and zirconium.

Table RAI-INV-5.1: Comparison of Concentrations before and after Chemical Cleaning

		Before (WSRC-STI-2007-00192)		After (SRNL-STI-2009-00492)	
Constituent	**Units**	**Aqua Regia**	**Peroxide Fusion**	**Aqua Regia**	**Peroxide Fusion**
Fe	mg/kg	3.83E+05	3.63E+05	1.8E+05	2.1E+05
Hg	mg/kg	1.29E+03	-	1.5E+03	-
Ni	mg/kg	4.62E+04	4.27E+04	8.3E+04	8.8E+04
Am-241	mCi/kg	5.07E+01	-	6.82E+01	6.41E+01
Pu-238	mCi/kg	2.02E+00	1.88E+00	2.68E+00	2.88E+00
Pu-241	mCi/kg	8.37E+00	7.73E+00	1.27E+01	1.32E+01

Since the process sample was considered to provide a good indication of the concentrations prior to chemical cleaning, and since there was a significant decrease in the Cs-137, Sr-90, and zirconium concentrations, the inventories of cesium, strontium, and zirconium were decreased by one order of magnitude.

The waste tank cleaning process was planned to include a combination of mechanical cleaning and chemical cleaning, and this fact was assumed in developing the HTF PA inventory assignments, based on the closure plan at the time of HTF PA inventory development. In the future, even if chemical cleaning is not utilized on all or some of the waste tanks, it is still expected that whatever cleaning process is used would result in a level of cleaning similar to what was assumed in determining the HTF PA inventory assignments. Any expected increase in inventory assignments would be evaluated as part of the SA process prior to a final decision to move forward with waste tank stabilization.

Because the bases for the inventory assignments might not be consistent, care should be taken in comparing assigned inventories to final inventories. The assigned inventory value of Zr-93 in Tanks 5 and 6 was based on a minimum analytical detection limit, not actual material inventories. Therefore, it would not be appropriate to judge acid cleaning effectiveness on a comparison of inventories with different bases. This was the reason for evaluating the Tanks 5 and 6 sample concentrations before and after chemical cleaning.

With regards to the Tanks 5 and 6 comparisons to inventory assignments, it should be noted that the volume projection was less than that assigned for other HTF waste tanks (i.e., the HTF waste tanks had residual volume assignments of 4,000 gallons as compared to volume assignments of 1,700 to 2,200 gallons for the FTF waste tanks). In addition, the final residual volumes in Tanks 5 and 6 were higher than the originally assigned FTF PA volume values. Also, based on historical data and service life, Tanks 5 and 6 were projected to have the highest radionuclide concentrations, as compared to other Type I tanks. It is expected that future FTF Type I tanks will be shown to have lower radionuclide concentrations than those observed in Tanks 5 and 6.

Finally, as to the question concerning the intent to chemically clean waste tanks containing zeolite; the system plan (which was used as a basis for the HTF PA inventory estimates) assumed that the waste tanks that contain zeolite would be chemically cleaned. As discussed above, the overall dose impact of not chemically cleaning waste tanks containing zeolite is expected to be minimal should this plan change.

Waste Release and Near-Field Transport (NF)

RAI-NF-1

DOE should provide additional support for the implicit conclusion that the carbon steel liner in contact with chloride solution would remain essentially passive due to an assumption of limited availability of oxygen to support chloride-induced corrosion.

Basis

The documents on life estimation of high level waste tank steel (SRNL-STI-2010-00047; WSRC-STI-2007-00061, Rev. 2) include computations that are strongly dependent on CO_2 and O_2 diffusion coefficients. Long steel lifetimes are dependent on the concrete vault being an effective diffusive barrier (See RAI-NF- 2). The corrosion rates for carbon steel in contact with chloride solution used in the performance assessment are practically in the passive regime, due to the assumed diffusion transport resistance by the concrete.

According to Table 4.2-32 in the HTF Performance Assessment, DOE used general corrosion rates of the carbon steel activated by chloride in the range of 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 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 those corrosion rates corresponds to passive corrosion. As a comparison, a liner 0.5-in thick would corrode in 12,500 years at a rate of 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 results are almost independent of the magnitude of the oxygen diffusion coefficient, provided that it is less than 10^{-4} cm^2/s.

The steel liner failure times for various tank types for the performance assessment are summarized in Table 4.2-32 of the HTF Performance Assessment. As previously stated, for Type I and II tanks the assumed value of the oxygen diffusion coefficient was 10^{-4} cm^2/s, independently of the case (A, B, C, D, or E). For Type III/IIIA and IV tanks, DOE assumed a value of the oxygen diffusion equal to 10^{-6} cm^2/s, also independently of the configuration. The numerical distinction to the performance assessment results of selecting and oxygen diffusivity of 10^{-4} cm^2/s or 10^{-6} cm^2/s is negligible, as associated liner corrosion rates are computed by the DOE to be in the passive range (due to the assumption of oxygen diffusion control of the corrosion rate).

In the computation of corrosion rates controlled by oxygen diffusion (chloride-induced corrosion case) that are described in SRNL-STI-2010-00047 and WSRC-STI-2007-00061, Rev. 2, assumptions that corrosion is uniform and that the oxygen reduction reaction ($1/2\ O_2 + H_2O + 2e^- \rightarrow OH^-$) happens at the same location as the iron oxidation reaction ($Fe \rightarrow Fe^{2+} + 2e^-$) are implicit. In SRNL-STI-2010-00047, the possibilities of galvanic corrosion and macro-cell corrosion (due to different levels of water saturation and oxygen availability) associated with the separation of anodic and cathodic regions on the liner are recognized. In general, anodic areas could be much smaller than cathodic areas, potentially leading to liner regions undergoing faster corrosion than computed under the assumption of uniformly spread corrosion. Oxygen reduction could occur on other metal surfaces with accessible oxygen, such as the rebar or transfer lines (the liner is possibly in electrical contact with concrete vault rebar and transfer lines). The availability of oxygen for corrosion reactions should be higher above the water line for partially or fully immersed tanks. Also, the chloride concentration in the concrete vault could be higher near the water line, due to evaporation; thus contributing to differentiation between cathodic and anodic areas in the liner and rebar.

The implicit conclusion in the DOE analyses that the carbon steel remains in the passive state depends strongly on considerations and assumptions of effective diffusivity through cementitious materials over long time periods (see RAI-NF- 2), uniformity of the corrosion process, and space-point balance of cathodic and anodic reactions.

Path Forward

Provide a technical basis for the implicit conclusion that the carbon steel liner in contact with chloride solution would remain passive, especially since (i) concrete could degrade as a function of time, (ii) oxygen availability varies along metal surfaces especially for tanks partially or fully immersed, and (iii) it is not strictly necessary for oxygen reduction to occur at the same location where carbon steel actively corrodes.

Response RAI-NF-1

The general approaches and assumptions of the liner analysis are described in HTF PA Section 4.2.2.2.6 (Carbon and Stainless Steel Material Properties) and the basis for the liner analysis is captured in *Life Estimation of High Level Waste Tank Steel for H-Tank Farm Closure Performance Assessment*, SRNL-STI-2010-00047. As stated in Section 4.2.2.2.6 of the HTF PA, “degradation of the waste tank steel encased in grouted conditions was estimated due to carbonation of the concrete leading to low pH conditions, or the chloride-induced depassivation of the steel leading to accelerated corrosion. Chloride-induced corrosion was determined to be the more aggressive phenomenon.” The PA does not assume that other corrosion mechanisms are not possible, but given that early liner failure time cannot be assumed to be conservative (as discussed later in this response), the deterministic HTF liner analyses includes what is believed to be the most probable and defensible corrosion scenario.

As background to discussing the steel passivity, a differentiation between microcell and macrocell corrosion must be described. With respect to microcell corrosion, a microcell consists of pairs of immediately adjacent anodes and cathodes that are microscopic in size. This produces uniform corrosion that is the result of carbonation of the concrete or uniformly high chloride concentrations over a wide area within the vicinity of the steel. In contrast, in macrocell corrosion, macrocells normally occur in the case of chloride-induced corrosion. They consist of anodically acting areas normally where the critical chloride concentration has been reached and large cathodes where oxygen is reduced that can be some distance (up to several meters) from the anode. [ISSN: 0950-0618] Macrocell corrosion is the type of corrosion that RAI-NF-1 is addressing.

The action of macrocells in buried structures is different from those that are exposed to the atmosphere. Two circumstances promote macrocell effects, while another mitigates the effect. First, the concrete is wetter in the buried structure than in the atmosphere. As a result, the conductivity of the concrete is higher in the buried structure, which decreases the ohmic drop and hence increases the cathodic area available for oxygen reduction and high corrosion currents result. Secondly, the soil surrounding the concrete has a higher hydraulic conductivity than the concrete. This allows the current to flow outside the concrete vault, which further reduces the ohmic drop. However, the mitigating aspect may have a greater effect. Oxygen diffuses with great difficulty through wet concrete and thus the rate at which it reaches the steel is greatly retarded. Depletion of oxygen at the surface causes the corrosion rate to decay to minimal values, even when it is initiated.

Another aspect that protects the liner is the thickness of the vault. Macrocell corrosion decreases as the thickness of the vault increases. [ISSN: 0950-0618] Thus, until the vault degraded to the point that there is less than a few inches of intact concrete, the liner would likely not experience macrocell corrosion.

Macrocell corrosion can occur when there are macroscopic defects in the concrete (cracks with large width, honeycombs, delamination, etc.). [ISBN: 3-527-30800-8] In this case, there is not only direct contact with the steel, but potentially more severe attack caused by galvanic coupling with passivated steel that is embedded in the concrete (note, passivated steel and active carbon steel result in a higher galvanic current than a stainless steel/active carbon steel corrosion rate). The increase in corrosion rate due to galvanic coupling is typically modest and can only become appreciable if the surface area of the cathode is much greater than the surface area of the anode. A significant macrocouple can only arise under very particular conditions. For example, the cracked area where the groundwater is in contact with the steel is de-aerated and the passivated steel area is exposed to aerated conditions. To account for the effect of galvanic corrosion, waste tanks and transfer lines in soils that contain groundwater are assumed to decay with higher rates of pitting and general corrosion. [SRNL-STI-2010-00047] However, because the concrete is essentially saturated, significant oxygen gradient would not be anticipated and the modeling approach used is reasonable.

The higher corrosion rates associated with the groundwater were addressed in the modeling of liner degradation within the comprehensive stochastic model. The results of the model, including the higher general corrosion rates for soils with groundwater, are shown in Table 4.2-31 of the HTF PA. The distributions of general corrosion rates with groundwater and without groundwater are compared in Figures 48 and 49 of *Life Estimation of High Level Waste Tank Steel for H-Tank Farm Closure Performance Assessment,* SRNL-STI-2010-00047. The corrosion rates assumed in soils with groundwater resulted in a significant decrease in the life expectancy of the waste tank liner. The maximum corrosion rate assumed in the distribution is compatible with the maximum general corrosion rate in soils with groundwater. [SRNL-STI-2010-00047]

A liner sensitivity analysis is provided in Section 5.6.7.6 (Liner Failure Times Analysis using the PORFLOW Deterministic Model) of the HTF PA. To study the impact of liner failure-time variability, a sensitivity study was performed using the PORFLOW deterministic model. To capture the impact of the time of liner failure on dose, 12 different flow fields reflecting different liner failure times were utilized. The PORFLOW deterministic model was run using the HTF PA Base Case (Case A) assumptions except that different flow fields were simulated reflecting four different liner failure times (immediate, early, moderate, late), as shown in the HTF PA, Table 5.6-47. The impact of liner failure time variability is displayed in the HTF PA, Figures 5.6-80 through 5.6-91. As shown in these PA figures (included here as Figures RAI-NF-1.1 through RAI-NF-1.12), there is no direct correlation between early liner failure and an increase in the peak dose.

Figure RAI-NF-1.1: No Fast Flow Path Type I Tank MOP Dose (Base Case)

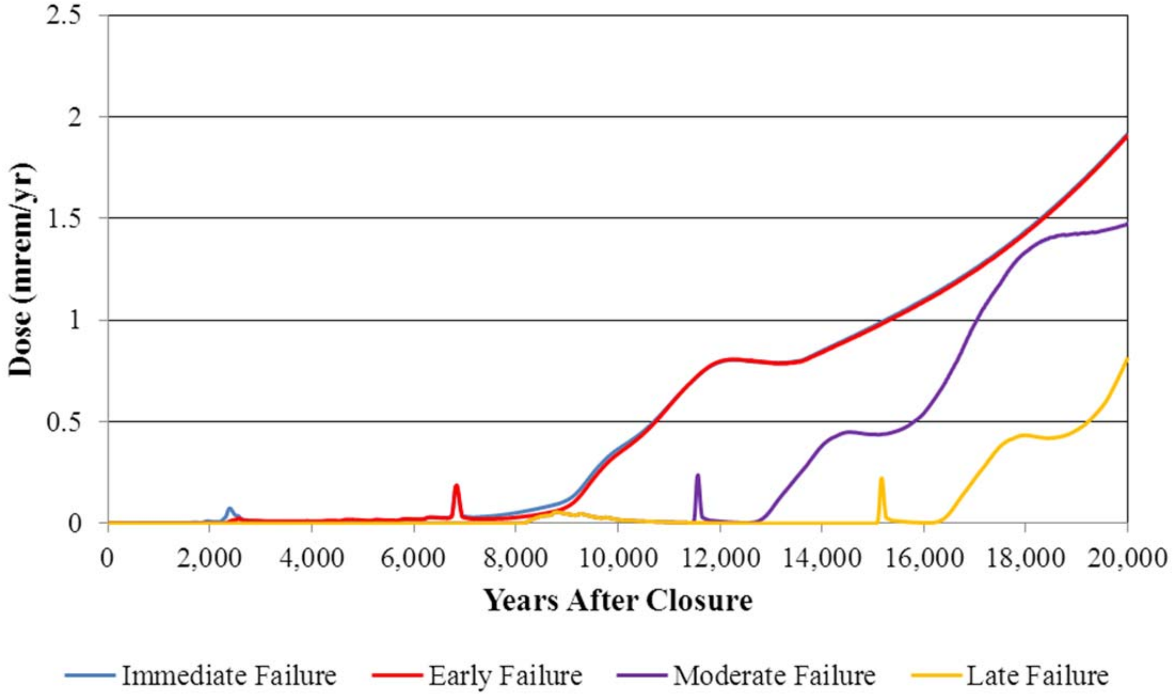


Figure RAI-NF-1.2: No Fast Flow Path Type II Tank MOP Dose (Base Case)

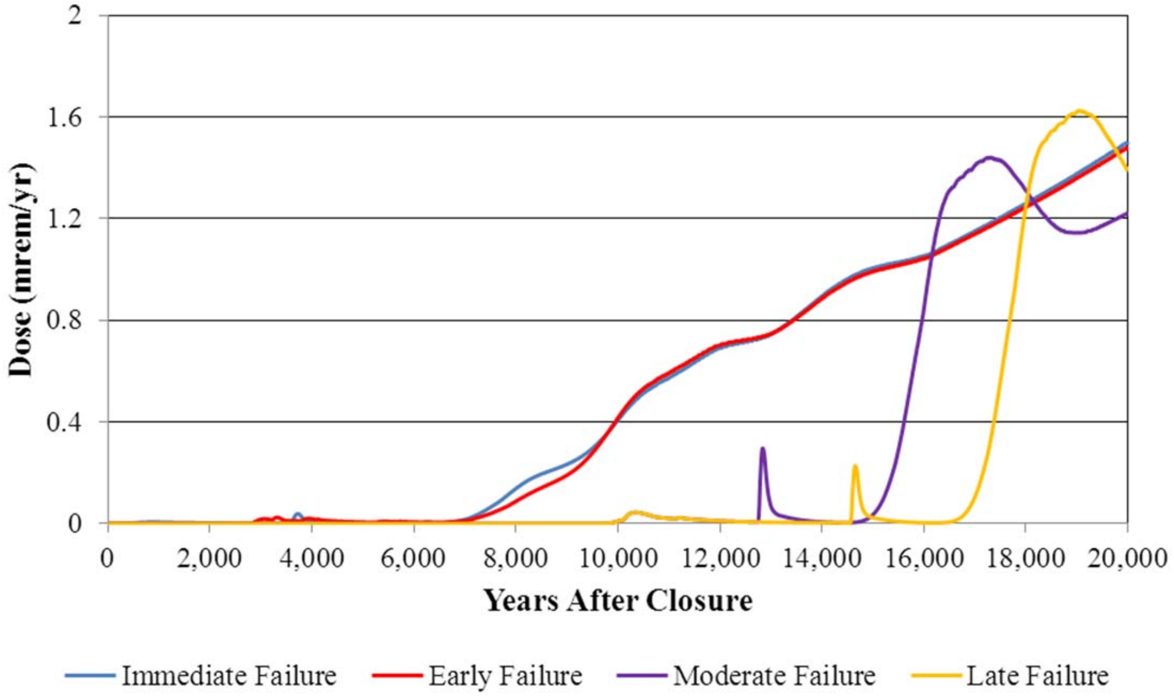


Figure RAI-NF-1.3: No Fast Flow Path Type IV Tank MOP Dose (Base Case)

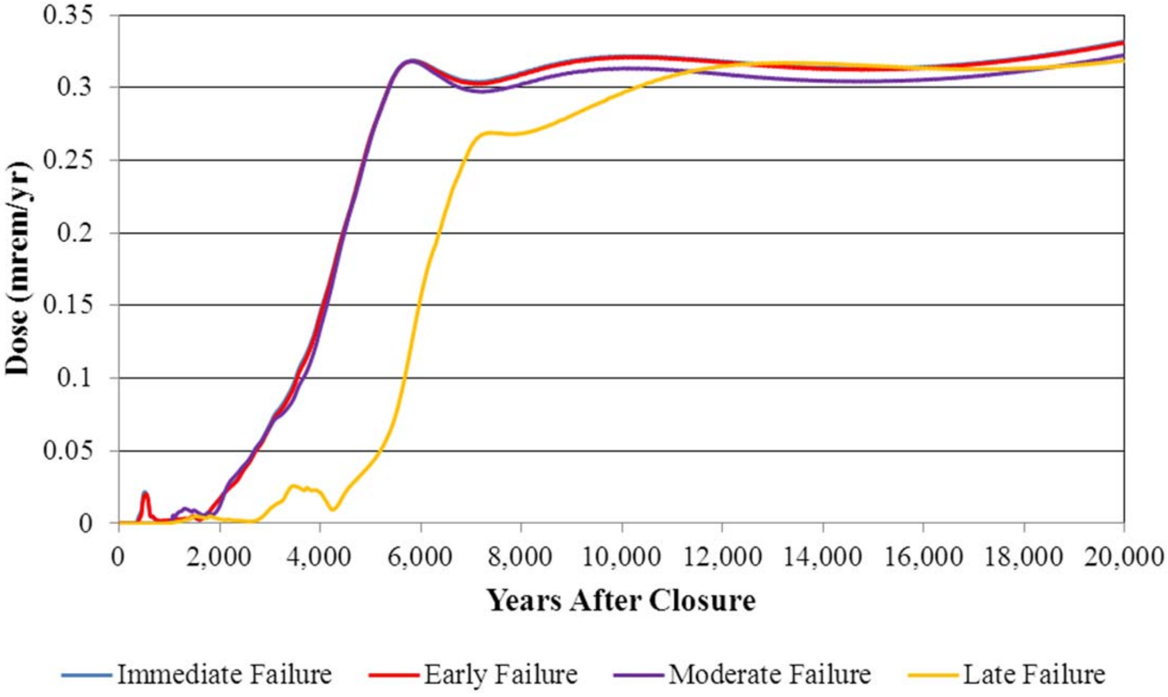


Figure RAI-NF-1.4: No Fast Flow Path Type III / IIIA Tank MOP Dose (Base Case)

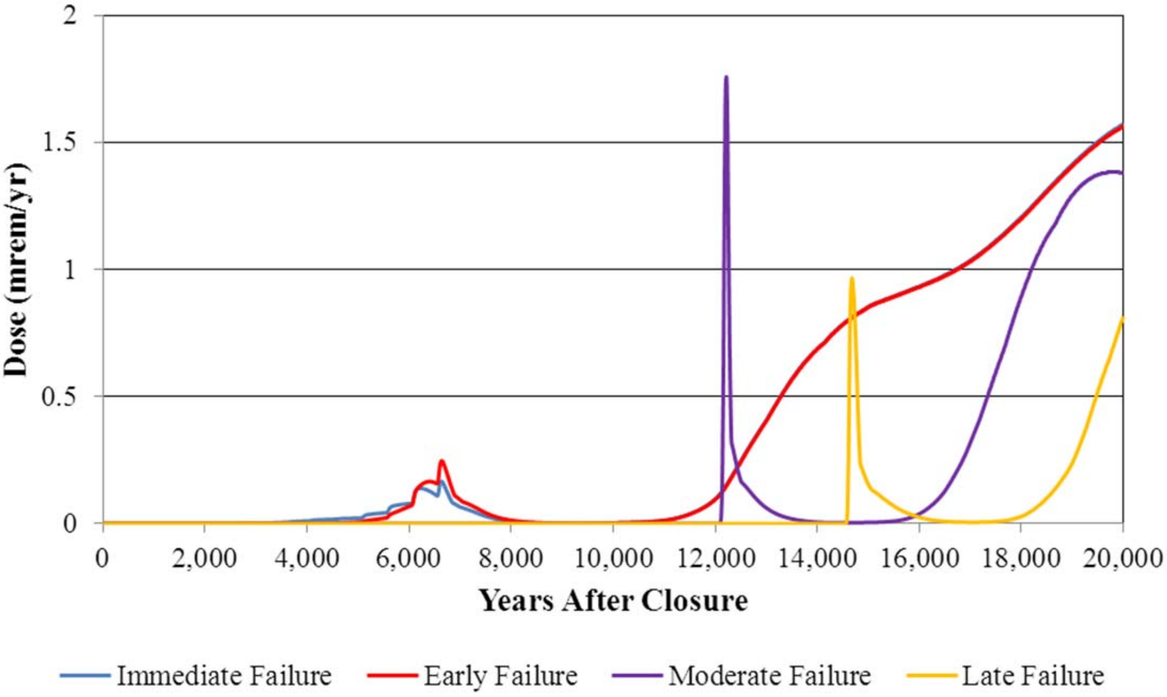


Figure RAI-NF-1.5: Partial Fast Flow Path Type I Tank MOP Dose (Cases B and C)

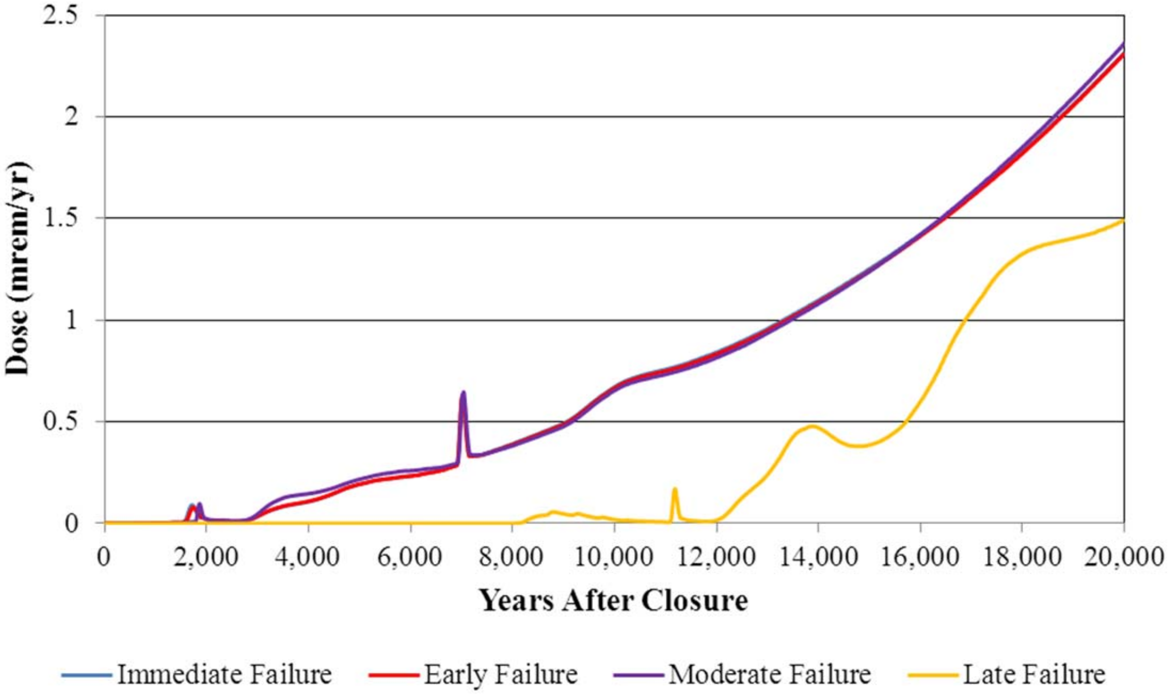


Figure RAI-NF-1.6: Partial Fast Flow Path Type II Tank MOP Dose (Cases B and C)

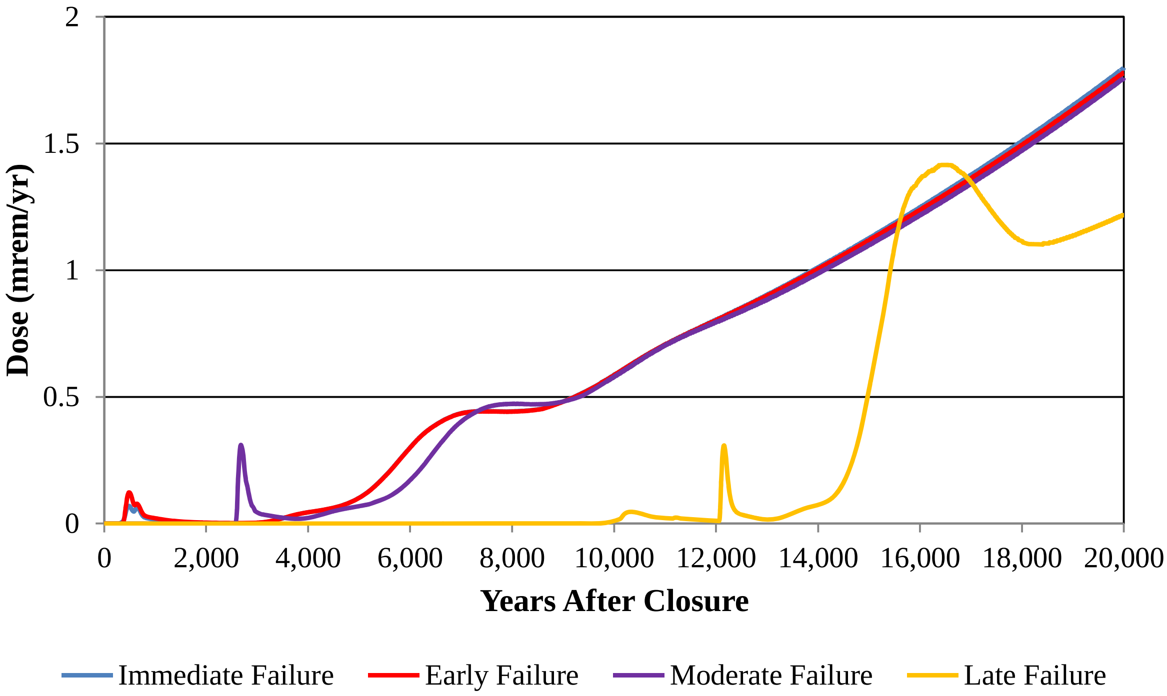


Figure RAI-NF-1.7: Partial Fast Flow Path Type IV Tank MOP Dose (Cases B and C)

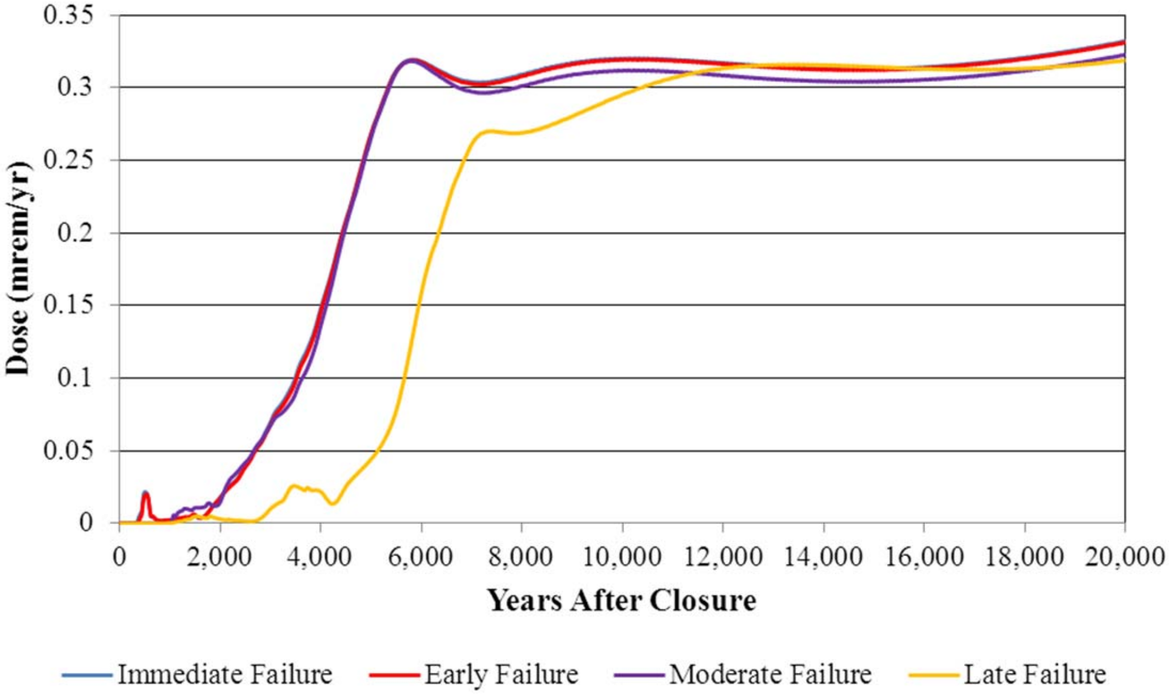


Figure RAI-NF-1.8: Partial Fast Flow Path Type III / IIIA Tanks MOP Dose (Cases B and C)

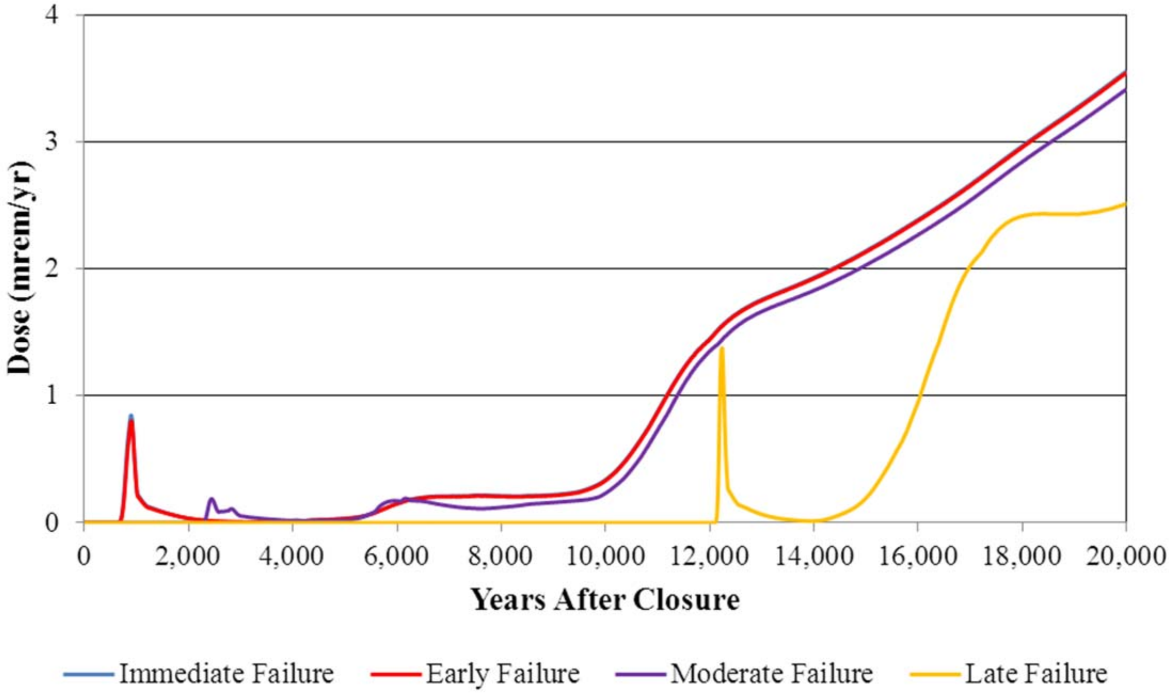


Figure RAI-NF-1.9: Full Fast Flow Path Type I Tank MOP Dose (Cases D and E)

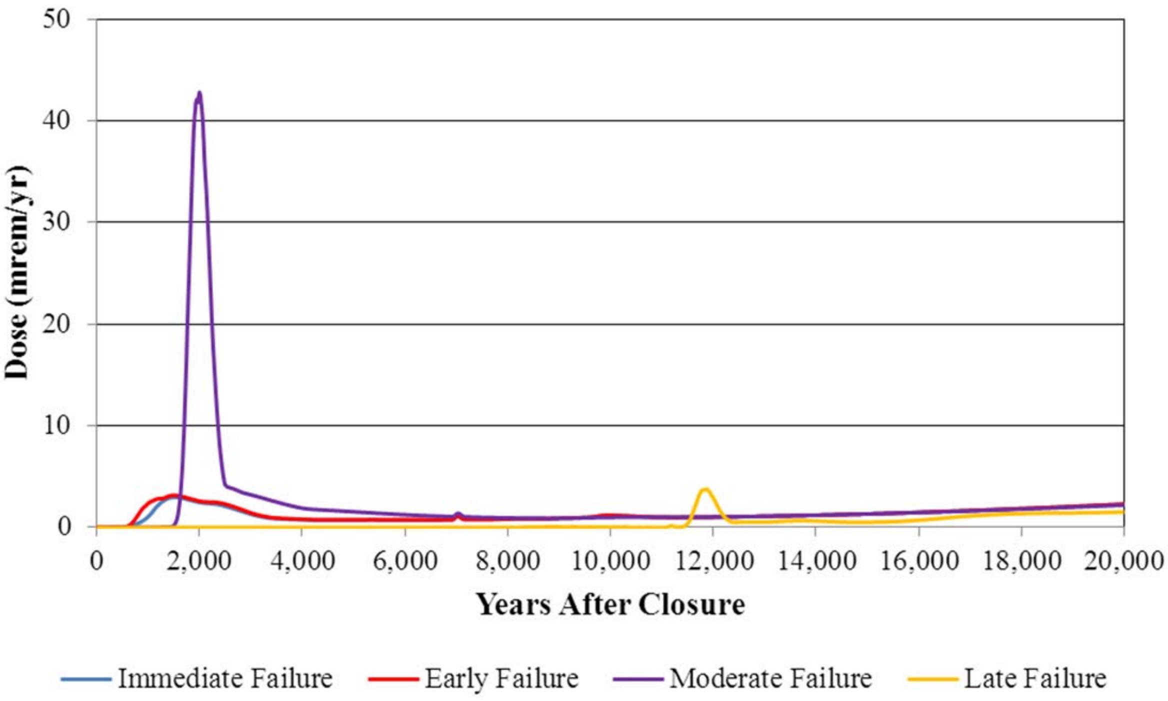


Figure RAI-NF-1.10: Full Fast Flow Path Type II Tank MOP Dose (Cases D and E)

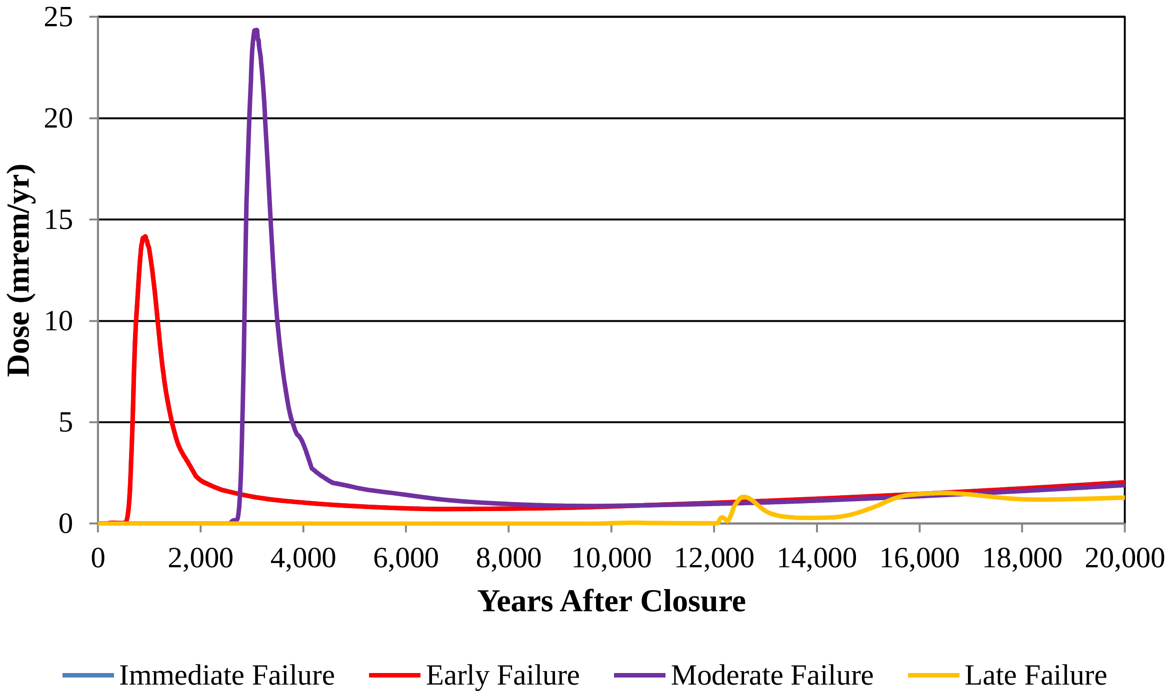


Figure RAI-NF-1.11: Full Fast Flow Path Type IV Tanks MOP Dose (Cases D and E)

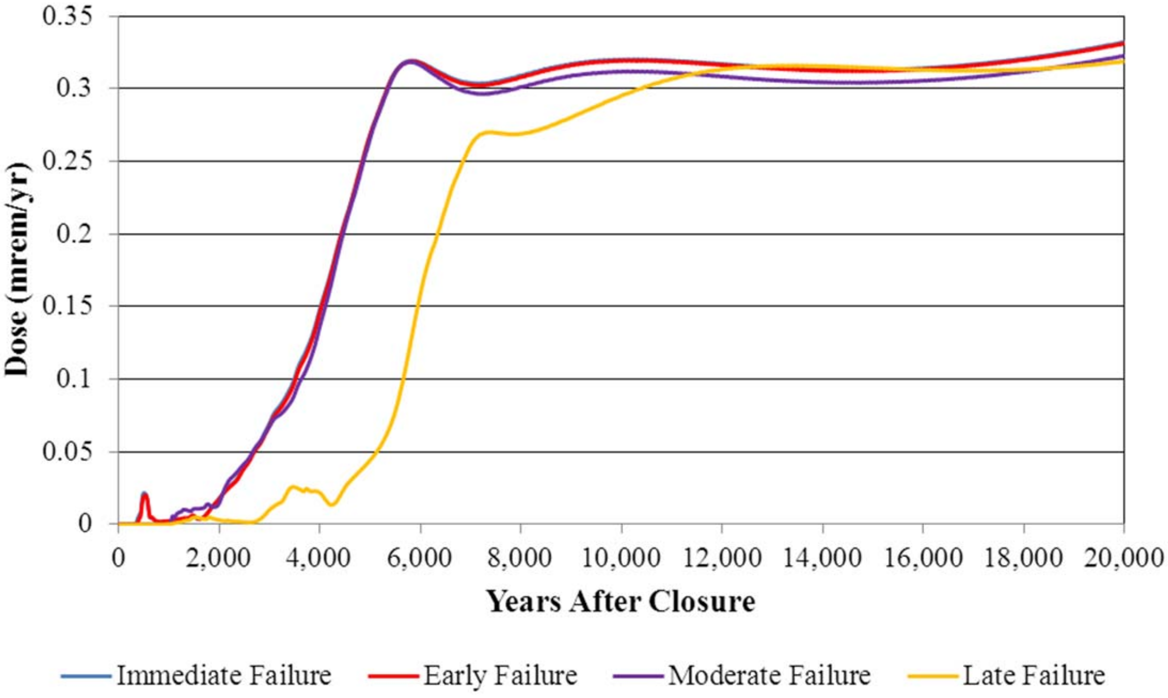
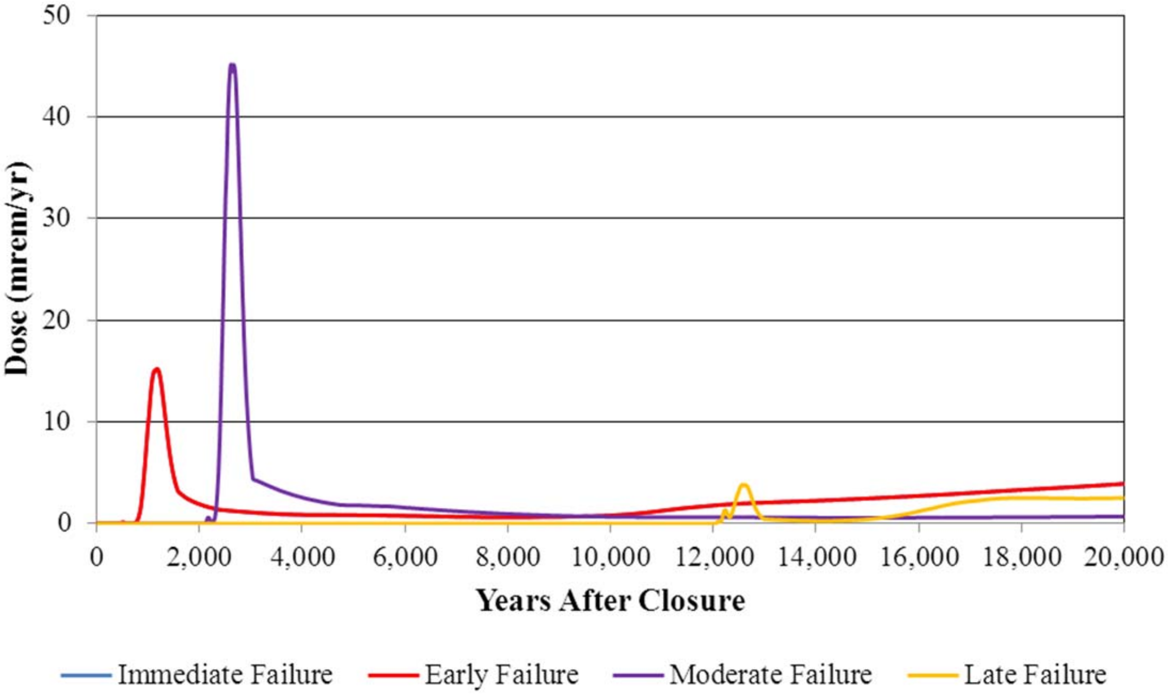


Figure RAI-NF-1.12: Full Fast Flow Path Type III / IIIA Tank MOP Dose (Cases D and E)



RAI-NF-2

Provide a technical basis for the assumption that the diffusion coefficients for carbon dioxide and oxygen that are used for steel liner lifetime estimates appropriately bound the permeability changes in concrete due to carbonation.

Basis

To estimate steel liner lifetimes in the HTF Performance Assessment base case (i.e., Case A), DOE assumes effective diffusion coefficients for carbon dioxide [$D(CO_2)$=1E-6 cm^2/s] and oxygen [$D(O_2)$=1E-4 cm^2/s (Type I/II); $D(O_2)$=1E-6 cm^2/s (Type III/IIIA/IV)] through cementitious materials. On page 317 of the HTF Performance Assessment, DOE indicates that diffusion coefficients are typically calculated or measured to be approximately 1.0E-8 cm^2/s.

Rebar corrosion is known to crack concrete and cause spallation, due to the volume expansion of corrosion products. Literature information relates the timing of cracking, due to carbonation, and the extent of cracking to the rebar cover. The rebar cover is the thickness of the concrete on top of the rebar. It is not clear how the analysis in SRNL-STI-2010-00035, Rev. 0 relates to information in the literature linking cracking to the cover thickness.

Analyses in the literature relate cracking of reinforced concrete to the cover thickness (e.g., Parrot, 1990; Andrade et al., 1993; Molina et al., 1993; Neville, 1996). The rebar cover in the reinforced concrete vaults are expected to be a few centimeters of concrete. Parrott (1990) states that cracks are not expected when the carbonation depth is less than one-half of the cover, but significant rebar corrosion, cracking and spallation can occur when the carbonation depth is in excess or well in excess of the cover thickness.

Cracks would affect permeability of concrete and the diffusivity of CO_2 through the system. Enhanced diffusivity of CO_2 would enhance the penetration rate of the carbonation front, which in turn would activate corrosion of deeper rebar, and potentially cause additional cracking of the concrete. Rebar could also actively corrode due to the presence of chloride in the groundwater.

Information is needed on whether a gradual rebar corrosion process and cracking of the concrete vault would affect significantly affect permeability of the concrete vaults thereby potentially reducing steel liner lifetime estimates (SRNL-STI-2010-00035, Rev. 0) used in the HTF Performance Assessment.

Path Forward

Provide a technical basis for the diffusion coefficients for carbon dioxide and oxygen assumed for steel liner lifetime projections. The basis should discuss whether the coefficients appropriately account for permeability changes of the concrete due to carbonation, especially since rebar can be located close to the vault surface and can exhibit enhanced corrosion rates due to carbonation at an earlier time.

Response RAI-NF-2

Intrinsic diffusion coefficients were used to model the depth of penetration for carbon dioxide and the flux of oxygen through the concrete matrix. [NUREG/CR-6070, NUREG/CR-5542] The intrinsic diffusion coefficient combines the effect of tortuosity, porosity and the diffusion coefficient of the species in water. Given that the concrete vault is subsurface, the pores of the concrete remain saturated, and therefore diffusion of carbon dioxide and oxygen are primarily in the liquid phase.

The water-cement ratio (WCR) of the concrete determines the porosity and permeability of the concrete matrix. The intrinsic diffusion coefficient as a function of WCR was empirically determined to range between 1E-07 and 1E-09 cm^2/s. A bounding diffusion rate of 1E-06 cm^2/s was selected for both carbon dioxide and oxygen for a non-submerged (i.e., above the water table) concrete vault. For submerged concrete vaults, the diffusion coefficient for oxygen was increased to 1E-04 cm^2/s. The result is a higher corrosion rate than observed in the grout.

The degradation analysis does not intrinsically assume that the diffusion coefficient will increase in a submerged structure. It is more likely that the penetration of the oxygen would be limited due to less oxygen availability compared to the non-submerged vault. What is expected to increase the penetration of oxygen is the presence of cracks. In the concrete chemical degradation assessment, *Chemical Degradation Assessment for the H-Area Tank Farm Concrete Tanks and Fill Grouts,* SRNL-STI-2010-00035, the permeability of the concrete is expected to increase by a factor of approximately 100 because of cracking. Thus if the penetration depth is increased by a factor of 100, the equation that relates penetration to the diffusion coefficient would indicate that the diffusion coefficient would increase by a factor of 10,000. Assuming this 1E+04 factor increase, the 1E-04 cm^2/s coefficient is a reasonable estimate for a vault with cracks (i.e., based on 1E-08 cm^2/s being a reasonable estimate for the diffusion coefficient in uncracked concrete) independent of the rebar location.

RAI-NF-3

The technical basis should be enhanced for the estimated longevity of reducing conditions that is important to the retention of redox-sensitive radionuclides in the waste tanks.

Basis

Reducing conditions are assumed in the HTF Performance Assessment to provide a significant chemical barrier to the release of redox-sensitive radionuclides. The longevity of reducing conditions is dependent on the ability of reduced sulfur species in the blast furnace slag to react with dissolved oxygen in the infiltrating water. The HTF Performance Assessment utilized Geochemist's Workbench to estimate the longevity of reducing conditions. The simulation used the mineral pyrite (FeS_2) to account for the reducing capacity of the grout (SRNL-STI-2012-00404, Rev. 0).

The lack of empirical data supporting the selection of pyrite to represent the grout reducing capacity means the calculated E_h transition time is uncertain. Recent research indicates that sulfide phases present in blast furnace slag might not be adequately accounted for by the use of pyrite as a proxy, as discussed in ML12272A082. PNNL-21723 reported observed reduced sulfur species in leachates from saltstone simulant experiments that also utilized blast furnace slag to impart reducing conditions. If the reduced sulfur phase(s) in blast furnace slag is more soluble than pyrite, the geochemical modeling in the HTF Performance Assessment would overestimate the longevity of reducing conditions in the grout.

Path Forward

DOE should provide additional support for the assumption that pyrite is a reasonable proxy in geochemical modeling for representing the reducing capacity. Additional support could include laboratory studies conducted to determine the reducing capacity that may be removed by leaching.

Response RAI-NF-3

Pyrite is used in the waste release model to account for reducing capacity in the waste tank grout. The mass of pyrite in the initial mineralogy used in the model was calculated based on the measured reducing capacity in the blast furnace slag component of the grout formulation. A fundamental assumption of the model is that the waste tank grout is fully hydrated and approaches chemical equilibrium. This basis for this assumption is that the grout is assumed to be in a saturated environment (near 100 % relative humidity) for hundreds to thousands of years prior to the failure of the waste tank liner. Under these conditions, the reduced sulfur responsible for much of the reducing capacity in the blast furnace slag will be incorporated into relatively stable phases in the grout.

Prior to grout hydration, reduced sulfur in the blast furnace slag may be more soluble than pyrite. Nevertheless, reduced sulfur is highly reactive and when it is released from the slag during hydration, it will react with iron and potentially other metals to form metal sulfides. Iron is the most abundant of these metals in the waste tank grout. Based on the waste tank grout formulation and the chemical composition of each of the components, there are more than 20,000 g/m^3 of Fe_2O_3 or 254 mol/m^3 of iron in the reducing grout. Much of this iron is attributed to the fly ash component of which a substantial portion is likely reduced iron in the form of magnetite (documented in *Petrology, Mineralogy, and Chemistry of Magnetically-Separated Sized Fly Ash,* S0016-2361(98)00132-X) or pyrite (documented in *Analysis of Fly Ash by X-Ray Emission Spectroscopy and Proton Microbeam Analysis,* 0016-2361(84)90337-5). The mass of pyrite used in the initial mineralogy of the waste release model was 816 g/m^3 or 13.6 mol/m^3 of reduced sulfur. Hence, there is more than enough iron in the waste tank grout to form relatively low-solubility reduced sulfur minerals by reaction with the reduced sulfur released from the hydration of blast furnace slag.

Hanford Site waste studies, *Equilibrium Solubility Model for Technetium Release from Saltstone Based on Anoxic Single-Pass Flow through Experiments,* PNNL-21723, have observed reduced sulfur species in leachate from saltstone simulant experiments. However, based on calculated saturation indices, they also suggested that pyrrhotite ($Fe_{1-x}S$) was the mineral phase controlling the reduced sulfur concentrations. Pyrrhotite, like pyrite, is an iron sulfide with a relatively low solubility. Regardless, the saltstone samples used in the Hanford Site experiments were cured for 30 days. Much longer periods of curing time are required to fully hydrate blast furnace slag. It was observed in *Internal Chemical Evolution of the Constitution of Blended Cements,* 0008-8846(88)90042-7, that only 60 % of blast furnace slag was hydrated after two years in an ordinary Portland cement/blast furnace slag sample (using a 70:30 mix ratio). Thus, release of soluble forms of reduced sulfur from blast furnace slag to the leachates of the Hanford Site experiments would not be unexpected. In a system in which the reduced sulfur was in contact with iron for a long period of time (e.g., a closed waste tank prior to liner failure), the sulfur would likely react with iron to form precursors of pyrite that would recrystallize to pyrite over time. Given the period of hundreds to thousands of years, it is reasonable to assume that the iron would react with reduced sulfur species released from blast furnace slag to form pyrite.

It is also important to note that use of pyrite to account for the reducing capacity in waste tank grout does not produce a significantly different result than if oxygen in infiltrating water were reacted stoichiometrically with the equivalents (eq) of reducing capacity. In the waste release model document, *Evolution of Chemical Conditions and Estimated Solubility Controls on Radionuclides in the Residual Waste Layer During Post-Closure Aging of High-Level Waste Tanks,* SRNL-STI-2012-00404, one cubic meter of grout contained 102 equivalents of reducing capacity. The influxing water contained an oxygen concentration of 2.19E-04 mol/L or 8.76E-04 eq/L. A pore volume in one cubic meter of grout contained 213 liters, yielding 0.19 equivalents of oxidation capacity per pore volume. Assuming stoichiometric reaction of the oxidizing equivalents in each pore volume of influxing water with the reducing equivalents of the grout, it would take 537 pore volumes (102/0.19) of influxing water to exhaust the reducing capacity. In the waste release model, the reducing capacity was exhausted in 523 pore volumes. [SRNL-STI-2012-00404] Hence, the use of pyrite to account for reducing capacity results in a timeframe for degradation of reducing conditions to oxidizing conditions that is not appreciably different from the commonly used stoichiometric approach.

To demonstrate the potential impact on dose due to uncertainty in the pore volumes required to exhaust the reducing capacity of the grout, a series of sensitivity runs were performed using the GoldSim model in deterministic mode. This non-mechanistic study artificially varied the number of pore volumes required for transitions by applying a multiplier to the pore volume value. No other modeling changes were made relative to the Base Case (Case A). Table RAI-NF-3.1 provides a summary of the sensitivity runs and the resulting peak doses. Figure RAI-NF-3.1 shows that varying the pore volume value effectively shifts the timing and magnitudes of the peak doses.

Table RAI-NF-3.1: Peak MOP Doses from Pore Volume Sensitivity

Pore Volume Sensitivity Modeling Runs									
Pore Volume Multiplier	0.05	0.1	0.25	0.5	1.0	2.0	5.0	10	20
Resulting Pore Volume Transition	26.2	52.3	131	262	523	1,046	2,615	5,230	10,460
Peak MOP Doses (mrem/yr)									
0 - 1,000 Years	0.52	0.52	0.52	0.52	0.51	0.51	0.51	0.51	0.51
0 - 10,000 Years	2.1	2.2	2.6	3.0	3.8	1.2	1.2	1.2	1.3
0 - 20,000 Years	20.0	21.3	20.3	13.0	12.3	11.4	13.0	35.0	16.4

Figure RAI-NF-3.1: Total MOP Doses from Pore Volume Sensitivity

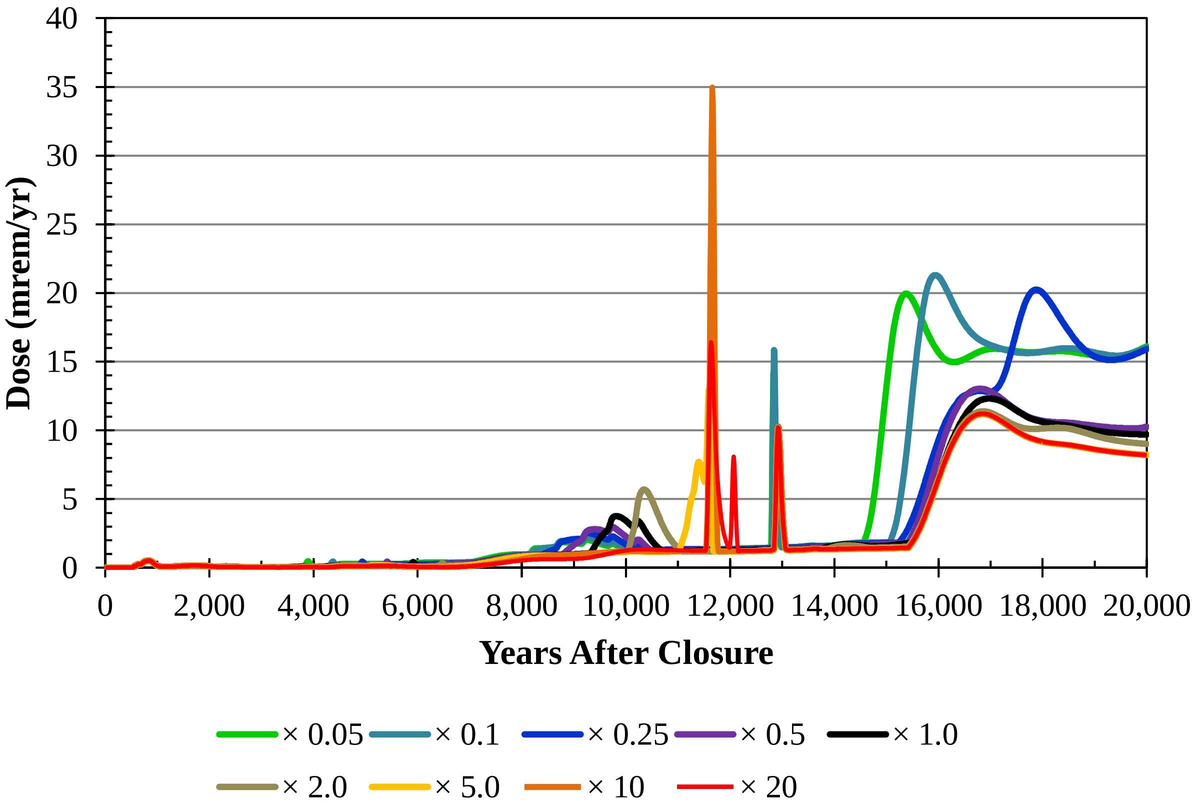


Table 7.1-2 of the HTF PA indicates "…emphasis was placed on minimizing uncertainty associated with doses within the initial 10,000 years after HTF [final facility] closure." Consistent with this approach, the Base Case run (i.e., pore volume multiplier equal to 1) resulted in the highest peak dose within the 10,000-year period. It should be noted that the most significant impacts only occur in the most bounding modeling runs (e.g., pore volume multipliers greater than 5), and even then the relatively high peak doses only occur beyond 10,000 years when multiple waste tank liners fail simultaneously. Further, only one of these runs (i.e., pore volume multiplier equal to 10) exceeded the 25 mrem/yr performance objective.

Also note that changing the number of pore volumes required for transition does not have a linear impact on the timing of the dose peaks (e.g., reducing the number of required pore volumes to 0.05 times the Base Case value does not move the 20,000 year peak dose into the first 1,000 years). As shown in Figure RAI-NF-3.1, the peak value over 20,000 years actually decreases from the ×10 to the ×20 multiplier. The lower peak value for the ×20 simulation, relative to the ×10 simulation, is associated with the relationship of the timing of the failure of the primary and secondary liners and the timing of the annulus K_d transitions in Tanks 9, 10, and 11. Because the annulus contains reducing grout, the timing of the transition from Reduced Region II to Oxidized Region II in the annulus of a Type I tank becomes a strong controlling factor in the peak values for the ×10 and ×20 simulations. When the annulus transition time (11,480 years after final facility closure) occurs shortly after the primary liner failure time (11,397 years after final facility closure), as in the ×10 simulation, a single peak occurs. When the annulus transition time occurs later (11,880 years after closure), a double peak occurs where the first peak reflects the initial release due to the primary and secondary liner failures and the second peak reflects the rapid release of the desorbed mass at the transition time. The higher single peak in the ×10 simulation is indicative of desorbed mass being released at the same time as the initial surge of mass flushed out by the occurrence of the liner failures.

RAI-NF-4

The technical basis for the ratio of 90% unconditioned groundwater to 10% grout conditioned water contacting the contaminated zone for conditions B, C, and D for submerged and partially submerged tanks throughout the period of performance is not well supported.

Basis

The HTF Performance Assessment assumes that the water contacting the contaminated zones in the submerged and partially submerged tanks consists of 90% unconditioned groundwater and 10% conditioned water that has migrated through the reducing grout. The assumption of 10% mixing of conditioned water provides a significant chemical barrier to radionuclide release. The basis for the 90:10 ratio approximates the modeled lateral-to-vertical groundwater flow velocities. However, the presence of engineered barriers at the HTF challenges this assumption.

The closure cap and the grouted tanks both decrease the likelihood that 10% of the water contacting the contaminated 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 of 10% of the water that contacts the contaminated zone will be conditioned by the overlying grout. It is not clear to NRC staff that overlying grout will be able to condition water migrating through preferential pathways within the tanks.

Path Forward

Provide additional technical bases for assuming that the water contacting the residual waste will consist of at least 10% water that is conditioned by the grout under closed conditions. DOE should consider the impact of time-dependent degradation of engineered barriers on this assumption. Alternatively, DOE could assess the sensitivity of the results to this assumption (e.g., assume a bounding condition of 100% of the water contacting the contaminated zone is unconditioned by the overlying grout).

Response RAI-NF-4

DOE does not believe that expected waste tank conditions after FTF final facility closure will permit the water that contacts the residual waste to be 100 % unconditioned. While the assumption regarding the percentage of water that is conditioned by the reducing grout under closed conditions was not based on waste tank-specific testing there are multiple waste tank conditions that support the assumption that the water interacting with the contamination zone is conditioned.

- In addition to reducing grout overlying the contamination zone, the waste tank annulus is also filled with reducing grout such that reducing grout would be present between preferential pathways and the contamination zone.
- Conditioning is a chemical process, not a mechanical process. While the contamination zone is modeled as being a discrete zone adjacent to the overlying reducing grout, in actuality the waste tank residuals will be in direct contact with the reducing grout on the waste tank floor. The residual waste will be chemically impacted by reducing grout properties (e.g., through diffusion) regardless of whether water travels though reducing grout to reach the contamination zone.

- Any preferential pathways that might exist are not expected to traverse the entire waste tank structure and therefore would not result in lateral pathways that accelerate flow across the waste tank boundary.
- The waste residuals are distributed across the entire waste tank bottom; even if a preferential pathway did manage to cause a localized infusion of unconditioned water, it would not cause the entire contamination zone to be impacted. The localized impact would potentially change the release rate for just a portion of the contamination zone, but changes of this type would tend to decrease the peak dose by causing contaminants to be released at different rates across the contamination zone instead of at a constant rate.
- Low reducing grout hydraulic conductivities early in the performance period would limit inflow of water laterally into the waste tank, ensuring that downward infiltrating flow continues to influence the contamination zone chemistry.
- Even with the HTF closure cap in place, there will be infiltration in the HTF. This infiltration will result in downward flow within the vicinity of the waste tanks, with the flow increasing over time as engineered barriers associated with HTF degrade.

Solubility Sensitivity Analyses

Varying the ability of the overlying reducing grout to condition the contamination zone would have an analogous impact as varying the speed at which the grout transitioned for reduced to oxidized conditions. To study the impact of reducing grout transition-time variability, a sensitivity study was performed using the PORFLOW Deterministic Model in HTF PA Section 5.6.7.2 (Grout Transition Time Analysis using the PORFLOW Deterministic Model). To simulate the possible effect that varying the waste-tank grout transition time might have on dose results, the waste-tank reducing grout transition times were modified so both earlier and later reducing grout transition times were used. The PORFLOW Deterministic Model was run for the Base Case (Case A) and for Case C using the updated reducing grout transition times. In addition to the expected grout transition time, a grout transition time of 0.5 and a grout transition time of twice what is expected were studied. The impact of grout transition time variability on dose for the Base Case and for Case C is displayed in HTF PA Figure 5.6-64 through Figure 5.6-69, which are included here as Figure RAI-NF-4.1 through Figure RAI-NF-4.6.

Figure RAI-NF-4.1: Base Case Grout Transition Time Study

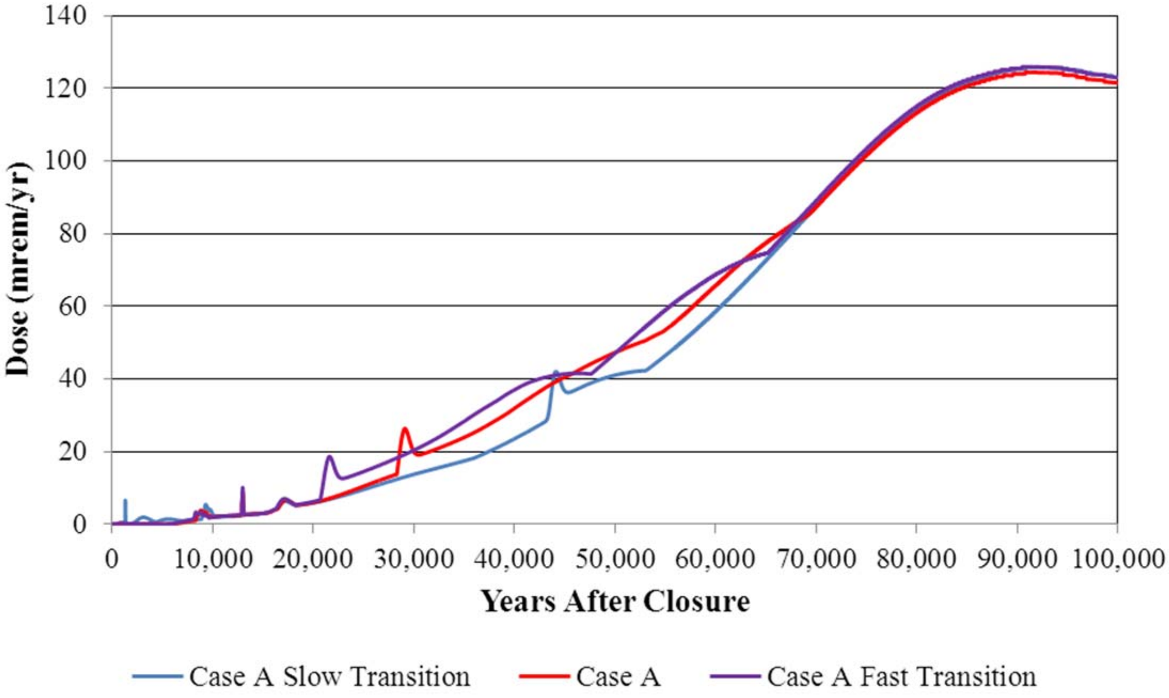


Figure RAI-NF-4.2: Case C Grout Transition Time Study

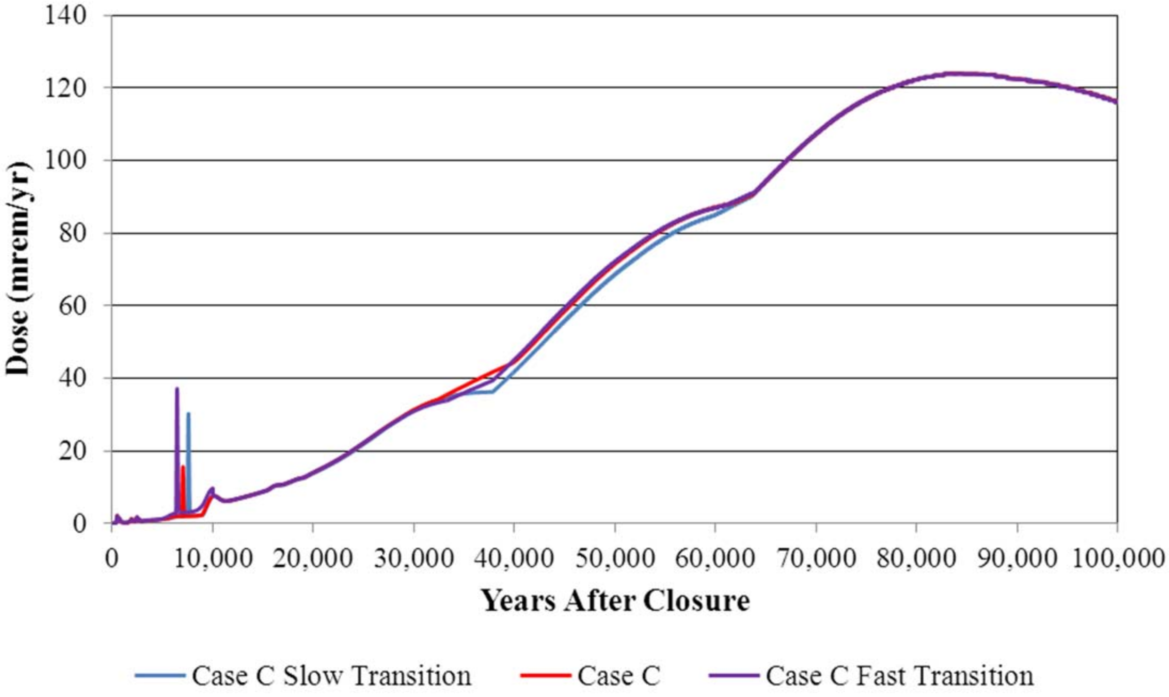


Figure RAI-NF-4.3: Individual Radionuclide Contributors to MOP Dose for Base Case Fast Grout Transition Time

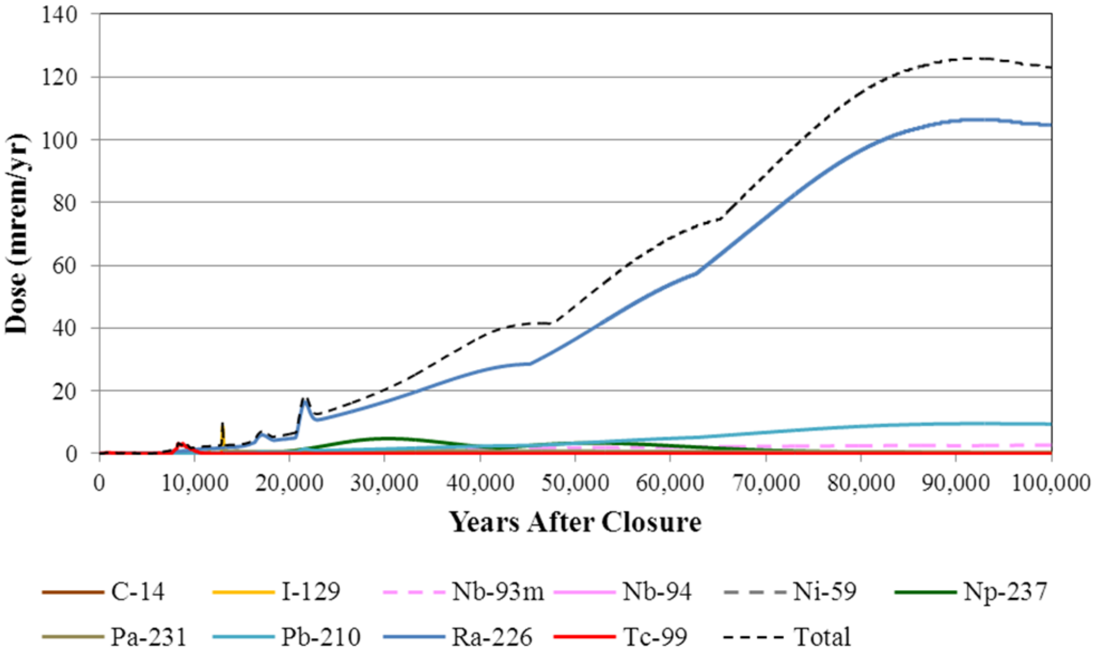


Figure RAI-NF-4.4: Individual Radionuclide Contributors to MOP Dose for Base Case Slow Grout Transition Time

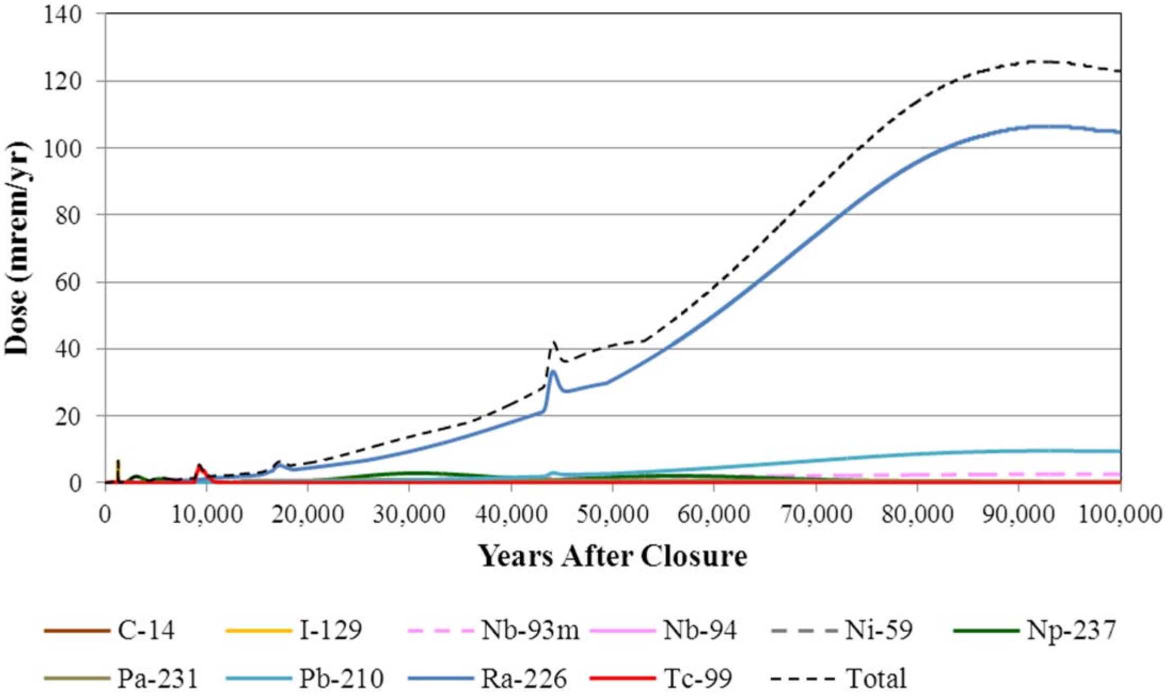


Figure RAI-NF-4.5: Individual Radionuclide Contributors to MOP Dose for Case C Fast Grout Transition Time

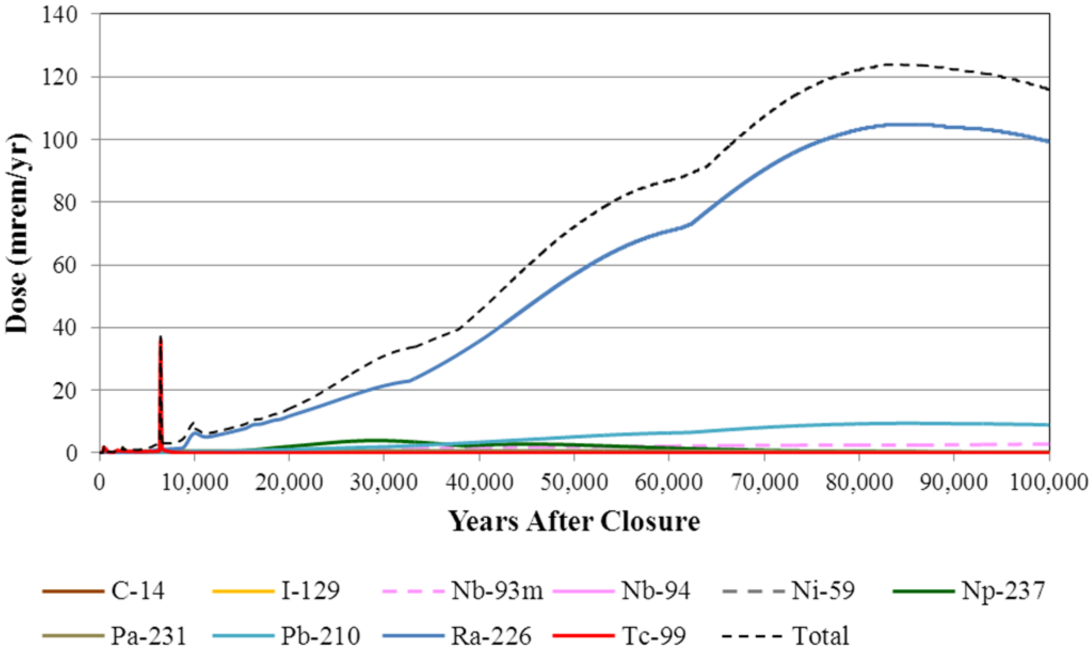
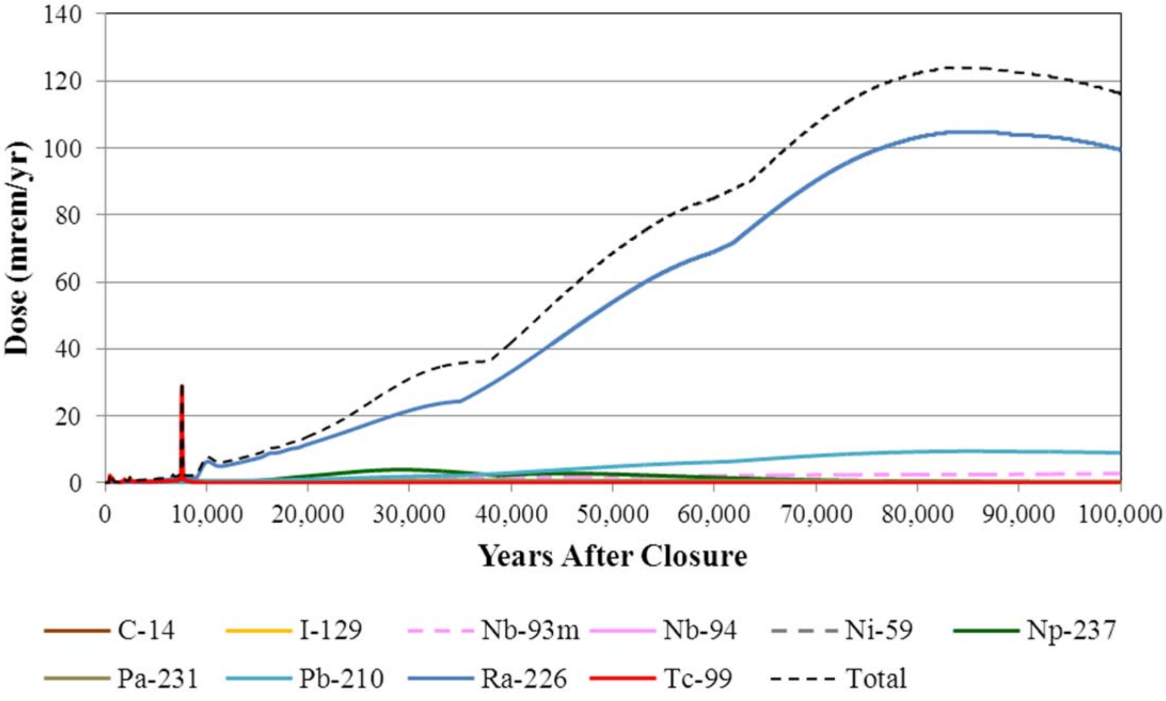


Figure RAI-NF-4.6: Individual Radionuclide Contributors to MOP Dose for Case C Slow Grout Transition Time

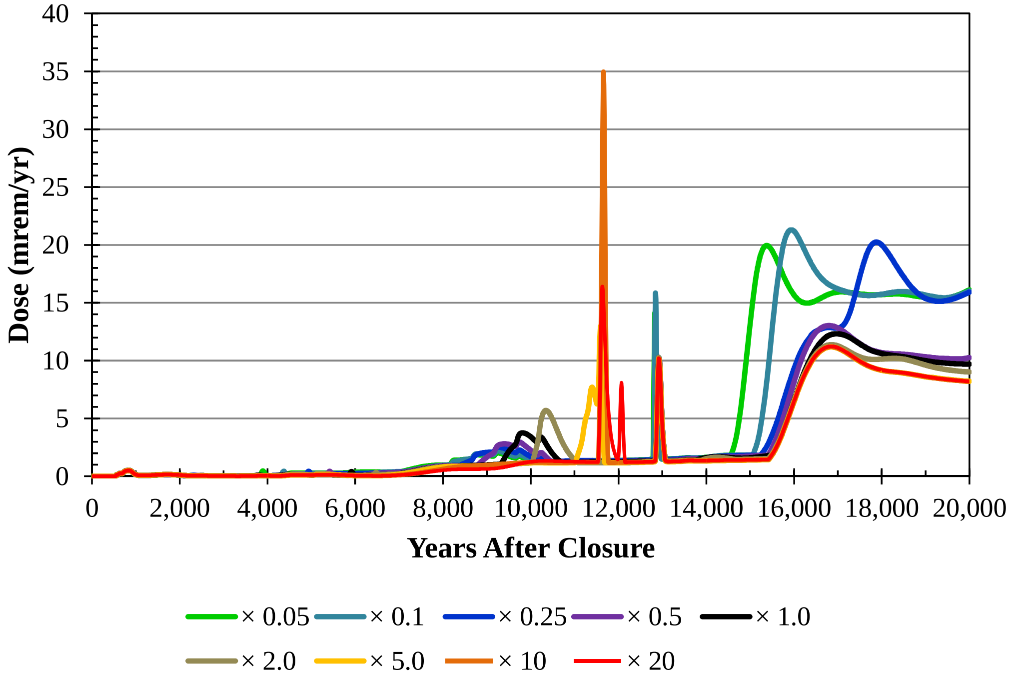


To illustrate further the potential impact on dose due to uncertainty in the pore volumes required to exhaust the reducing capacity of the grout, a series of sensitivity runs were performed using the GoldSim model in deterministic mode (these sensitivity runs are discussed further in the response to RAI-NF-3). This non-mechanistic study artificially varied the number of pore volumes required for transitions by applying a multiplier to the pore volume value. No other modeling changes were made relative to the Base Case. Table RAI-NF-4.1 provides a summary of the sensitivity runs and the resulting peak doses. Figure RAI-NF-4.7 shows that varying the pore volume value effectively shifts the timing and magnitudes of the peak doses.

Table RAI-NF-4.1: Peak MOP Doses from Pore Volume Sensitivity

Pore Volume Sensitivity Modeling Runs									
Pore Volume Multiplier	0.05	0.1	0.25	0.5	1.0	2.0	5.0	10	20
Resulting Pore Volume Value	26.2	52.3	131	262	523	1,046	2,615	5,230	10,460
Peak MOP Doses (mrem/yr)									
0 - 1,000 Years	0.52	0.52	0.52	0.52	0.51	0.51	0.51	0.51	0.51
0 - 10,000 Years	2.1	2.2	2.6	3.0	3.8	1.2	1.2	1.2	1.3
0 - 20,000 Years	20.0	21.3	20.3	13.0	12.3	11.4	13.0	35.0	16.4

Figure RAI-NF-4.7: Total MOP Doses from Pore Volume Sensitivity



RAI-NF-5

The concentration of dissolved oxygen in the groundwater that was assumed in the geochemical modeling for the submerged and partially submerged tanks might be underestimated.

Basis

The assumed low dissolved oxygen in groundwater prolongs the reducing conditions for the submerged and partially submerged tanks relative to the non-submerged tanks. DOE relies on 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 stated 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 (ML13126A127). 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.

Path Forward

DOE should evaluate the impact of using dissolved oxygen values that are consistent with measurements of unimpacted groundwater dissolved oxygen values across SRS (i.e., how many pore volumes would be required to transition the grout from reducing to oxidizing conditions). Alternatively, DOE could collect additional dissolved oxygen measurements within the HTF at locations and elevations that are in closer proximity to the tanks.

Response RAI-NF-5

The response to RAI-NF-3 included several sensitivity runs evaluating the number of pore volumes required for chemical transitions, and concluded that the current HTF PA approach maximizes the dose results in 10,000 years. Therefore, any reasonable change to the pore volume values would not adversely impact results of the HTF PA. However, the sensitivity runs documented in the response to RAI-NF-3 modified the pore volume values for both the submerged and the non-submerged waste tank transitions, making it difficult to evaluate impacts specific to submerged waste tanks. This response uses an additional sensitivity model to evaluate the impact from using other SRS measurements of dissolved oxygen values to determine transition timing for cementitious materials (e.g., waste tank reducing grout) for submerged waste tanks.

A representative concentration for dissolved oxygen was determined for use in the sensitivity runs. Table RAI-NF-5.1 shows the averaged dissolved oxygen values, as measured from other SRS well locations, both within and outside of the GSA.

Table RAI-NF-5.1: P Well Data, Averaged By Well

General Location	Well ID	Dissolved Oxygen (mg/L)
adjacent to C Reactor	P-18D	8.2
adjacent to F Area; part of P28 cluster	FC-2E	7.2
adjacent to F Area; part of P28 cluster	FC-2F	7.6
adjacent to H Area	P-27D	1.4
adjacent to P Reactor	P-24D	8.6
adjacent to R Reactor	P-20D	6.7
B area	P-29D	7.9
center of site; no facilities nearby	P-19D	7.4
N boundary of site; no facilities nearby	P-16D	8.9
NE boundary of site; no facilities nearby	P-17D	6.9
SE site; no facilities nearby	P-13D	8.3
SE site; no facilities nearby	P-21D	7.3
SE site; no facilities nearby	P-22D	8.2
SW site; no facilities nearby	P-23D	7.9
T Area	P-26D	6.0

[Developed from Attachment 1 of SRNL-L3200-2011-00011]

The mean dissolved oxygen concentration from all other wells is approximately 7.6 mg/L, ignoring the data specific to Well P-27D. This concentration shall be used as the representative SRS value for this sensitivity analysis. The ratio of the representative SRS value to the Well P-27D (7.6 mg/L to 1.4 mg/L) is approximately 5.4.

In *Evolution of Chemical Conditions and Estimated Solubility Controls on Radionuclides in the Residual Waste Layer During Post-Closure Aging of High-Level Waste Tanks,* SRNL-STI-2012-00404*,* it is explained that the “number of pore volumes required to reach the redox transition is inversely proportional to the dissolved oxygen concentration in the infiltrating water. Hence, at lower dissolved oxygen concentrations the number of pore volumes of infiltrating fluid required to cause the redox transition would be proportionally higher.” By applying an assumed dissolved oxygen concentration that is 5.4 times larger than previously used, the transitions would occur when the pore volume count is 5.4 times smaller than previously determined. In the HTF PA Base Case (Case A) modeling, the reducing grout in the submerged waste tanks transitioned from Submerged Condition C to Submerged Condition D after flushing through 1,787 pore volumes and to Oxidizing Region III after 2,442 pore volumes. The transition from Submerged Condition D to Oxidized Region III is not affected by the dissolved oxygen concentration; therefore, using the representative SRS value for dissolved oxygen would result in a modified transition for Submerged Condition C to Submerged Condition D of 331 pore volumes.

A deterministic sensitivity run was conducted using the HTF GoldSim Model. The model used the Base Case configuration and assumptions, except the submerged pore flush values were modified to reflect the SRS representative concentration of dissolved oxygen. As with the modeling performed for the HTF PA, the Submerged Condition parameters (i.e., number of pore volume flushes required for transition) were used to simulate the behavior of materials in the submerged waste tanks and the partially submerged waste tanks, as appropriate. The resulting dose is shown in Figure RAI-NF-5.1 and summarized in Table RAI-NF-5.2.

Figure RAI-NF-5.1: Dose Comparison of the HTF PA Base Case vs. Dissolved Oxygen Sensitivity Case

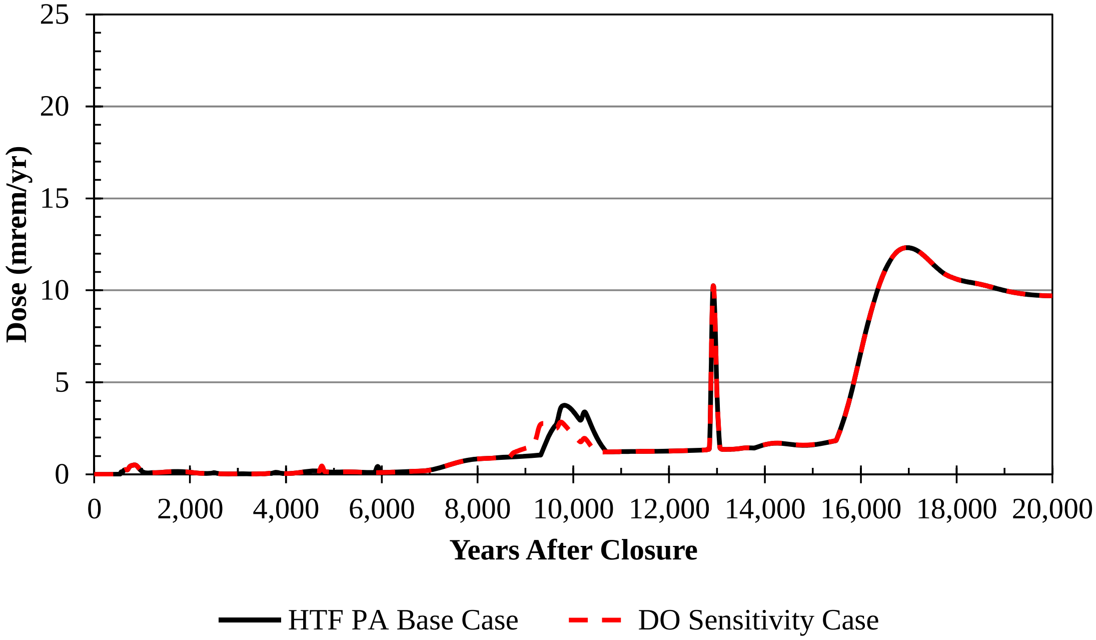


Table RAI-NF-5.2: Peak Dose Comparison of the HTF PA Base Case vs. Dissolved Oxygen Sensitivity Case

	0-1,000 Years		0-10,000 Years		0-20,000 Years	
Modeling Case	**Peak Doses (mrem/yr)**	**Time of Peak (yr)**	**Peak Doses (mrem/yr)**	**Time of Peak (yr)**	**Peak Doses (mrem/yr)**	**Time of Peak (yr)**
HTF PA Base Case	0.51	840	3.8	9,820	12.3	16,930
Dissolved Oxygen Sensitivity Case	0.52	840	2.8	9,740	12.3	16,930

The dose comparison between the Base Case modeling and the dissolved oxygen sensitivity modeling shows that the largest difference occurs between 9,000 and 11,000 years after HTF final facility closure and there is minimal impact on the peak dose in the 0 to 10,000-year period.

Using fewer pore volumes to determine the transition timing for submerged waste tanks results in earlier peak doses, but these earlier peak doses are of smaller magnitude. Although the release is earlier, the hydraulic properties of the cementitious materials (which have not yet fully degraded) retard contaminant transport, such that the peak of the release is lower. As such, the pore volume values assumed for the HTF PA are reasonably conservative, relative to this dissolved oxygen sensitivity. Therefore, using the lower dissolved oxygen concentration, as reported from Well P-27D, is a reasonable modeling approach for the HTF PA.

RAI-NF-6

The assumed solubility values for plutonium in the deterministic and probabilistic analyses are not well supported.

Basis

The assumed solubility of plutonium under reducing and oxidizing conditions in the HTF Performance Assessment provide a significant chemical barrier to waste release. The HTF Performance Assessment relies on a study in SRNL-STI-2012-00404, Rev. 0 that assumes $PuO_{2(am,\ hyd)}$ will control the solubility at 3E-11 mol/L under the three assumed chemical conditions. This assumption appears to be based on the thermodynamic calculations reported in SRNL-STI-2012-00087, Rev. 0. In this report, DOE argues that plutonium particles will be present as $PuO_{2(am,\ hyd)}$ based on thermodynamic stability. NRC staff are concerned that this assumption is not well supported and is inconsistent with Savannah River Site observations.

Residual waste samples from Tank 18F were indicative of a plutonium carbonate phase that had a significantly higher solubility than the phase assumed in the HTF Performance Assessment (ML12272A082). As discussed by NRC staff in the technical review, 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 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 grouted conditions. In addition, the HTF Performance Assessment has insufficient support for: (i) the as-modeled E_h threshold at which the solubility of plutonium significantly increases and (ii) the assumption that the E_h of infiltrating water will remain less than this threshold value during oxidizing conditions.

Path Forward

During the final characterization of residual tank waste, DOE should: (i) demonstrate that the quantity of high-solubility plutonium phase(s) remaining in the tanks is not risk-significant or (ii) if the quantity is risk significant, the high-solubility phase(s) will convert to lower-solubility phases under reducing grouted conditions. DOE should also provide additional support for the assumption that the E_h of infiltrating water will remain below the E_h threshold upon which plutonium transitions to a higher solubility phase than assumed in the HTF Performance Assessment.

Response RAI-NF-6

Section 3.3.1.2 of the *Savannah River Site Liquid Waste Facilities Performance Assessment Maintenance Program FY2013 Implementation Plan*, SRR-CWDA-2013-00049, identifies a plan to develop “a series of analytic methods to be used to test the solubility of plutonium, neptunium, uranium, and technetium under various simulated waste tank chemistry conditions using actual waste tank residuals.” Note that the “rate of performing activities will be dependent on analytical method maturity and budgetary constraints.” [SRR-CWDA-2013-00049] Until new data and testing is developed, DOE will continue to rely upon the best available current data and assumptions to assess reasonable assurance as described below.

Justification for the Assumed Plutonium Phase

As presented in *Form and Aging of Plutonium in Savannah River Site Waste Tank 18*, SRNL-STI-2012-00106, most of the plutonium entered into the waste tanks at SRS as freshly precipitated amorphous plutonium hydroxide ($PuO_{2(am,hyd)}$) and co-precipitated within a mixture of hydrous metal oxide phases containing metals such as iron, aluminum, manganese, and uranium. Additionally, the Tank 18 Plutonium Form and Aging report states that $PuO_{2(am,hyd)}$ is likely to be present in deposits and scales on the steel surfaces of the waste tank. [SRNL-STI-2012-00106]

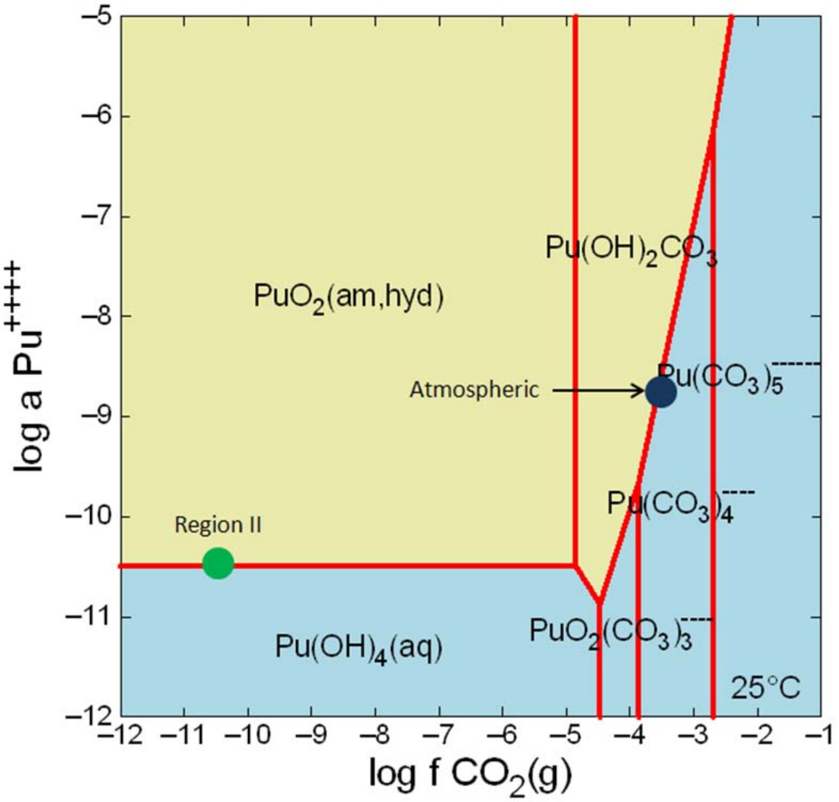
Although $PuO_{2(am,hyd)}$ is a likely form of plutonium present in the SRS waste tanks, the Tank 18 Plutonium Form and Aging report indicates that over “the operational period and after closure… $PuO_{2(am,hyd)}$ (will transform) to a more crystalline form of plutonium dioxide, $PuO_{2(c)}$.” [SRNL-STI-2012-00106] Referring to Tank 18 residual waste, the Tank 18 Plutonium Form and Aging report states that given various plutonium solubility studies from a literature review:

> *“… the $PuO_{2(am,hyd)}$ in Tank 18, whether as separate solid phases, adsorbed on the surface of other solids, or incorporated in coprecipitated solid phases, would be expected to undergo Ostwald ripening, albeit at a very slow rate, to produce a more crystalline solid phase. This is also true for the other amorphous metal hydroxides and hydrous metal oxides in the residues such as iron and aluminum. However, it would not be expected that the ripening process would continue to the completely dehydrated and crystalline $PuO_{2(c)}$ form. It is very likely, that the plutonium-containing solids would retain a surface layer of $PuO_{2(am,hyd)}$ due to the continual presence of water and reactions with products from the radiolysis of water.”* [SRNL-STI-2012-00106]

Even without OA cleaning, the stable phase for plutonium would be $PuO_{2(am,hyd)}$ after grouting. After water-washing Tank 18 the fluid remaining in the waste tank had a high pH, and was exposed to the atmosphere, which resulted in relatively high concentrations of dissolved CO_3^{-2} in the fluids. Under these conditions plutonium carbonate phases become the stable phases; however, washing a waste tank is expected to dissolve these phases. Furthermore, due “to the high alkalinity and low carbonate concentration in the grout formulation, it is expected that upon interaction with the grout and associated pore fluids, the plutonium carbonate complexes will transform back into plutonium hydroxide.” [SRNL-STI-2012-00106]

Figure RAI-NF-6.1 shows the stability regions of $PuO_{2(am,hyd)}$ and $Pu(OH)_2CO_3$ relative to log fugacity of CO_2 at high pH (11.1) when a waste tank is open to atmosphere and when a waste tank is grouted (Region II). When a waste tank is open to the atmosphere, the carbonate phase is thermodynamically favored. When a waste tank is grouted, $PuO_{2(am,hyd)}$ is favored.

Figure RAI-NF-6.1: Stability of Plutonium Phases Relative to Log Fugacity of CO_2 at pH=11.1 and E_h = +0.24 Volts



Using x-ray diffraction analyses, *Analysis of Samples from Tank 6F Chemical Cleaning*, SRNL-STI-2009-00493, describes the presence of plutonium oxalate hydrate in Tank 6 solids after washing with OA was completed. As conditions change during closure operations (from pre-washed, to OA washed, to grouted), the stable plutonium phase will change. The final phase change is rapid relative to the time the plutonium will spend within a grouted waste tank prior to any fluid release from the waste tank. Given the long time period that the residual waste will be exposed to grout-influenced pore fluids, it is anticipated that plutonium will exist primarily as $PuO_{2(am,hyd)}$ when the waste tank primary liners fail. Thus, assuming $PuO_{2(am,hyd)}$ as the controlling phase for plutonium, rather than the plutonium carbonate phase as indicated by the Tank 18 samples, is appropriate for PA modeling.

Justification for the Assumed E_h Values

The solubility of $PuO_{2(am,hyd)}$ in Region II conditions is sensitive to E_h at values above approximately +0.45 volts at which point Pu(IV) becomes less stable than Pu(V) and Pu(VI). There is no guarantee that E_h values will always remain below +0.45 volts or that plutonium will always behave as predicted by E_h and pH values; however, the preponderance of evidence from cementitious systems and natural systems suggest that E_h values will remain below +0.45 volts:

- *Application of Portland Cement-Based Materials to Radioactive Waste Immobilization,* 0956-053X(92)90044-J, reported that E_h values of ordinary Portland cement should be between 0.0 and +0.1 volts.
- Values of oxidized Portland cements reported in the *Fundamental Aspects of Cement Solidification and Stabilisation,* 0304-3894(96)01805-5, suggest E_h should be between +0.1 and +0.2 volts.
- E_h values for a non-SRS tank grout were measured at approximately +0.15 volts. [BNL-82395-2009]
- E_h values average +0.05 volts for the control saltstone sample that did not contain blast furnace slag used in *Equilibrium Solubility Model for Technetium Release from Saltstone Based on Anoxic Single-Pass Flow through Experiments,* PNNL-21723.
- The E_h values of natural waters are rarely in equilibrium with dissolved oxygen, predominantly because of slow reaction kinetics for oxidation by dissolved oxygen. [ISBN: 0-02-367412-1]
- Despite the persistent presence of oxygenated groundwater flowing throughout the SRS, the E_h values are substantially lower than expected for equilibrium with oxygen. [SRNL-STI-2012-00087, SRNL-STI-2012-00404]

Thus, the E_h values used to calculate radionuclide solubilities in the waste release model (i.e., +0.24 and +0.29 volts for Oxidized Regions II and III) are reasonable and appropriate for this analysis.

Assessment of Potential Risks

The HTF PA included a sensitivity analysis, performed with the deterministic PORFLOW model, which studied the impact of solubility variability on doses associated with key radionuclides (see HTF PA, Section 5.6.7.3). This non-mechanistic, thought exercise artificially imposed specific solubility values on the radionuclides of interest, including Pu-239, and held the solubility values constant regardless of changing waste tank conditions. The solubility values were chosen to provide a range of variability, including pessimistic values. The deterministic PORFLOW model was run using the Base Case (Case A) assumptions, except that the selected solubility values were varied. Figure RAI-NF-6.2 shows the impact of varying the plutonium solubility with respect to the dose contribution from Pu-239 (this figure was developed from HTF PA Figures 5.6-70 through 5.6-73).

Figure RAI-NF-6.2: Comparison of PORFLOW Plutonium Solubility Studies Compared to Base Case

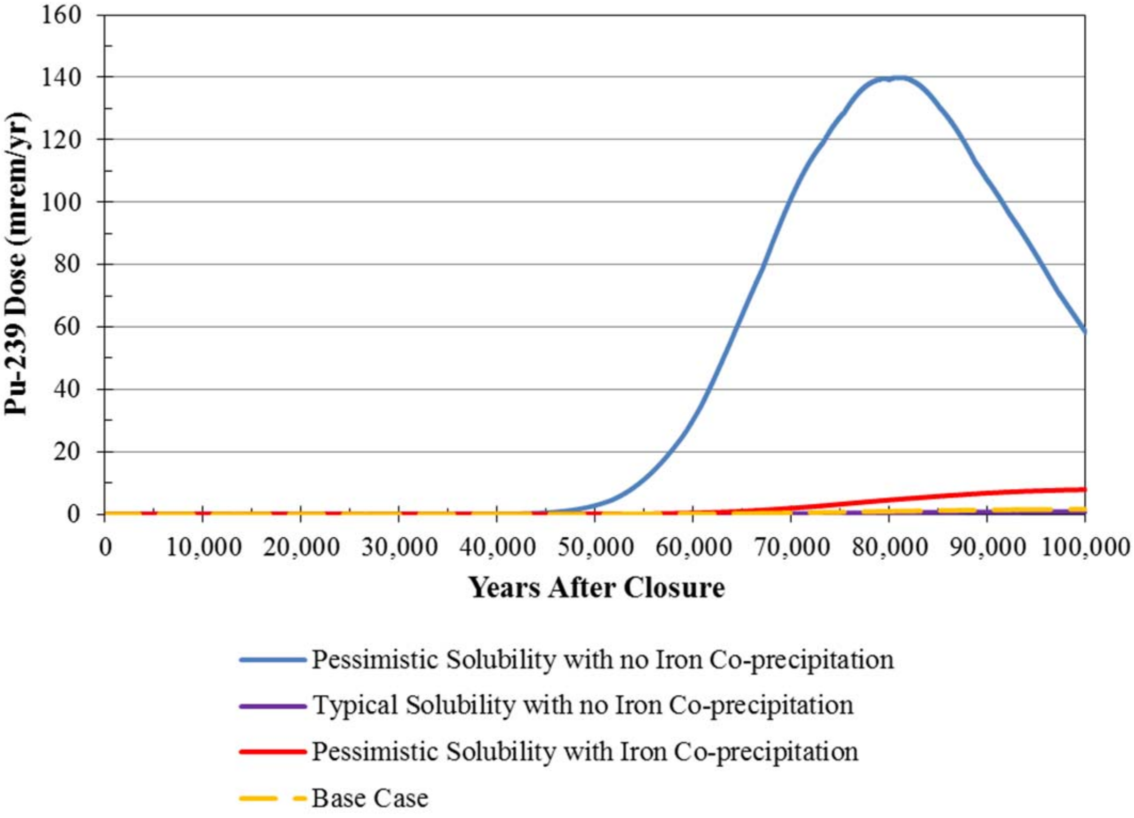


Table RAI-NF-6.1 provides a summary of these runs and the relative impact to dose. These results show that within the first 40,000 years there is not significant variability in the dose contributions from Pu-239. Only the most bounding study (i.e., solubility of 1.1E-07 mol/L) shows significant dose impacts. Even under this extreme condition, doses exceeding 25 mrem/yr do not occur until nearly 60,000 years after HTF final facility closure and the associated doses are significantly less than 500 millirem, the accepted dose standard for an inadvertent intruder. Note that this plutonium solubility value of 1.1E-07 mol/L is unrealistically conservative as it exceeds the maximum solubility value that was determined by assuming; 1) complete equilibrium with dissolved oxygen (i.e., ≤ 7.8E-08 mol/L), and 2) no iron co-precipitation (≤ 3.2E-11 mol/L). [SRNL-STI-2012-00087, SRNL-STI-2012-00404]

Table RAI-NF-6.1: Summary of PORFLOW Plutonium Solubility Studies with Base Case

		Peak Doses (mrem/yr)		
Solubility Study	**Pu-239 Solubility (mol/L)**	**in 1,000 Years**	**in 10,000 Years**	**in 100,000 Years**
Pessimistic Solubility with no Iron Co-precipitation	1.1E-07	0	< 0.1	140
Typical Solubility with no Iron Co-precipitation	1.6E-11	0	< 0.1	0.8
Pessimistic Solubility with Iron Co-precipitation	1.6E-10	0	< 0.1	7.9
Base Case	3.2E-11	0	< 0.1	1.6

Finally, the HTF PA GoldSim Model was used to perform an additional set of deterministic runs. These new modeling runs provide additional insights to the sensitivity of Pu-239 doses relative to plutonium solubility. For this analysis, as with the PORFLOW sensitivity analysis (described in the HTF PA), all of the Base Case assumptions were used except the plutonium solubility value was varied for each run. The objective of this analysis was to identify a hypothetical "worse-case" solubility value for plutonium (i.e., the solubility value that would result in the highest possible dose results). By identifying this value, the relative risk importance of plutonium solubility may be better understood.

The GoldSim model was deterministically executed numerous times out to 300,000 years (note that these models applied very coarse time steps of 150-years after the first 20,000 years). Each run used a different solubility value for plutonium, ranging from 3.2E-11 mol/L to 1E-04 mol/L and no solubility control. Figure RAI-NF-6.3 shows the peak Pu-239 doses from the runs as a function of the applied plutonium solubility value. Solubilities less than 1E-08 mol/L result in peak Pu-239 doses between 6 and 22 mrem/yr (between 90,000 and 165,000 years after HTF final facility closure). From 1E-08 mol/L to about 1E-06 mol/L, the peak Pu-239 dose increases as a function of the solubility value, peaking around 200 mrem/yr (around 75,000 years after HTF final facility closure), then the doses slowly decrease (at a solubility of 1E-04 mol/L, the peak dose is 192 mrem/yr, occurring around 75,000 years after HTF final facility closure). Note that these applied solubility values are non-mechanistic and discussed as part of a thought exercise to gaining understanding; they do not represent any specific phases of plutonium. These runs merely provide insight into the relationship between plutonium solubility and corresponding Pu-239 doses.

Figure RAI-NF-6.3: Peak Pu-239 Doses from GoldSim Solubility Study

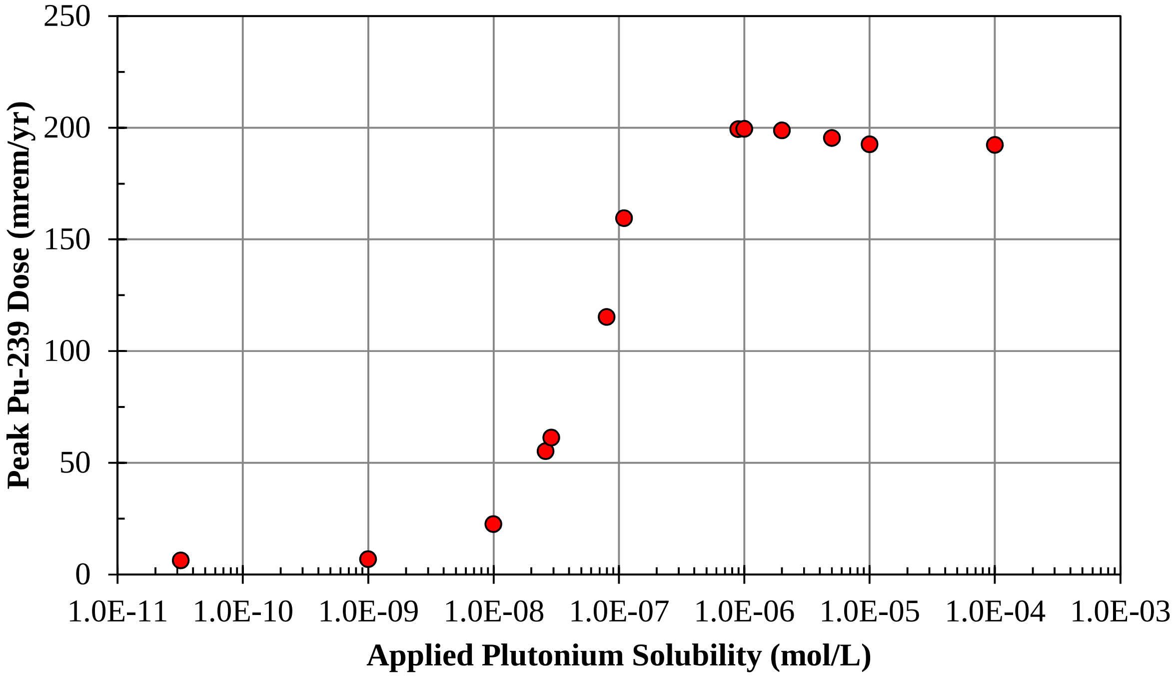


Table RAI-NF-6.2 summarizes these deterministic HTF PA GoldSim Model results. Regardless of the applied solubility value, the Pu-239 dose contribution does not reach 1.0 mrem/yr until approximately 50,000 years after HTF final facility closure. Beyond 50,000 years, the highest peak doses only occur when the solubility exceeds expected values given the chemical conditions of the waste tank grout and natural system.

Table RAI-NF-6.2: Summary of GoldSim Plutonium Solubility Studies

Solubility (mol/L)	Peak Dose (mrem/yr)	Time of Peak Dose (yr)	Time When 25 mrem/yr is Exceeded (yr)
3.2E-11	6.3	164,150	N/A
1.0E-09	6.8	161,300	N/A
1.0E-08	22	90,050	N/A
2.6E-08	55	89,750	70,550
2.9E-08	61	89,750	69,650
8.0E-08	115	87,650	65,900
1.1E-07	160[a]	80,150	62,150
3.0E-07	191	75.800	59,150
9.0E-07	199	74,600	57,650
1.0E-06	200	74,600	57,650
2.0E-06	199	74,300	57,650
5.0E-06	195	74,900	58,100
1.0E-05	193	75,050	58,400
1.0E-04	192	75,350	58,400
No Sol. Control	192	75,200	58,400

a: As described in Section 5.6.2 of the HTF PA (GoldSim Benchmarking), the GoldSim model is a close approximation, but not an exact match to the PORFLOW model; hence some differences in the results between PORFLOW and GoldSim modeling are expected. For example, Table RAI-NF-6.1 shows a peak Pu-239 dose of 140 mrem/yr using a solubility of 1.1E-07 mol/L, rather than the 160 mrem/yr shown here.

Given the conditions of the system, as explained above, the highest of these solubility values that may be considered possible (albeit conservative) is the value of 2.9E-08 mol/L. This value represents the measured concentration from the Tank 18 heel when plutonium solubility was likely controlled by a carbonate phase. [SRNL-STI-2012-00106] DOE maintains that 3.2E-11 mol/L represents the most reasonable estimate for plutonium solubility within the post-closure environment. This exercise provides further support that the peak doses associated with Pu-239 will remain below 25 mrem/yr well past 10,000 years regardless of the solubility value assumed for plutonium.

RAI-NF-7

DOE does not provide a sufficient basis for assuming that 100% of the technetium-99 is co-precipitated with iron.

Basis

The assumption that 100% of the Tc-99 is co-precipitated with iron significantly limits its release under all chemical conditions. Based on SRNL-STI-2012-00404, Rev. 0, DOE assumes that tank washing will effectively remove all of the more soluble Tc-99. The authors also cite several studies from Hanford that provide evidence of Tc-99 being co-precipitated with iron. However, DOE has not demonstrated that tank washing will remove all of the more soluble technetium-99 and that the remaining Tc-99 will be co-precipitated with iron.

Path Forward

If the inventory of Tc-99 is not reduced to an insignificant risk level, DOE should provide experimental evidence to support the assumption that technetium solubility is controlled by iron co-precipitation under all chemical conditions.

Response RAI-NF-7

The final residual Tc-99 inventories in the SRS waste tanks cleaned to date (i.e., Tanks 5, 6, 17, 18, 19, 20) have all been reduced to an insignificant risk level (i.e., Tc-99 residual inventories have ranged from 0.1 curie to 1.7 curie) consistent with the assumption that the bulk of technetium within the waste tanks is relatively, soluble and will be removed during the waste removal process. Because none of the waste tanks cleaned to date have had final residual Tc-99 inventories which pose a significant risk to a future hypothetical MOP or inadvertent intruder, the assumed solubility values utilized for Tc-99 in the HTF PA modeling, and the potential impact to dose if the Tc-99 solubility assumptions are not valid, has not impacted conclusions regarding peak dose results.

At the completion of waste removal activities for waste tanks that have not yet been cleaned, DOE will perform a final characterization of the residuals remaining in the waste tank. The final residual Tc-99 inventories will be determined and the potential impacts of the Tc-99 inventories will be evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1. Typically, new fate and transport modeling will be performed as part of the SA process, replacing the HTF PA assigned waste tank-specific Tc-99 inventories with the final residual characterization data. The results of the waste tank-specific SAs are then evaluated to determine if any of the new information impacts HTF PA conclusions. While it typically would not be necessary to replicate all of the HTF PA sensitivity and uncertainty analyses in an SA, it may be appropriate to include additional analyses to evaluate the new or unique waste tank-specific information. In the case of Tc-99 inventory, a significant increase in the Tc-99 remaining at waste tank operational closure (i.e., significantly above what was evaluated in the HTF PA) may require additional Tc-99 related sensitivity analyses. For example, higher Tc-99 inventory values may warrant additional evaluation of the impact of Tc-99 solubility assumptions on the peak dose results and may eventually lead to additional experimental studies.

The HTF PA included a sensitivity analysis, performed with the deterministic PORFLOW model that considered the impact of solubility variability on doses associated with key radionuclides including Tc-99 (see HTF PA, Section 5.6.7.3). This non-mechanistic thought exercise artificially imposed specific solubility values on the radionuclides of interest and held the solubility values constant regardless of changing waste tank conditions. The solubility values were chosen to provide a range of variability, including pessimistic values. The deterministic PORFLOW model was run using the Base Case (Case A) assumptions, except that the selected solubility values were varied. Figure RAI-NF-7.1 shows the impact of varying the technetium solubility with respect to the dose contribution from Tc-99 (this figure was developed from HTF PA Figures 5.6-70 through 5.6-73).

Figure RAI-NF-7.1: Comparison of PORFLOW Technetium Solubility Studies with the Base Case

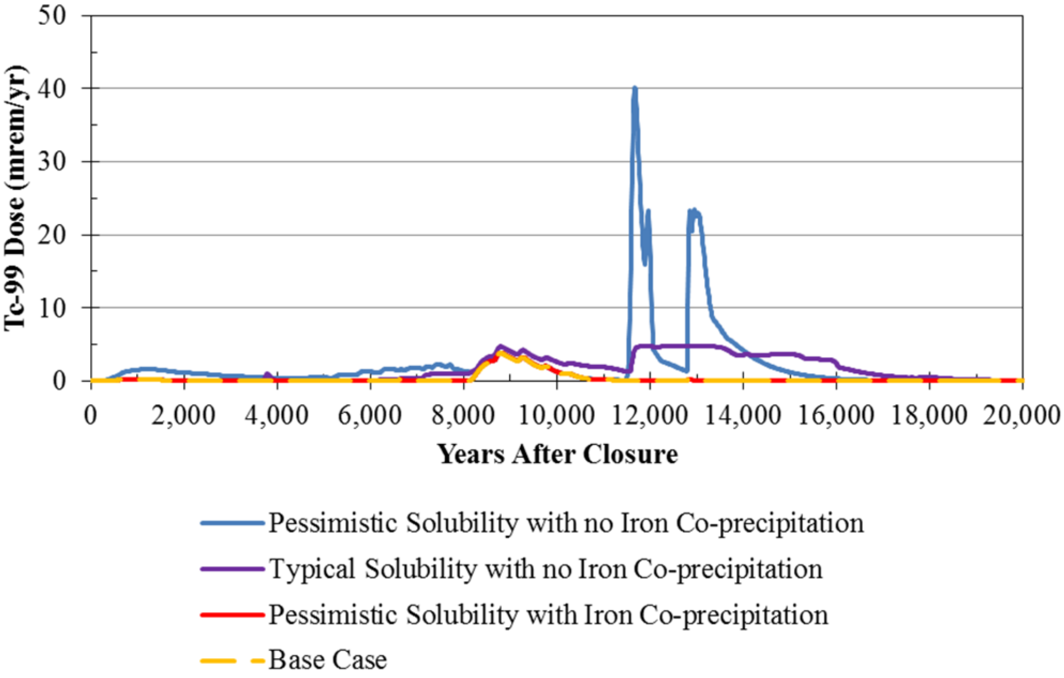


Table RAI-NF-7.1 provides a summary of the PORFLOW runs and the relative impact to dose. The results show that within the first 10,000 years after HTF final facility closure there is no significant variability in the dose contributions from Tc-99. Only the most bounding study, pessimistic solubility with no iron co-precipitation (i.e., instantaneous release of technetium) shows significant dose impacts. Even under the extreme condition of assuming instantaneous release of technetium, doses above 5 mrem/yr do not occur until beyond 10,000 years after HTF final facility closure, as controlled by the failure timing of the waste tank primary liners and the peak dose, assuming simultaneous failures of each tank type, of only 40 mrem/yr.

Table RAI-NF-7.1: Summary of PORFLOW Technetium Solubility Studies with Base Case

		Peak Tc-99 Dose (mrem/yr)		
Solubility Study	**Assumed Tc-99 Solubility[(a)]**	**in 1,000 Years**	**in 10,000 Years**	**in 20,000 Years**
Pessimistic Solubility with no Iron Co-precipitation	instantaneous release	2	4	40
Typical Solubility with no Iron Co-precipitation	1.10E-08	0.3	5	5
Pessimistic Solubility with Iron Co-precipitation	1.30E-12	0.3	4	4
Base Case	Varies[(b)]	0.3	4	4

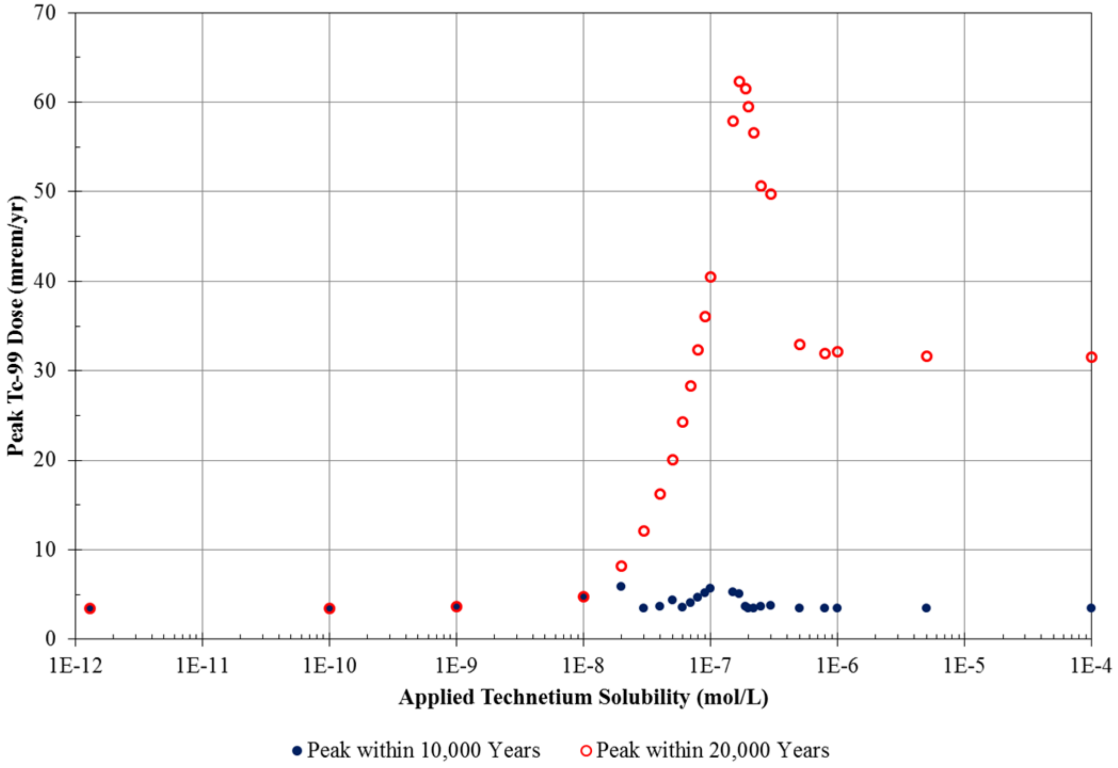
(a): Assumed solubility values were based on SRNL-STI-2012-00404.

(b): Unlike the sensitivity studies, the Base Case configuration varied technetium solubility based on the chemical condition of the cementitious environment. Base Case solubility values for technetium ranged from 1E-15 mol/L to 3E-12 mol/L. See Table 4.2-14 of the HTF PA.

Finally, the HTF PA GoldSim Model was used to perform a new set of deterministic sensitivity runs. These new modeling runs provide additional insights to the sensitivity of Tc-99 doses relative to technetium solubility. For these deterministic sensitivity analyses, as with the PORFLOW sensitivity analyses (described in Section 5.6.7.3 of the HTF PA), all of the Base Case assumptions were used except that the technetium solubility value was varied for each run. The varied technetium solubility values were held constant for all time, regardless of chemical conditions. The objective of these deterministic sensitivity analyses was to identify a hypothetical "worse-case" solubility value for technetium (i.e., the solubility value that would result in the highest possible dose results). By identifying this "worse-case" value, the relative risk importance of technetium solubility may be better understood.

The HTF GoldSim Model was deterministically executed 25 times, each time using a different solubility value for technetium, ranging from 1.3E-12 mol/L to 1E-04 mol/L. This sensitivity did not evaluate solubilities less than 1.3E-12 mol/L because it was determined that values less than 1E-08 mol/L did not show significantly variation in the dose results. Figure RAI-NF-7.2 shows the peak Tc-99 doses from all 25 runs as a function of the applied technetium solubility value. Solubilities between 1.3E-12 mol/L and 1E-08 mol/L result in peak Tc-99 doses of approximately 3.5 mrem/yr. For solubility values from 1E-08 mol/L to about 1.7E-07 mol/L, the peak Tc-99 doses increase, reaching approximately 65 mrem/yr within 20,000 years when the technetium solubility is approximately 1.7E-07 mol/L. The doses decrease to approximately 32 mrem/yr starting near a solubility value of 1E-06 mol/L.

Figure RAI-NF-7.2: Peak Tc-99 Doses from GoldSim Solubility Study



The higher peak doses from using solubility values between 1E-07 mol/L and 1E-06 mol/L, rather than a dose peak associated with no solubility controls, are counterintuitive and illustrate the non-linearity of the system. Within this range, the Tc-99 inventories in sources that release early (e.g., from the annulus and sand pads) are retarded by the solubility control, such that once the waste tank liners fail, the inventories combine to maximize the peak dose at the compliance point.

Note that these applied solubility values are non-mechanistic and do not represent any specific phases of technetium. These HTF GoldSim Model runs merely provide insight into the relationship between technetium solubility and corresponding Tc-99 doses.

Table RAI-NF-7.2 summarizes the results of the HTF GoldSim Model solubility study. Regardless of the applied solubility value, the peak Tc-99 dose contribution never exceeds 6 mrem/yr within the first 10,000 years after HTF final facility closure. Beyond 10,000 years after HTF final facility closure, the highest peak doses occur when the Tc-99 solubility values greatly exceed those expected, given the chemical conditions imparted by the waste tank fill grout.

Table RAI-NF-7.2: Summary of GoldSim Technetium Solubility Studies(a)

	Peak Tc-99 Dose in 10,000 Years		Peak Tc-99 Dose in 20,000 Years	
Solubility Case	**Year of Peak**	**Dose (mrem/yr)**	**Year of Peak**	**Dose (mrem/yr)**
Base Case	9,810	3.5	9,810	3.5
1.3E-12 mol/L	9,810	3.5	9,810	3.5
1E-10 mol/L	9,810	3.5	9,810	3.5
1E-9 mol/L	9,810	3.6	9,810	3.6
1E-8 mol/L	9,820	4.8	9,820	4.8
2E-8 mol/L	9,770	5.9	14,080	8.2
3E-8 mol/L	9,810	3.5	13,000	12
4E-8 mol/L	8,660	3.7	13,010	16
5E-8 mol/L	8,370	4.4	13,000	20
6E-8 mol/L	7,760	3.5	12,990	24
7E-8 mol/L	7,760	4.1	13,000	28
8E-8 mol/L	7,490	4.7	12,990	32
9E-8 mol/L	7,480	5.2	13,020	36
1E-7 mol/L	7,390	5.7	13,000	40
1.5E-7 mol/L	7,410	5.2	12,940	58
1.7E-7 mol/L	7,360	5.1	12,920	62
1.9E-7 mol/L	7,320	3.6	12,890	62
2E-7 mol/L	9,810	3.5	12,880	60
2.2E-7 mol/L	9,810	3.5	12,850	57
2.5E-7 mol/L	6,940	3.6	12,920	51
3E-7 mol/L	6,870	3.8	12,900	50
5E-7 mol/L	9,810	3.5	12,890	33
8E-7 mol/L	9,810	3.5	12,900	32
1E-6 mol/L	9,810	3.5	12,900	32
5E-6 mol/L	9,810	3.5	12,900	32
1E-4 mol/L	9,810	3.5	12,900	32
No Control	9,810	3.5	12,900[a]	31

(a): As described in Section 5.6.2 of the HTF PA (GoldSim Benchmarking), the GoldSim model is a close approximation, but not an exact match to the PORFLOW model; hence, some differences in the results between PORFLOW and GoldSim modeling are expected. For example, Figure RAI-NF-7.1 shows a peak Tc-99 dose, using no solubility controls, of approximately 40 mrem/yr (near 11,500 years), rather than the 31 mrem/yr (at 12,900 years) as shown here.

RAI-NF-8

The assumption that solubility limits apply to radionuclides that migrate upward from annular waste into the contaminated zone is not well supported.

Basis

Although no solubility control is assumed for radionuclides associated with waste initially located in tank annuli, solubility controls are realized for radionuclides that are able to diffuse into the contaminated zone from the annulus. For example, DOE estimates that technetium-99 in Tank 16H is present in higher quantities in the annulus (primary sand pad) compared to the contaminated zone with no effective transport barrier between the two (Tank 16 is assumed to have a failed liner initially; the tank liner would constitute a transport barrier to upward diffusion if it were effective). The Tc-99 inventory located within the primary liner is constrained to low aqueous concentrations owing to solubility controls associated with iron co-precipitated phases that are placed on this constituent in the contaminated zone. The assumption that solubility limiting phases would control the solubility of radionuclides that diffuse upward from the annulus is not well supported, particularly for radionuclides (i.e., technetium) that DOE models as co-precipitated with iron phases. Given the large concentration gradient and small diffusion length between the primary sand pad, where the bulk of Tank 16H annulus contamination is placed in the PORFLOW® model, and the contaminated zone, a significant portion of Tc-99 diffuses into the contaminated zone where it is retained for most of the simulation timeframe due to very low solubility limits associated with iron co-precipitation. Although Tc-99 in Tank 16H may not be risk-significant, risk-significant quantities of key radionuclides may be present in Tank 16H or other tanks and experience the same phenomena.

Path Forward

Assess the sensitivity of solubility control on that portion of annular inventory that diffuses into the contaminated zone on tank releases (or resultant doses). DOE should also provide a basis for solubility limits in the contaminated zone for radionuclides that are transported upward from the annulus and sand pads in Type I and II tanks. In particular, the basis should address the appropriateness of iron co-precipitated phases to limit solubility of radionuclides that have migrated into the contaminated zone within the primary tank liner.

Response RAI-NF-8

DOE agrees that the modeling approach used in the HTF PA may, in some instances result in non-mechanistic radionuclide behavior due to practical simplifications applied to the conceptual model. For example, specific material conditions (e.g., waste tank grout in Oxidized Region III) were assigned a single set of solubility controls for each radionuclide rather than applying multiple sets of solubility controls to model variability in radionuclide-specific phase stability (i.e., it is likely that some radionuclides may be present in different phases within the same material). Although it is not always realistic, this simplified approach facilitates modeling of solubility controls that are appropriate under most conditions. The relative risk significance of these simplifications is lessened because current modeling assumptions work to maximize the peak doses within the performance period (as demonstrated by a number of alternative sensitivity models shown in the responses to other RAIs). Furthermore, while the inventories projected to remain at closure within the annuli and the sand pads of the Type I and Type II tanks are typically considered bounding in nature (Tank 16 annulus being the exception), these inventories are small relative to the in-tank inventories of the entire HTF.

Justification for the Current Modeling Approach

When originally developing the model configurations for the HTF PA, the residual annulus and sand-pad inventories were not anticipated to be significant contributors to the overall HTF dose. This informed premise is based on the relatively insignificant total curies present outside the confines of the primary tanks. When modeling the annulus and sand pads, no special effort was made to prevent material in these regions from migrating into the contamination zone within the waste tanks (i.e., adding to the in-tank inventories) because this modeling assumption is expected to maximize the overall HTF peak doses.

Future PA Revisions and the Role of SAs

As required by DOE Manual 435.1-1, maintenance of the HTF PA will include future updates to incorporate new information, update model codes, consider actual residual inventories, etc., as appropriate. Section 8.2 of the HTF PA states that as "additional data becomes available …additional modeling may be required." Each time additional modeling is performed, DOE will evaluate whether or not an update to the Base Case (Case A) or a revision of the PA is needed. PA maintenance and the potential impact of new information are evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1. New fate and transport modeling will be performed, as required, through the SA process, replacing PA assigned inventories with the final residual characterization data. If other new information has been identified (e.g., updated K_d values developed through research or experimentation), this new data would also be evaluated through inclusion in the SA. The results of the waste tank-specific SAs are then evaluated to determine if new information impacts PA-based conclusions. While it typically would not be necessary to replicate all of sensitivity and uncertainty analyses from the PA as part of a SA, it may be appropriate to include additional analyses to evaluate the new or unique waste tank-specific information.

As waste tank-specific SAs are prepared, it is anticipated that sensitivity analyses will be performed to explicitly address both individual waste tank inventories and unique waste tank conditions, such as potential preferential pathways (e.g., degraded liners or in-leakage pathways). Because different as-modeled waste tank conditions may prove to be conservative or non-conservative (with respect to timing and/or magnitude of peak doses) depending on other corresponding waste tank conditions, these analyses are more suited to waste tank-specific sensitivity analyses than to changes to the overarching Base Case assumptions. For example, modeling assumptions that cause the annulus inventories to be released relatively slowly (thereby adding to releases from the in-tank inventories) versus relatively rapidly would impact peak doses differently depending on the quantity (total curies) and nature of the inventory remaining in a waste tank annulus (e.g., short-lived versus long-lived radionuclides). In a similar manner, preferential pathways such as failed liners can have very radionuclide-specific dose impacts, with earlier liner failures not necessarily corresponding to greater peak doses, as demonstrated in HTF PA Section 5.6.7.6 (Liner Failure Times Analysis using the PORFLOW Deterministic Model). Additionally, if the closure inventory for a specific radionuclide is determined to be significantly different than previously projected, the impact of the modeling simplification discussed in the RAI would need to be further evaluated.

Evaluation of Potential Risks from Annulus and Sand Pad Inventories

To better quantify dose risks associated with the residual inventories that have been assigned to the annulus and sand pads at final facility closure, a number of additional deterministic model analyses were performed using the GoldSim model. These deterministic analyses isolate the potential dose impacts from the annulus and the sand pads inventories by modifying the assumed inventories, as well as changing other modeling assumptions to exaggerate the dose impacts.

The first set of deterministic analyses illustrate the relative magnitude of doses associated with the various modeled inventory sources (i.e., residual waste assumed within the waste tanks and ancillary structures versus waste assumed within the annulus and sand pads, when all sectors are considered). Figure RAI-NF-8.1 shows peak dose results from all sectors, using Base Case modeling assumptions, from the various inventory sources. This figure shows dose on a logarithmic scale, as the dose results from the sand pads inventories are relatively small and are not easily distinguishable on a linear scale.

Figure RAI-NF-8.1: Base Case MOP Dose Contributions by Inventory Sources (All Sectors)

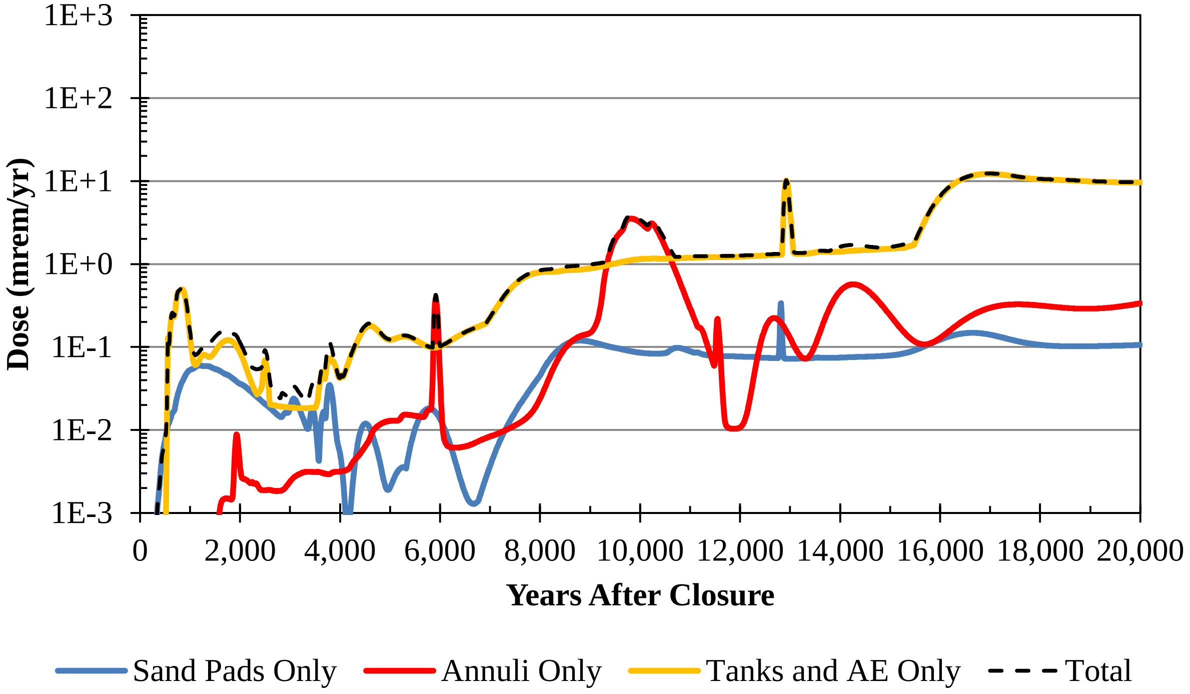


Table RAI-NF-8.1 provides a summary of the peak dose contributions from each of the inventory sources.

Table RAI-NF-8.1: Summary of Peak MOP Doses by Inventory Sources

Source	Peak MOP Doses (mrem/yr)		
	in 1,000 Years	in 10,000 Years	in 20,000 Years
Sand Pads Only	5.3E-02	1.2E-01	3.4E-01
Annuli Only	9.2E-05	3.5E+00	3.5E+00
Waste Tanks and Ancillary Structures Only	4.9E-01	1.1E+00	1.2E+01
Total	5.1E-01	3.8E+00	1.2E+01

As expected, the dose from the waste tanks and ancillary structures inventories dominate these results with a peak dose of approximately 12 mrem/yr within 20,000 years after all waste tank liners fail. The dose associated with the sand pads inventories never reach more than 0.4

mrem/yr within 20,000 years. The dose from the annulus inventories were generally about one order of magnitude less than the dose from the waste tanks and ancillary structures, although annulus inventories do drive the dose between 9,000 and 10,500 years, with a peak dose of about 3.5 mrem/yr. As discussed in the response to RAI-INV-2, the goal during the development of the annulus inventory estimates was to ensure conservative values were assigned, not necessarily representative future projections.

The bulk of the dose in the 9,000 to 10,500 year period is attributed to waste releasing from the annulus once the annulus grout becomes oxidized. Regardless, the peak dose associated with the annular waste is only about one-third of the peak dose associated with the in-tank waste when examining the full 20,000 years that were simulated for this analysis. Despite the conservative inventory assignments, if these peaks occur earlier (i.e., within the 10,000-year period), they would not result in peak doses that exceed 25 mrem/yr.

For the second set of additional deterministic analyses, only the annulus and sand pads inventories (combined) were considered. These deterministic analyses exaggerate the potential dose risks by modifying the Base Case modeling assumptions to create extremely pessimistic conditions. Results from four deterministic analyses are shown in Figure RAI-NF-8.2. The dose is on a logarithmic scale, as the dose results are not easily distinguishable on a linear scale. The first curve (in red) provides the Base Case dose from the annulus and sand pads inventories. The second curve (blue) assumes no solubility controls in any cementitious materials (i.e., waste tank grout, annulus grout, basemat concrete, and wall concrete) and applies the K_d values for Oxidized Region III to these materials thus effectively reducing the significance of the concrete and grout as barriers to release. The third curve (yellow) assumes an extremely pessimistic flow condition by applying Parametric Flow Run 65 (see Table 4.4-14 of the HTF PA). This deterministic analysis incorporates a number of pessimistic modeling assumptions:

- Waste tank liners fail at time zero (this allows flow to overwhelm the effects of any diffusion from the annulus into the contamination zone)
- Full fast flow (uses the Case E waste tank configuration to model a channel through the grout and basemat with no flow impedance)
- Fast hydraulic degradation of cementitious materials (degradation timing is divided by two)
- No closure cap in place (maximizes infiltration and flow)

Finally, the fourth curve (green, dashed) combined the assumptions from the second and third curves to provide insights into the dose impact from combining these pessimistic (unrealistic) assumptions. Table RAI-NF-8.2 provides a summary of each of the sensitivity models.

Figure RAI-NF-8.2: Dose Results from Sensitivity Studies of Annulus and Sand Pad Inventories (All Sectors)

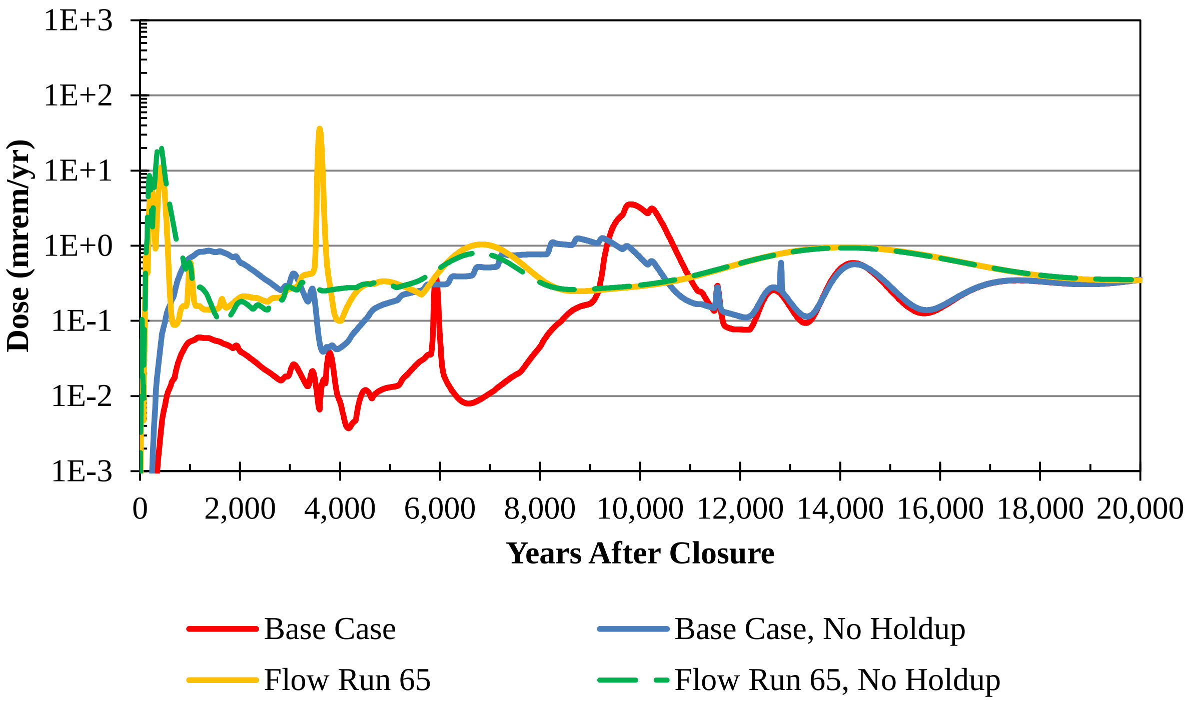


Table RAI-NF-8.2: Summary of Sensitivity Models to Evaluate Annulus and Sand Pad Inventories

Deterministic Analyses	Peak MOP Doses (mrem/yr)		
	in 1,000 Years	in 10,000 Years	in 20,000 Years
Base Case	5.3E-02	3.6E+00	3.6E+00
Base Case, No Holdup	6.9E-01	1.3E+00	1.3E+00
Flow Run 65	1.1E+01	3.6E+01	3.6E+01
Flow Run 65, No Holdup	2.5E+01	2.5E+01	2.5E+01

These results show that assuming the grout and concrete are ineffective barriers in the no holdup cases actually reduces the magnitudes of the overall peak doses contributed by the annulus and sand pads inventories. The pessimistic properties assumed for the grout and concrete allow waste within the annulus and sand pads to release at the same time, such that a compounding effect would be expected; however, there are two features of the system that diminish this compounding effect. First, materials within the system are not yet fully degraded, so early release of the annular waste results in slower transport. Second, when the grout is an ineffective barrier the Tc-99 inventory is released gradually (resulting in a "bleeding-off" effect over a longer period of time) rather than retaining the mass (resulting in a relatively high-magnitude flushing effect when the annular grout becomes oxidized).

The Flow Run 65 sensitivity analysis only resulted in a peak dose of 36 mrem/yr (using normal grout and concrete properties) and 25 mrem/yr (using the pessimistic grout and concrete properties). Note that the HTF PA reported a peak dose of 239 mrem/yr for Case E (see Table 5.6-44 of the HTF PA). The Case E sensitivity model, as reported in the HTF PA, used the Case E waste tank configuration (i.e., fast flow), but the other modeling assumptions were kept the same as the Base Case. Despite this, the peak dose from using all of the inventory sources (including the waste tanks) was nearly an order of magnitude greater than the annulus and sand pads inventory doses. This again demonstrates that even under the most pessimistic conditions the waste tank inventories dominate the dose contributions.

Conclusion

The upward migration of annular waste into the contamination zone is a consequence of a modeling simplification. Although this simplification results in unexpected radionuclide behavior for a small fraction of the total contaminants, the overall effect is that some waste is held up within the contamination zone, resulting in higher overall doses once the waste tank inventories are released. Therefore, this modeling simplification is generally conservative and appropriate for assessing future risk.

RAI-NF-9

Solubility values used in the probabilistic modeling do not adequately account for uncertainty.

Basis

In Section 5.6.3.3 of the HTF Performance Assessment, 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®.

The probabilistic model in the HTF Performance Assessment assumes a discrete distribution for the solubility controlling phases, as discussed in Section 5.6.3.3 of the HTF Performance Assessment. DOE assumes that the probability of iron co-precipitation for plutonium, technetium, neptunium, and uranium is 50% with the remaining 50% assigned to approximate the solubility values listed in Table 4.2-11 of the HTF Performance Assessment. The values listed in Table 4.2-11 of the HTF Performance Assessment are approximately consistent with the base case values for the deterministic modeling for plutonium, neptunium, and uranium. For technetium, DOE assumes that it is co-precipitated with iron in the deterministic base case rather than using its solubility value listed in Table 4.2-11 of the HTF Performance Assessment. However, DOE has only provided indirect evidence of iron co-precipitation of highly radioactive radionuclides. Furthermore, DOE does not account for the possibility of higher solubility phases due to the presence of carbonate ions even though analyses of residual waste from Tank 18F identified a uranyl carbonate phase and were also indicative of a plutonium carbonate phase (ML12272A082). In addition, the base case solubility values do not represent a reasonable upper solubility limit.

To help account for the uncertainty in relying on thermodynamic data for solubility values, DOE utilized the uncertainty information provided by the Nuclear Energy Agency (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, exceed the base case solubility values assumed in the probabilistic model for plutonium, neptunium, and uranium.

It is not clear how the uncertainty in thermodynamic data was 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, SRNL-STI-2012-00404, Rev. 0 indicates that the uncertainty in thermodynamic data can result in two orders of magnitude variation in solubility. Accordingly, varying the chemical transition times would not account for the uncertainty in thermodynamic data related to the solubility of plutonium.

In Section 5.6.7.3 of the HTF Performance Assessment, DOE provides a one-off deterministic sensitivity analysis of plutonium, neptunium, technetium, and uranium solubility values. The pessimistic solubility values that DOE assumed in study 1 provide only limited risk insight. Unless implausible, these pessimistic values should be considered to develop the solubility value distributions in the full probabilistic analysis.

NRC staff disagree with DOE's assertions that: (i) solubility controlling phases for plutonium, neptunium, technetium, and uranium were conservatively selected and (ii) 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, NRC staff is concerned that the assumed probability distributions in the HTF Performance Assessment for plutonium, neptunium, technetium, and uranium are optimistic.

Path Forward

DOE should revise the assumed probability distributions with more defensible values and provide revised results from the probabilistic analysis.

Response RAI-NF-9

New probability distributions were developed to evaluate the solubility uncertainty concern using values from HTF PA Tables 5.6-10 through 5.6-12 and 5.6-46. Tables RAI-NF-9.1 through RAI-NF-9.5 present these revised distributions. These were implemented in the probabilistic Base Case (Case A) GoldSim model and the resulting doses were evaluated against the results from the probabilistic Base Case GoldSim model described in Section 5.6.4 of the HTF PA. In most cases, the new sampling values introduce pessimistic values, relative to those that were assumed for the HTF PA.

Table RAI-NF-9.1: Probability Distributions for Various Phases Controlling Reduced Region II Solubility

	Controlling Phase	Solubility (mol/L)	Probability
Plutonium	$PuO_{2(am,\ hyd)}$	3.2E-11	0.25
	Normal Iron co-precipitation	7.6E-13	0.25
	Pessimistic Iron co-precipitation	1.6E-10	0.25
	Pessimistic No co-precipitation	1.1E-07	0.25
Neptunium	$NpO_{2(am,\ hyd)}$	9.9E-10	0.25
	Normal Iron co-precipitation	4.6E-15	0.25
	Pessimistic Iron co-precipitation	3.0E-13	0.25
	Pessimistic No co-precipitation	1.0E-04	0.25
Technetium	$TcO_2.1.6H_2O$	1.1E-08	0.25
	Normal Iron co-precipitation [a]	1.1E-14	0.25
	Pessimistic Iron co-precipitation	1.3E-12	0.25
	Pessimistic No co-precipitation	Instantaneous release	0.25
Uranium	$UO_{2(am,\ hyd)}$	4.6E-09	0.25
	Normal Iron co-precipitation	2.4E-12	0.25
	Pessimistic Iron co-precipitation	1.8E-10	0.25
	Pessimistic No co-precipitation	3.4E-04	0.25

(a): Iron co-precipitation values used for technetium in deterministic simulations.

Table RAI-NF-9.2: Probability Distributions for Various Phases Controlling Oxidized Region II Solubility

	Controlling Phase	Solubility (mol/L)	Probability
Plutonium	$PuO_{2(am,\ hyd)}$	3.2E-11	0.25
	Normal Iron co-precipitation	7.4E-12	0.25
	Pessimistic Iron co-precipitation	1.6E-10	0.25
	Pessimistic No co-precipitation	1.1E-07	0.25
Neptunium	$NpO_{2(am,\ hyd)}$	2.5E-07	0.25
	Normal Iron co-precipitation	4.4E-14	0.25
	Pessimistic Iron co-precipitation	3.0E-13	0.25
	Pessimistic No co-precipitation	1.0E-04	0.25
Technetium	No solubility control	Instantaneous release	0.25
	Normal Iron co-precipitation [a]	1.1E-13	0.25
	Pessimistic Iron co-precipitation	1.3E-12	0.25
	Pessimistic No co-precipitation	Instantaneous release	0.25
Uranium	$UO_3{\cdot}2H_2O$	5.1E-05	0.25
	Normal Iron co-precipitation	2.3E-11	0.25
	Pessimistic Iron co-precipitation	1.8E-10	0.25
	Pessimistic No co-precipitation	3.4E-04	0.25

(a): Iron co-precipitation values used for Technetium in deterministic simulations.

Table RAI-NF-9.3: Probability Distributions for Various Phases Controlling Oxidized Region III Solubility

	Controlling Phase	Solubility (mol/L)	Probability
Plutonium	$PuO_{2(am,hyd)}$	3.2E-11	0.25
	Normal Iron co-precipitation	1.5E-13	0.25
	Pessimistic Iron co-precipitation	1.6E-10	0.25
	Pessimistic No co-precipitation	1.1E-07	0.25
Neptunium	$NpO_{2(am,hyd)}$	1.7E-06	0.25
	Normal Iron co-precipitation	8.8E-16	0.25
	Pessimistic Iron co-precipitation	3.0E-13	0.25
	Pessimistic No co-precipitation	1.0E-04	0.25
Technetium	No solubility control	Instantaneous release	0.25
	Normal Iron co-precipitation	2.1E-15	0.25
	Pessimistic Iron co-precipitation	1.3E-12	0.25
	Pessimistic No co-precipitation	Instantaneous release	0.25
Uranium	$UO_3{\cdot}2H_2O$	4.3E-06	0.25
	Normal Iron co-precipitation	4.5E-13	0.25
	Pessimistic Iron co-precipitation	1.8E-10	0.25
	Pessimistic No co-precipitation	3.4E-04	0.25

Table RAI-NF-9.4: Probability Distributions for Various Phases Controlling Submerged Condition C Solubility

	Controlling Phase	Solubility (mol/L)	Probability
Plutonium	$PuO_{2(am,hyd)}$	3.3E-11	0.25
	Normal Iron co-precipitation	3.2E-10	0.25
	Pessimistic Iron co-precipitation	1.6E-10	0.25
	Pessimistic No co-precipitation	1.1E-07	0.25
Neptunium	$NpO_{2(am,hyd)}$	9.9E-10	0.25
	Normal Iron co-precipitation	6.1E-13	0.25
	Pessimistic Iron co-precipitation	3.0E-13	0.25
	Pessimistic No co-precipitation	1.0E-04	0.25
Technetium	$TcO_2.1.6H_2O$	4.0E-09	0.25
	Normal Iron co-precipitation	2.7E-12	0.25
	Pessimistic Iron co-precipitation	1.3E-12	0.25
	Pessimistic No co-precipitation	Instantaneous release	0.25
Uranium	$UO_{2(am)}$	4.5E-09	0.25
	Normal Iron co-precipitation	1.8E-10	0.25
	Typical No co-precipitation [a]	4.3E-06	0.25
	Pessimistic No co-precipitation	3.4E-04	0.25

(a): Note that for uranium, the normal iron co-precipitation value is the pessimistic iron co-precipitation value. Therefore, for solubility variability and conservatism the “typical no iron co-precipitation” value is assumed as one of the sampled values in Submerged Condition C.

Table RAI-NF-9.5: Probability Distributions for Various Phases Controlling Submerged Condition D Solubility

	Controlling Phase	Solubility (mol/L)	Probability
Plutonium	$PuO_{2(am,hyd)}$	3.2E-11	0.25
	Normal Iron co-precipitation	1.2E-13	0.25
	Pessimistic Iron co-precipitation	1.6E-10	0.25
	Pessimistic No co-precipitation	1.1E-07	0.25
Neptunium	$NpO_{2(am,hyd)}$	1.8E-05	0.25
	Normal Iron co-precipitation	2.3E-16	0.25
	Pessimistic Iron co-precipitation	3.0E-13	0.25
	Pessimistic No co-precipitation	1.0E-04	0.25
Technetium	No solubility control	Instantaneous release	0.25
	Normal Iron co-precipitation	1.0E-15	0.25
	Pessimistic Iron co-precipitation	1.3E-12	0.25
	Pessimistic No co-precipitation	Instantaneous release	0.25
Uranium	$UO_3 \cdot 2H_2O$	1.8E-06	0.25
	Normal Iron co-precipitation	6.7E-14	0.25
	Pessimistic Iron co-precipitation	1.8E-10	0.25
	Pessimistic No co-precipitation	3.4E-04	0.25

Figure 5.6-31 of the HTF PA is reproduced here as Figure RAI-NF-9.1 and shows results to 20,000 years after final facility closure. This figure is presented to show the results of the solubility distributions that were sampled in the HTF PA. Figure RAI-NF-9.2 shows the equivalent results, using the revised sampling distributions considered above.

Figure RAI-NF-9.1: Statistical Time History of MOP Doses for Base Case (0 to 20,000 Years), Using HTF PA Solubility Distributions

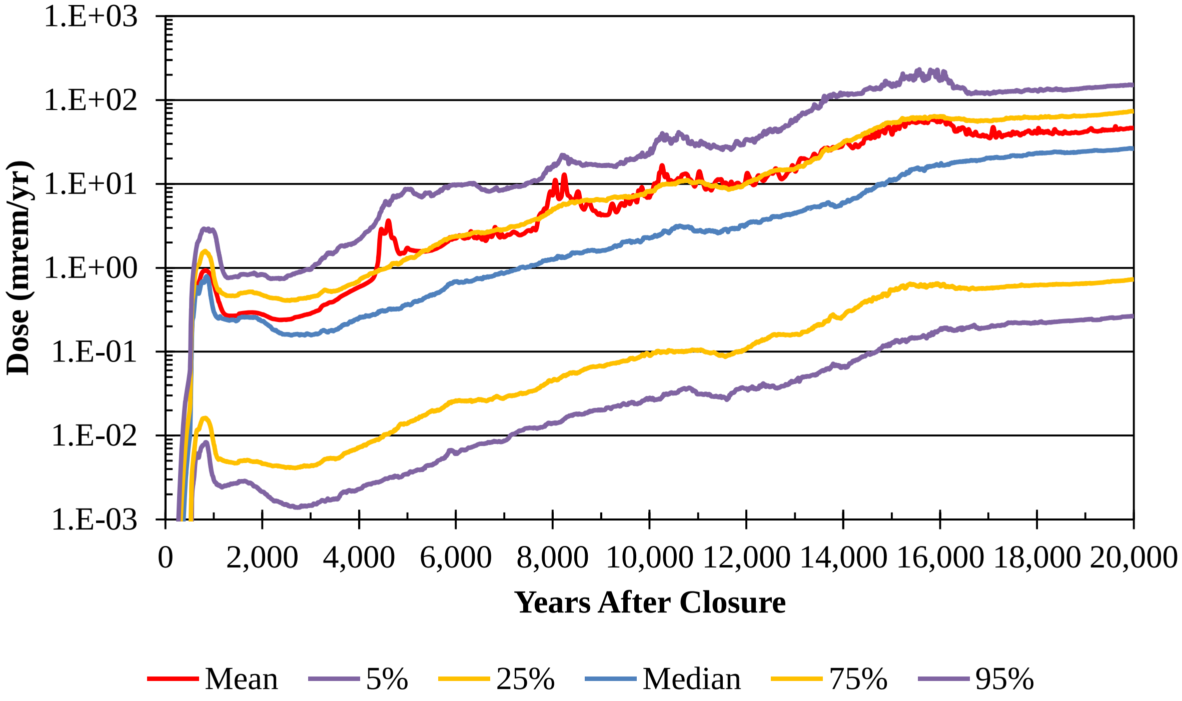
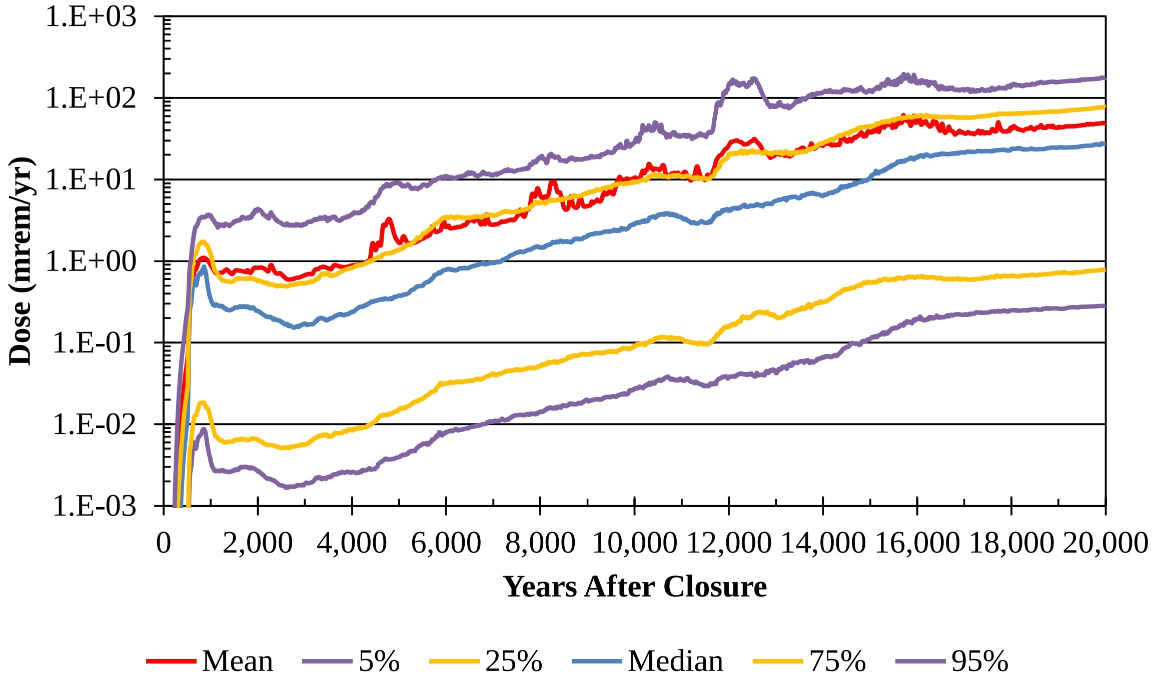


Figure RAI-NF-9.2: Statistical Time History of MOP Doses for Base Case (0 to 20,000 Years), Using Revised Solubility Distributions



The resulting doses are not significantly different from those reported in the HTF PA. The minor differences are more noticeable when comparing specific radionuclide dose contributions. Figures RAI-NF-9.3 and RAI-NF-9.4 show the mean radionuclide dose contributions from the HTF PA and from using the revised solubility sampling, respectively. Most notable is the change to the behavior in the Tc-99 dose contribution, which has less variability in early times due to higher solubility limits in the submerged waste tanks for Condition C.

Figure RAI-NF-9.3: Radionuclide-Specific Mean MOP Dose any Sector within 20,000 Years, Base Case, Using HTF PA Solubility Distributions

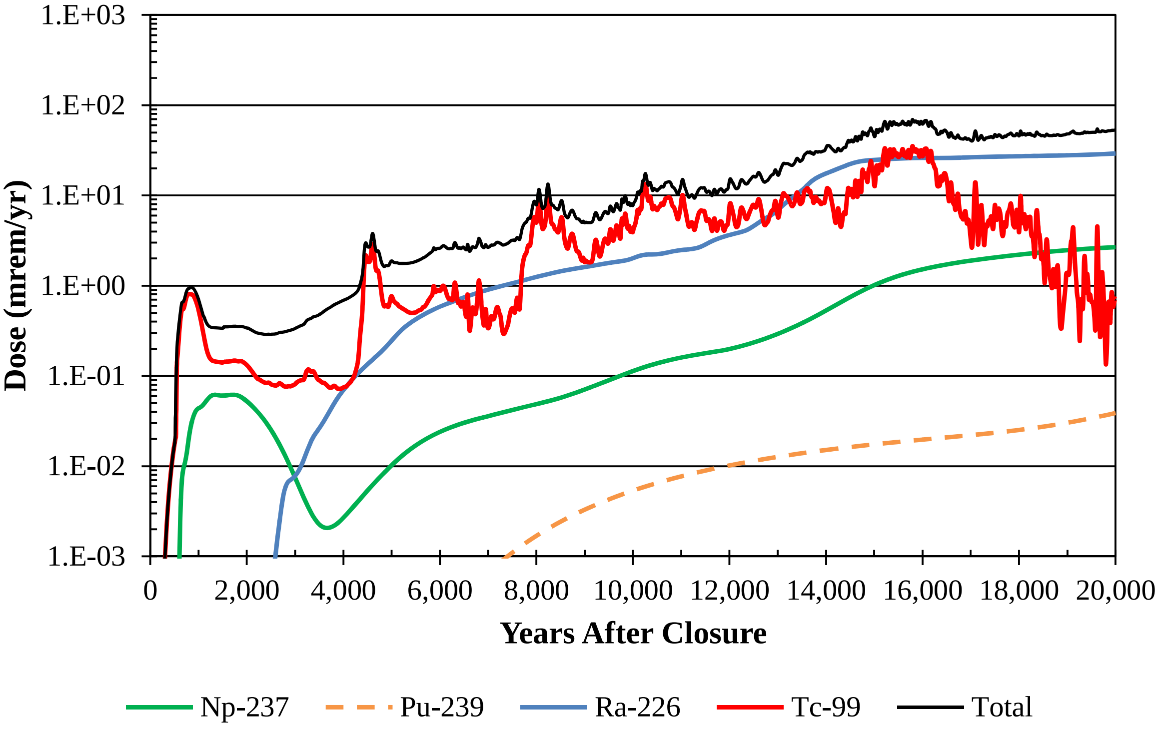


Figure RAI-NF-9.4: Radionuclide-Specific Mean MOP Dose any Sector within 20,000 Years, Base Case, Using Revised Solubility Distributions

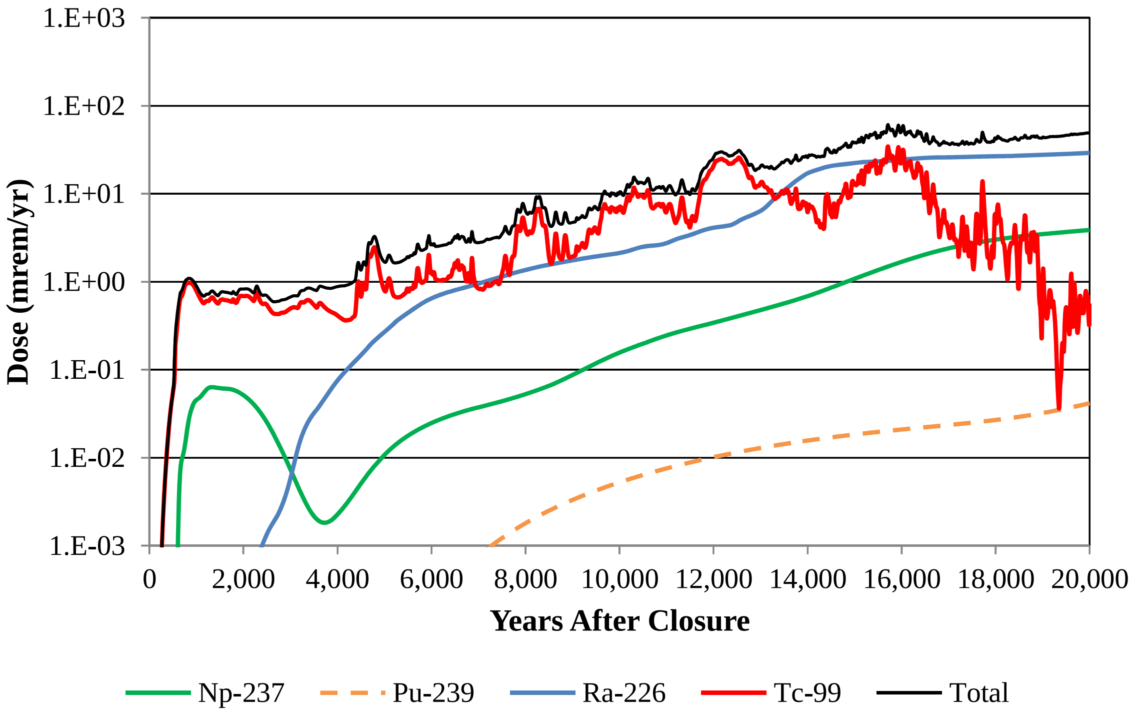


Table RAI-NF-9.6 provides a summary of the resulting peaks of the mean doses. These values show that the revised sampling approach provides doses that are not appreciably different from those in the HTF PA. Introducing additional uncertainty through increased solubility variability does not have a significant impact on the overall dose results.

Table RAI-NF-9.6: Comparison of Peaks of Mean Doses Using HTF PA Solubility Sampling vs. Revised Solubility Sampling

Peaks	Np-237	Pu-239	Ra-226	Tc-99	Total[a]
Using HTF PA Solubility Sampling					
0-1,000 Years	0.04	< 1.0E-3	< 1.0E-3	0.81	0.93
0-10,000 Years	0.11	0.01	2.0	10	13
0-20,000 Years	2.7	0.04	29	35	63
Using Revised Solubility Sampling					
0-1,000 Years	0.05	< 1.0E-3	< 1.0E-3	0.98	1.1
0-10,000 Years	0.16	0.01	2.1	7.6	11
0-20,000 Years	3.9	0.04	29	35	61

(a): Different radionuclides will impact individual sector doses in varying ways; therefore, the total peak dose may not always be equal to the sum of the individual peak doses from each of the radionuclides in individual sectors.

RAI-NF-10

The assumption that there is not a preferential pathway through the tank vaults and grout in the base case is not well supported. Consideration of a preferential pathway is especially important for Type I and II tanks that are submerged or partially submerged, with several containing significant quantities of radionuclides outside of the primary liner.

Basis

DOE assigns a probability of 75% to the base case (Table 5.6-5 of the HTF Performance Assessment), assuming that preferential pathways are not likely to occur throughout the entire period of performance. Based on historical evidence of waste release from Tank 16H into the environment (DP-1358) and operational observations of groundwater in-leakage over a relatively short timeframe, NRC staff are 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 Performance Assessment. 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. In addition, the grouting of the tanks will reduce the hydraulic head associated with the submerged and partially submerged tanks; however, the hydraulic gradient of the Upper Three Runs aquifer will still provide a small hydraulic driving force.

Path Forward

DOE should provide additional technical basis for assuming that preferential pathways are a low probability scenario in light of observations suggesting preferential pathways already exist through concrete vaults. Alternatively, DOE could include the presence of preferential pathways in their base case analysis or treat the various cases as independent conceptual models and report the conditional results (i.e., unweighted by their likelihood).

Response RAI-NF-10

Given the designed conservatism in the modeling approach described in Section 7.2 of the HTF PA, DOE believes that the selected assumptions and parameter values are appropriate for understanding the future risk from HTF closure activities on human health and the environment.

Consideration for Preferential (or Fast Flow) in the HTF PA

Although not included explicitly in the Base Case (Case A) of the HTF PA, DOE's hybrid modeling approach does include fast flow modeling configurations through the uncertainty analyses and sensitivity analyses models (specifically deterministic Cases B, C, D, and E and the probabilistic "All Cases" HTF PA models). Additionally, a number of the reasonably conservative modeling assumptions used in various cases approximate the effects of preferential flow paths.

Basis for Excluding Fast Flow Modeling from the Base Case Configuration

The fast flow path modeling configurations are not appropriate for Base Case modeling. The DOE's near-field models based on the PORFLOW code are 2-D axi-symmetric representations of closed waste tanks. The fast flow paths in the alternative model configurations implicitly fully circumscribe the waste tanks. These hypothetical scenarios were developed as thought exercises to better understand system behavior when there is maximum contact between infiltrating water and the waste layer, and the waste tank fill grout chemistry has no influence in waste layer chemistry. For example, this effect is achieved in Cases D and E by creating an outer annular entry path (between the grout and the primary liner) and an exit path at the waste tank centerline, with both entry and exit fully connected to the external environment. Thus, the modeling configuration assumes the entire waste layer is contacted by water flowing radially inward at a high rate.

While fast flow path scenarios such as Cases D and E are useful sensitivity cases to help bound consequences, they are not predicted configurations nor reflective of expected conditions. The DOE acknowledges the possibility of gaps and/or higher permeability zones that could form an isolated pathway for faster flow within a closed waste tank. However, DOE believes such pathways would be limited in their connectivity throughout the system and/or influence only a fraction of the waste layer compared to the modeling scenarios. Therefore, hypothetical fast flow configurations such as Cases D and E are not considered reflective of probable future conditions. Rather, these fast flow path cases are considered to be bounding (i.e., worst-case) representations of various fast flow path scenarios and assigned lower probabilities in the probabilistic simulations.

The specific inclusion of a fast flow path within the Base Case model is deemed inappropriate for the following reasons: 1) the design of the fast flow configurations maximize the risk to an extent that is overly conservative (i.e., unrealistic); 2) the bias towards conservatism already in the Base Case, and applications of the risk-informed approach would be compounded by the inclusion of fast flow paths; and 3) the use of alternative models in the hybrid approach already provide a vehicle for evaluating the uncertainty and risks associated with fast flow paths.

Basis for Waste Tank Configuration Weightings in the “All Cases” Model

The probability weightings used to sample the various waste tank configurations in the “All Cases” probabilistic model of the HTF PA was provided in Table 5.6-5 of the HTF PA. These weighting assignments were developed using a qualitative but informed approach because a quantitative estimate was not directly possible. The rationale used to assign probabilities of occurrence (weightings) for the Base Case and the four alternative configurations is discussed below.

These weightings were designed to provide insights into the significance of various modeling parameters with respect to risk. The Base Case is considered a realistic combination of probable and conservative assumptions that provides reasonably conservative results. The Base Case assumptions represent the best estimate for each assumption (with a bias toward conservatism) using currently available information. Given this position, and the intent of the probabilistic model, applying greater weight to the sampling of the Base Case configuration is appropriate and served as the starting point for the probability weighting assignments.

The fast flow path configurations represented by Cases B, C, D, and E were not considered likely to occur immediately, at the time of closure, or within the first 1,000 to 10,000 years of performance, especially with the waste tank liners remaining mainly intact. Therefore, the Base Case was assigned a 75 % probability of occurrence, and the remaining fast flow cases a collective probability of 25 %.

Cases B through E generally fall in two subgroups: 1) Cases B and C model partial fast flow paths (i.e., through the grout but not the basemat); and 2) Cases D and E model full fast flow paths (i.e., through the grout and the basemat). The grout is considered more likely to have a fast flow path (e.g., shrinkage gap) relative to the basemat thus a lower probability is applied to Cases D and E. Therefore, Cases B and C were given a probability of occurrence equal to 20 %, while D and E have the collective probability equal to 5 %.

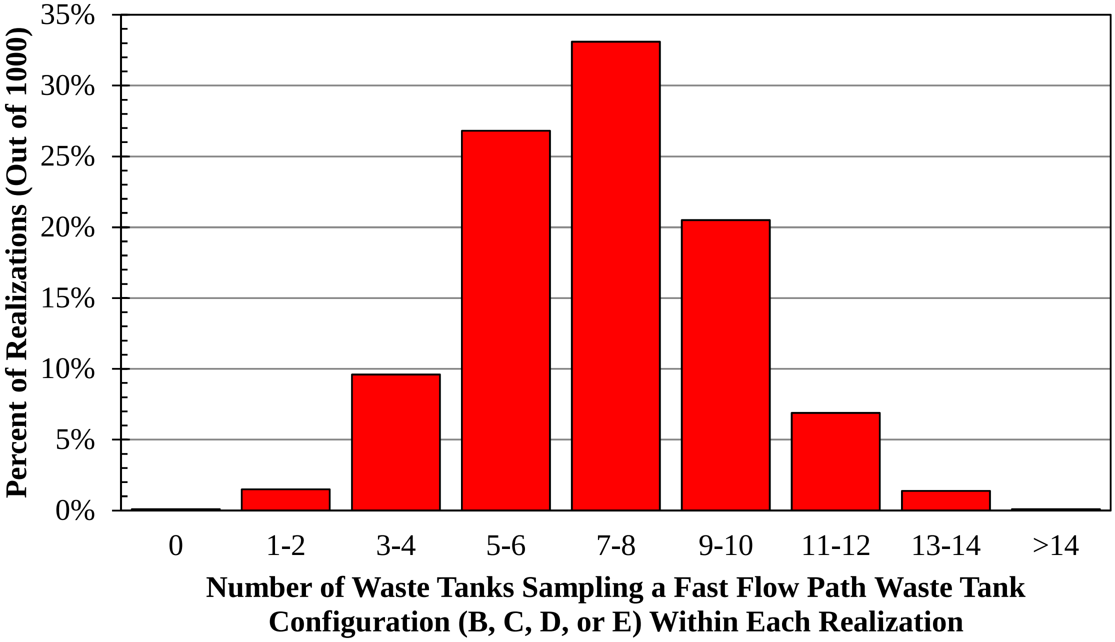
The probabilities applied to Cases B and C and to Cases D and E were further refined assuming the likelihood of instantaneous cementitious material degradation at, or before, 501 years following HTF final facility closure. A gradual cementitious degradation was considered more likely; therefore, Cases B and D were given lower probabilities of occurrence relative to Cases C and E. It should be noted that while Case E is not the case with the lowest weighting, there are numerous factors (as discussed in the response to CC-INT-2) that jointly make Case E an extremely unlikely scenario.

Waste Tank-Specific Sampling

It is also important to note that variability in the waste tank configurations is sampled independently in the probabilistic model for each waste tank. With 29 waste tanks in the HTF system, any single probabilistic realization is likely to sample multiple waste tank configurations. As such, the probability of all waste tanks within the HTF system modeling the same configuration is much smaller (i.e., even realizations with mostly Case A waste tank configurations generally include some waste tanks with fast flow paths that potentially increase the peak doses).

Figure RAI-NF-10.1 shows how sampling waste tank configurations independently for each waste tank results in multiple realizations that include consideration for the effects of fast flow. More than one-third of the realizations in this set of 1,000 “All Cases” realizations included seven or eight waste tanks (out of 29) that were modeled with fast flow path configurations. Almost 30 % of the realizations sampled nine or more waste tanks with fast flow path configurations. Only one realization out of 1,000 included no fast flow path configurations.

Figure RAI-NF-10.1: Waste Tank Configuration Sampling from the “All Cases” Probabilistic Model



Evaluation of Alternative Probability Weightings

Uncertainty around the Base Case is addressed in the HTF PA using additional supporting models (i.e., alternate modeling cases and probabilistic analyses). These additional models provide insights into the Base Case results and overall system performance.

Section 4.4.2 of the HTF PA explains that the alternative modeling configurations represent postulated conditions that may be present, without regard for the mechanisms leading to those conditions. There is a variety of mechanisms that can lead to fast flow paths penetrating from the top to the bottom of the waste tank grout. These alternate modeling configurations should not be interpreted as representing a specific mechanism for fast flow path formation or the belief that a given set of conditions would actually be present in the closed HTF system at some future point. The configurations modeled in Cases B, C, D, and E encompass various mechanisms and provide information on the risk-significance of early failure and fast flow paths, relative to the Base Case, as well as providing understanding of system performance while considering uncertainty and variability.

To evaluate further the impact on dose due to variations in the probability weightings assigned to each waste tank configuration, additional probabilistic models were developed. In these two alternative probability studies, the weightings for Cases B, C, D, and E were increased to provide additional insights regarding the sensitivity of doses relative to each of these alternative waste tank configurations. In the first alternative probability study, the weightings for these alternative waste tank configurations were doubled (e.g., the weighting for Case B increased from 5 % to 10 %); and in the second alternative probability study, the weightings for these alternative waste tank configurations were tripled (e.g., the weighting for Case B increased from 5 % to 15 %). Table RAI-NF-10.1 summarizes the various probability weighting assignments for each probabilistic run.

Table RAI-NF-10.1: Configuration Weightings for Probabilistic Modeling

Case	100 % Base Case Weighting (Case A)	75 % Base Case Weighting (All Cases)	50 % Base Case Weighting ("Alt. 1")	25 % Base Case Weighting ("Alt. 2")
A	100 %	75 %	50 %	25 %
B	0 %	5 %	10 %	15 %
C	0 %	15 %	30 %	45 %
D	0 %	1.25 %	2.5 %	3.75 %
E	0 %	3.75 %	7.5 %	11.25 %

Note that these alternative weightings do not reflect the expected or nominal conditions of this system; rather these values were artificially changed to evaluate the overall impact from increased variability among the various waste tank configurations.

Figures RAI-NF-10.2 through RAI-NF-10.5 show the mean, median, 75th percentile and 95th percentile dose results, respectively, from each set of probabilistic models. Table RAI-NF-10.2 shows peak doses from the statistical results for these various probabilistic models.

Figure RAI-NF-10.2: Mean Values from Four 1,000-Realization Probabilistic Modeling Runs

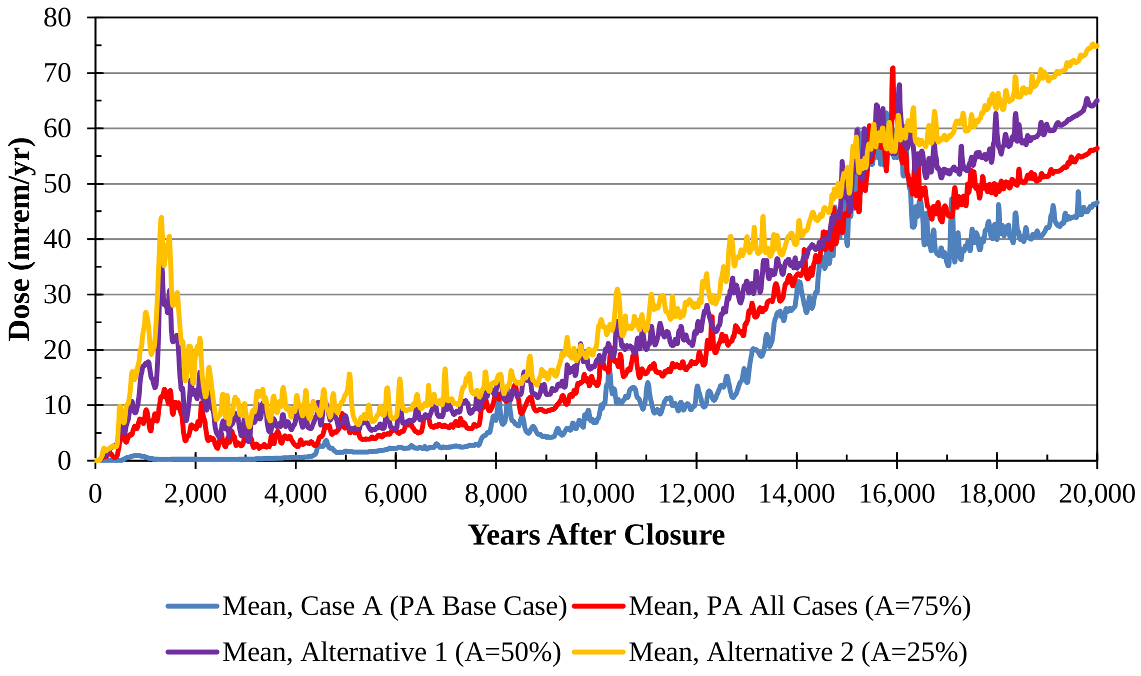


Figure RAI-NF-10.3: Median Values from Four 1,000-Realization Probabilistic Modeling Runs

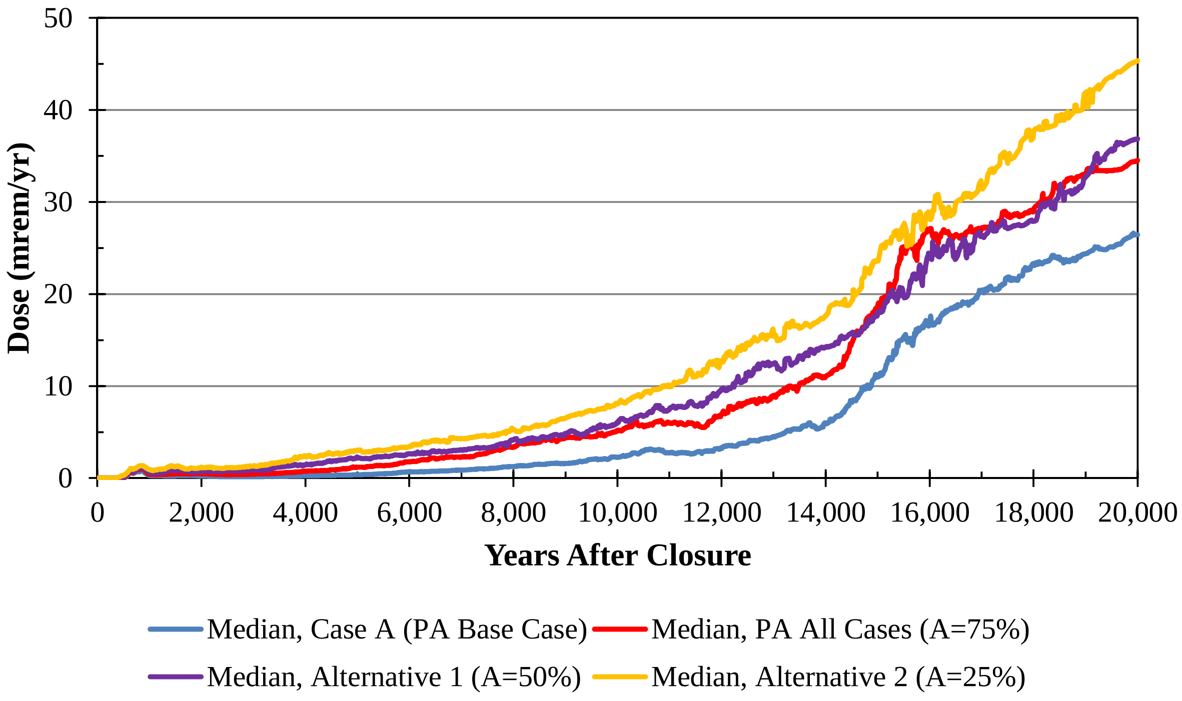


Figure RAI-NF-10.4: 75th Percentile Values from Four 1,000-Realization Probabilistic Modeling Runs

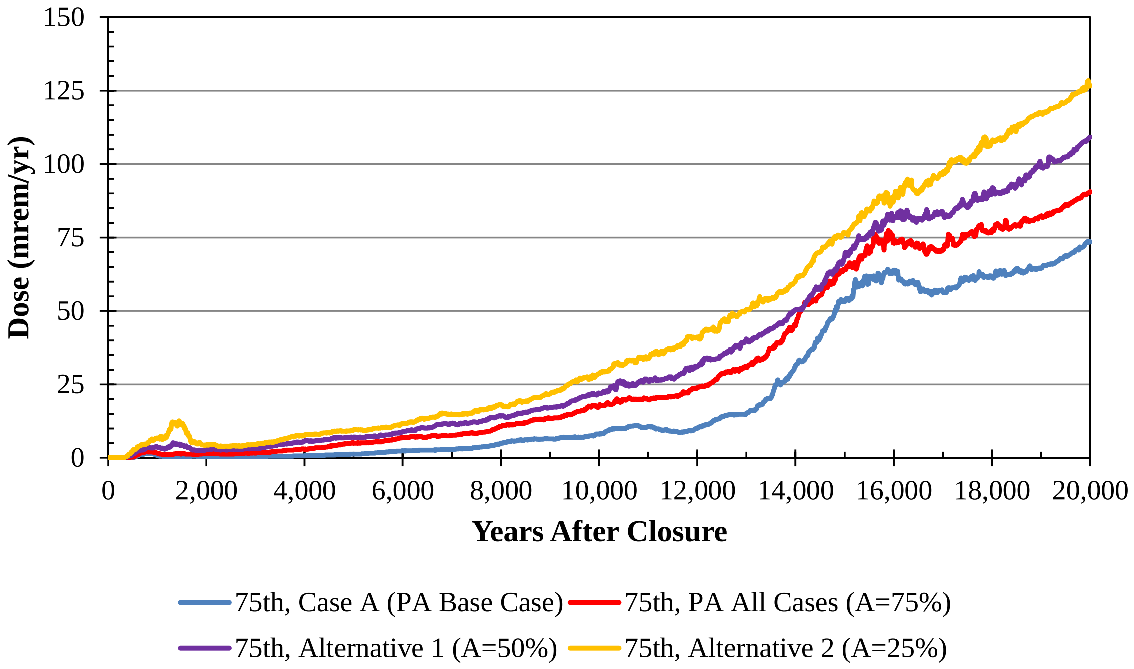


Figure RAI-NF-10.5: 95th Percentile Values from Four 1,000-Realization Probabilistic Modeling Runs

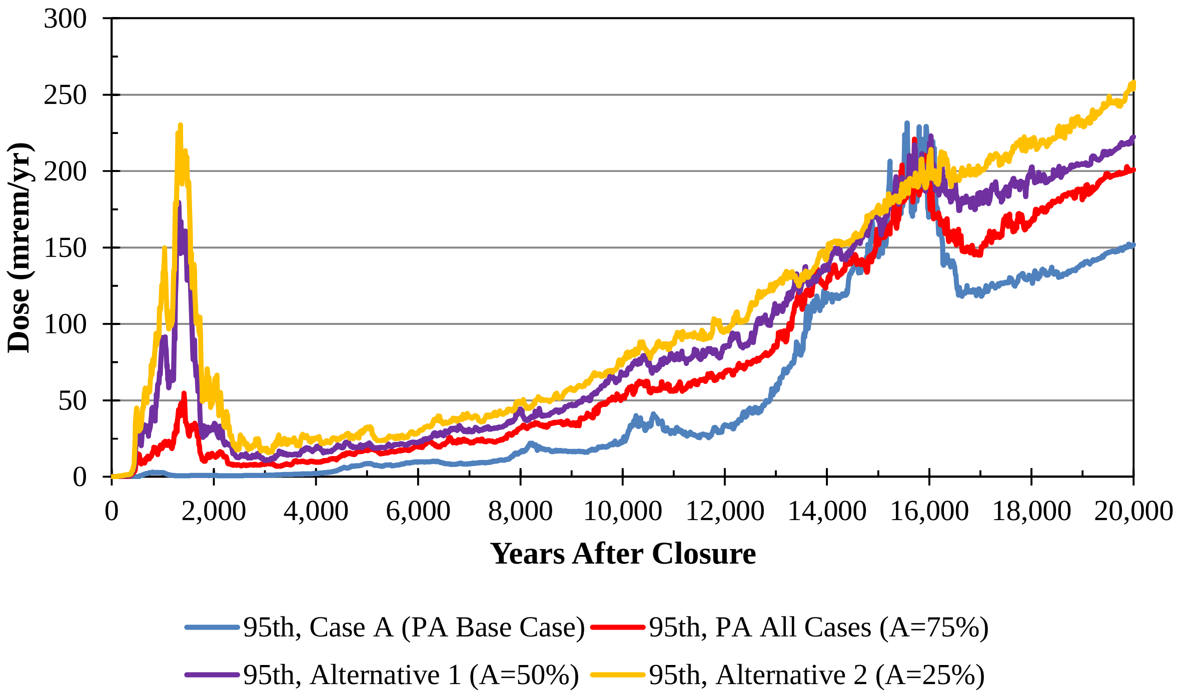


Table RAI-NF-10.2: Peak Doses from of the Statistical Results

Modeling Case	Peak of the Means (mrem/yr)	Peak of the Medians (mrem/yr)	Peak of the 75th Percentiles (mrem/yr)	Peak of the 95th Percentiles (mrem/yr)
Endpoint: MOP Dose within 10,000 years				
Case A (PA Base Case)	13	2.3	8.2	24
All Cases (Case A = 75 %)	15	5.6	18	58
Alternative Case 1 (Case A = 50 %)	36	6.1	22	180
Alternative Case 2 (Case A = 25 %)	44	8.1	29	230
Endpoint: MOP Dose within 20,000 years				
Case A (PA Base Case)	63	27	74	232
All Cases (Case A = 75 %)	71	35	91	221
Alternative Case 1 (Case A = 50 %)	68	37	109	223
Alternative Case 2 (Case A = 25 %)	75	45	128	258

Observations on the Probabilistic Model Results

The following observations are made from Table RAI-NF-10.2 and Figures RAI-NF-10.2 through RAI-NF-10.5:

- The differences between the shapes of the median and the mean curves indicate that the mean values are being driven higher by outlying realizations that represent a confluence of multiple unlikely factors (e.g., early liner failure, fast flow configurations for waste tanks nearest the HTF boundary, fast cementitious material degradation, conservative sampling of parameters in the dose calculations, etc.). This is also evident in the differences between the 75th and 95th percentile dose curves.
- In 20,000 years, the peaks of the 95th percentile statistical results do not vary significantly as a function of configuration weightings.
- Within each set of statistical results, the peaks vary by less than one order of magnitude.
- None of the median dose curves exceeds 25 mrem/yr before 14,000 years following HTF final facility closure.
- None of the 75th percentile dose curves exceed 25 mrem/yr before 9,000 years following HTF final facility closure.
- None of the curves, not even the 95th percentile dose curves, result in “significant” doses through 20,000 years.
- In summary, increasing the probability of fast flow configurations does increase the doses, as expected; however, the increases are not significant except when multiple low-probability conditions converge to drive up the doses. These observations are consistent with results already described within the HTF PA when discussing the probabilistic model results.

Conclusion

Although DOE agrees that fast flow paths are not explicitly modeled in the deterministic Base Case itself, the hybrid approach for modeling does include specific fast flow paths in both the deterministic sensitivity analysis (i.e., Cases B, C, D, and E) and the probabilistic uncertainty analysis. Additional analyses, as presented in this RAI response, further support these conclusions.

RAI-NF-11

The assumption that the flow of water through the preferential pathway is limited to water that has infiltrated through the closure cap, may underestimate the release of short-lived radionuclides for submerged and partially submerged tanks.

Basis

Alternative cases B-E in the HTF Performance Assessment are designed to account for mechanisms that could result in the occurrence of a preferential pathway. The HTF Performance Assessment assumes that flow through the preferential pathway is only from water that has infiltrated through the closure cap. The closure cap is designed to limit infiltration early in the performance period, providing time for shorter-lived radionuclides (e.g., Sr-90 and Cs-137) to decay to less significant levels. However, operational experience suggests that groundwater in-leakage through a preferential pathway could occur earlier in the performance period. The assumption of limiting flow through the preferential pathway to water that has infiltrated through the closure cap could significantly underestimate the release of short-lived radionuclides due to groundwater in-leakage.

Path Forward

DOE should evaluate the potential release of radionuclides due to groundwater in-leakage via a preferential pathway through the submerged and partially submerged tanks.

Response RAI-NF-11

As required by DOE Manual 435.1-1, maintenance of the HTF PA will include future updates to incorporate new information, update model codes, consider actual residual inventories, etc., as appropriate. Section 8.2 of the HTF PA states that as "additional data becomes available …additional modeling may be required." Each time additional modeling is performed, DOE will consider the need to update the Base Case (Case A) scenario and revise the PA. PA maintenance and the potential impact of "new information" are evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1. New fate and transport modeling will be performed as required through the SA process, replacing the PA assigned inventories with the final residual characterization data. If other "new information" has been identified (e.g., updated K_d values developed through research experimentation), this new data would also be evaluated through inclusion in the SA. The results of the waste tank-specific SA are then evaluated to determine if any of the new information impacts PA-based conclusions. While it typically would not be necessary to replicate all of the PA sensitivity and uncertainty analyses in an SA, it may be appropriate to include additional analyses to evaluate the new or unique waste tank-specific information.

As waste tank-specific SAs are prepared it is anticipated that sensitivity analyses will be performed that explicitly address both individual waste tank inventories and unique waste tank conditions, such as potential preferential pathways (e.g., degraded liners or in-leakage pathways). Because different as-modeled waste tank conditions may prove to be conservative or non-conservative (with respect to timing and/or magnitude of peak doses) depending on other corresponding waste tank conditions, these analyses are more suited to waste tank-specific sensitivity analyses than to changes to the overarching Base Case assumptions. For example, modeling assumptions that cause the annulus inventory to be released relatively slowly (thereby adding to releases from the waste tank primary inventory) versus released relatively rapidly impacts peak doses differently depending on the quantity (total curies) and nature (short-lived versus long-lived radionuclides) of the inventory remaining in a waste tank annulus. In a similar manner, preferential pathways such as failed liners can have very radionuclide-specific dose impacts, with earlier liner failures not necessarily corresponding to greater peak doses, as demonstrated in HTF PA Section 5.6.7.6 (Liner Failure Times Analysis using the PORFLOW Deterministic Model).

The various modeling configurations (Cases A through E) included in the HTF PA provide a range of outcomes from which various sub-configurations can be evaluated, even though the configurations themselves may represent non-mechanistic or unlikely conditions. For example, the hypothetical fast flow path scenarios were designed to produce maximum contact between infiltrating water and the waste layer. This effect is achieved in Case E by creating an outer annular entry path (between the grout and primary liner) and an exit path at the waste tank centerline, both fully connected to the external environment. This modeling approach results in the entire waste layer (which, in itself is modeled as a high-conductivity zone in Case E) being contacted by water flowing radially inward at a high rate. The chemistry of the waste tank reducing grout is assumed to have no influence in waste layer chemistry, despite its direct contact. It is then assumed all tanks would be subject to this condition. While fast flow path scenarios, such as Case E, are useful sensitivity configurations to help bound consequences, they are meant to be simplifications, and are not meant to be predictive nor reflective of precise conditions for an entire waste tank type. With respect to preferential pathways, the current modeling configurations are expected to bracket any preferential pathway scenario. For example, in the contamination zone, the radially inward Darcy velocities for Case E reach magnitudes much larger than the Darcy velocities in saturated zone. Groundwater in-leakage via a preferential pathway through the submerged and partially submerged waste tanks would not significantly increase the peak dose since the presence of annulus grout would inhibit in-leakage through the waste tank vault. In the absence of preferential pathways that traverse the entire waste tank structure, lateral pathways would not be expected to accelerate outward migration of contaminants. In addition, the near-field waste fluxes are typically dominated by downward flow (driven by infiltration) not by horizontal flow.

RAI-NF-12

In Tanks 9H and 10H (Type I), the loading of the annular source term in the reducing grout and the location of the preferential pathway appear to be unrealistic.

Basis

In DOE's PORFLOW® model (see Figure 1), the annular waste in Tanks 9H and 10H is loaded within the bottom of the annular reducing grout. Loading of the annular waste into the bottom of the reducing grout in DOE's PORFLOW® model assigns 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 cases B-E does not intersect the annular waste in Tanks 9H and 10H. Based on operational experience, NRC staff is concerned that groundwater in-leakage into the annular region could occur and that the associated risk with this scenario is not adequately addressed within the Performance Assessment.

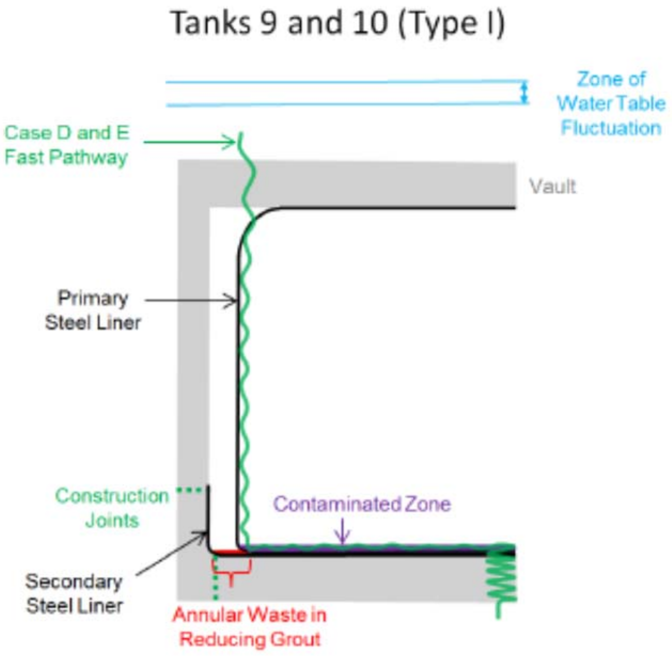


Figure 1. Illustration of the as-modeled annular waste and fast pathway in Tanks 9H and 10H

Path Forward

DOE should 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 Tanks 9H and 10H.

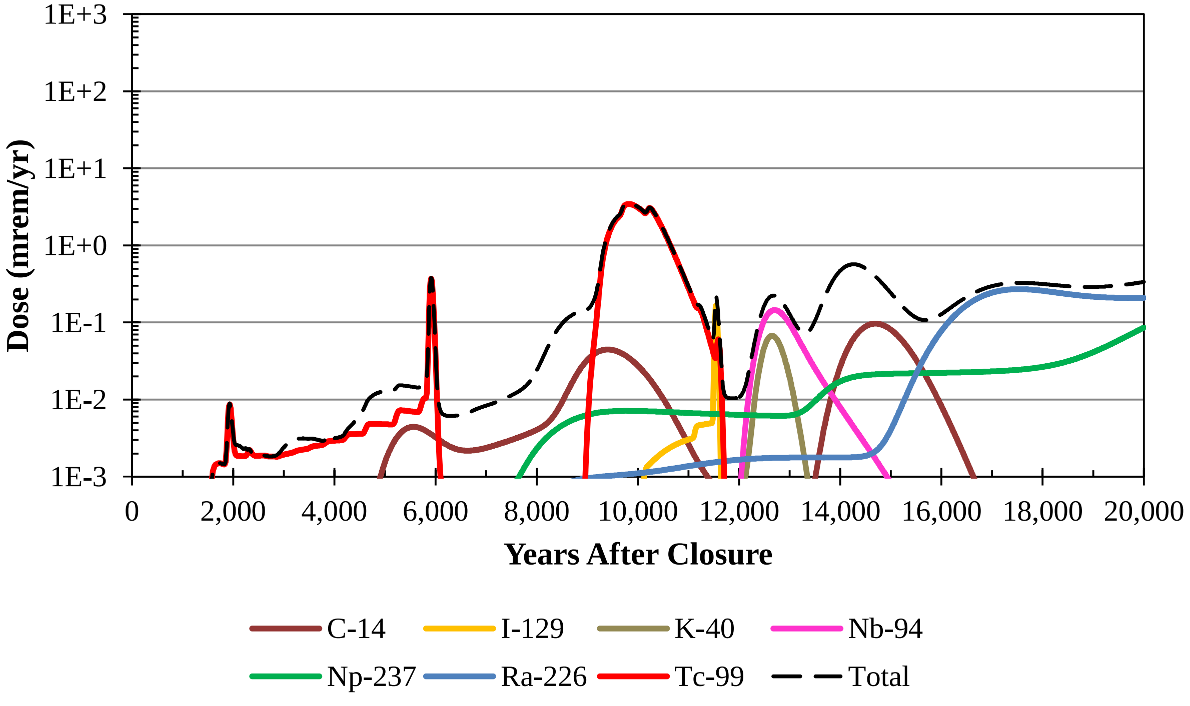
Response RAI-NF-12

Waste inventories assumed within the annulus are not expected to contribute significantly to risk. When modeling the annulus and sand pads, no special effort was made to prevent material in these regions from becoming held up within the annular grout or contamination zone of the waste tanks (i.e., adding to the in-tank inventories) because this modeling assumption is expected to maximize the overall HTF peak doses.

As discussed in the response to RAI-NF-8, diffusion of contaminants from the annulus and sand pads inventories into annular or waste tank grout is an unintended consequence of a modeling simplification. The response to RAI-NF-8 described that this modeling approach is believed to be conservative and not inappropriate in the context of the HTF PA and its associated purposes.

For further insight, and to address more quantitatively this RAI the GoldSim model was used in deterministic mode to develop a set of additional figures showing the radionuclide-specific dose contributions from the assumed annular inventories. These figures show the peak radionuclide doses, regardless of sector. Figure RAI-NF-12.1 shows the radionuclide contributions to the MOP dose from only the annular inventories, using Base Case (Case A) modeling assumptions.

Figure RAI-NF-12.1: MOP Radionuclide Dose Contributions Using Base Case Modeling Assumptions (Annular Inventories)

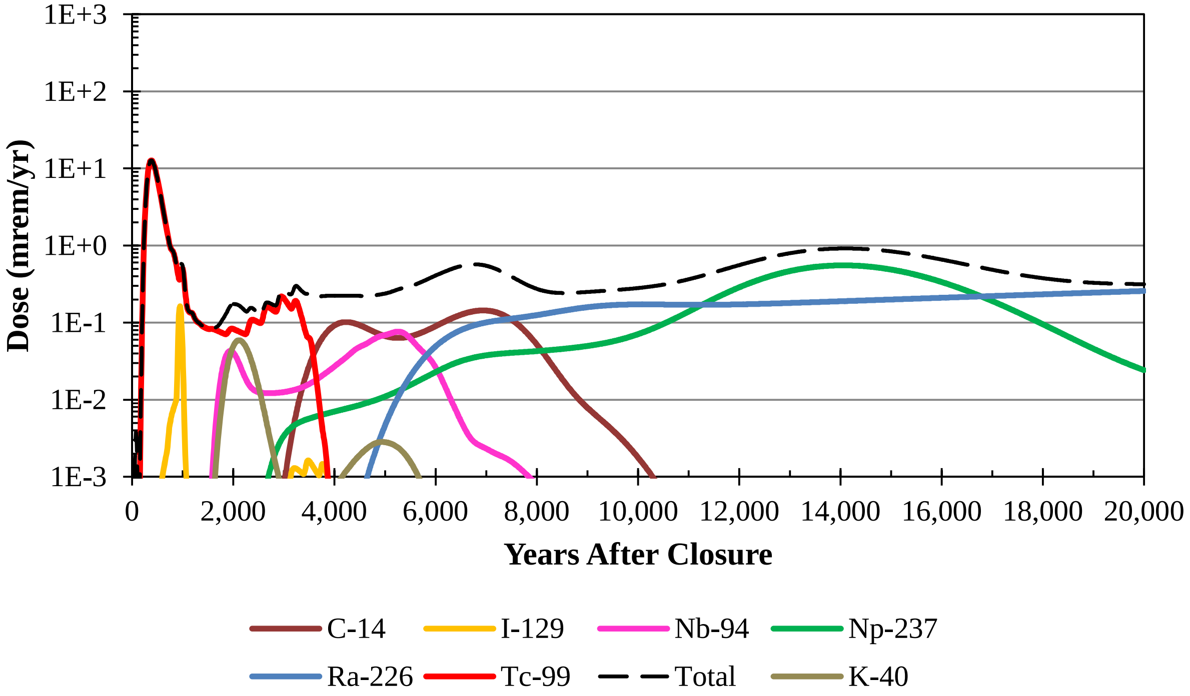


The peak dose in Figure RAI-NF-12.1, near 9,800 years after final facility closure, occurs due to the oxidation of the annular grout for the Type I tanks with intact liners (Tanks 9, 10, and 11). Note that in the GoldSim model, this transition occurs around 8,700 years. Prior to this transition, less than 1 % of the Tc-99 within the annulus inventories of Tanks 9, 10, and 11 is released. The combined annulus inventory doses from these three waste tanks peaks at approximately 3.5 mrem/yr.

In comparison, the primary and secondary waste tank liners are assumed initially failed in Tank 12, allowing more groundwater to flow through the waste tank such that the annular grout transitions earlier (around 6,000 years after final facility closure). The Tc-99 in the Tank 12 annulus inventory provides a peak dose of approximately 0.4 mrem/yr.

Figure RAI-NF-12.2 shows the radionuclide contributions to the MOP dose, regardless of sector, from only the annular inventories, using pessimistic modeling assumptions. These assumptions are defined in the response to RAI-NF-8 (for the model identified as "Flow Run 65, No Holdup"). This pessimistic model releases the majority of the contaminants at the start of the simulation and includes assumptions that accelerate flow.

Figure RAI-NF-12.2: MOP Radionuclide Dose Contributions Using Pessimistic Modeling Assumptions (Annular Inventories)



In this pessimistic modeling case, 50 % of the Tc-99 inventories associated with the annular contributions are released within the first 500 years and more than 99 % are released in the first 3,000 years. In this extremely pessimistic modeling scenario, the total dose from the annular contributions within all the HTF waste tank annulus inventories peaks at approximately 13 mrem/yr (around 400 years after final facility closure). This compares to the Base Case modeling approach, which peaked at less than 4 mrem/yr (around 9,800 years after final facility closure). These results indicate that earlier release of the annulus inventories is not likely to alter the risk significance.

Future PA Revisions and the Role of SAs

As required by DOE Manual 435.1-1, maintenance of the HTF PA will include future updates to incorporate new information, update model codes, consider actual residual inventories, etc., as appropriate. Section 8.2 of the HTF PA states that as "additional data becomes available …additional modeling may be required." Each time additional modeling is performed, DOE will evaluate whether or not an update to the Base Case or a revision of the PA is needed. PA maintenance and the potential impact of new information is evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1. New fate and transport modeling will be performed, as required, through the SA process, replacing PA assigned inventories with the final residual characterization data. If other new information has been identified (e.g., updated K_d values developed through research or experimentation), this new data would also be evaluated through inclusion in the SA. The results of the waste tank-specific SAs are then evaluated to determine if new information impacts PA-based conclusions. While it typically would not be necessary to replicate all of sensitivity and uncertainty analyses from the PA as part of a SA, it may be appropriate to include additional analyses to evaluate the new or unique waste tank-specific information.

As waste tank-specific SAs are prepared it is anticipated that sensitivity analyses will be performed to address explicitly both individual waste tank inventories and unique waste tank conditions, such as potential preferential pathways (e.g., degraded liners or in-leakage pathways). Because different as-modeled waste tank conditions may prove to be conservative or non-conservative (with respect to timing and/or magnitude of peak doses) depending on other corresponding waste tank conditions, these analyses are more suited to waste tank-specific sensitivity analyses than to changes to the overarching Base Case assumptions. For example, modeling assumptions that cause the annulus inventories to be released relatively slowly (thereby adding to releases from the in-tank inventories) versus relatively rapidly would impact peak doses differently depending on the quantity (total curies) and nature of the inventory remaining in a waste tank annulus (e.g., short-lived versus long-lived radionuclides). In a similar manner, preferential pathways such as failed liners can have very radionuclide-specific dose impacts, with earlier liner failures not necessarily corresponding to greater peak doses, as demonstrated in HTF PA Section 5.6.7.6 (Liner Failure Times Analysis using the PORFLOW Deterministic Model). Additionally, if the closure inventory for a specific radionuclide is determined to be significantly different than previously projected, the impact of the modeling simplification discussed in the RAI would need to be further evaluated.

RAI-NF-13

In Tanks 14H and 16H (Type II), it is not clear to what extent the preferential pathway interacts with the waste located in the primary and secondary sand pads.

Basis

The potential release of radionuclides from the sand pads in Tanks 14H and 16H could be limited by the amount of water flowing through the preferential pathway and/or diffusion of the radionuclides out of the sand pads (See RAI-NF-8). The sand pads in Tanks 14H (primary sand pad only) and 16H (primary and secondary sand pads) contain a significant amount of activity. In the HTF Performance Assessment, 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). The extent to which the sand pad inventories are contacted by flow in the preferential pathway is not clear. In addition, the HTF Performance Assessment does not discuss the extent of diffusion out of the sand pads and into the adjacent cementitious materials.

It appears that a potentially significant fraction of the highly radioactive radionuclide inventory is diffusing into the basemat and/or contaminated zone prior to release. Diffusion of radionuclides out of the primary and secondary sand pads is facilitated in the model by a high diffusion coefficient, large concentration gradient, small diffusion length, assumption of no steel liners, and a delay in the flow through the fast pathway due to closure cap. Although the steel liners are not assumed to not be intact for Tanks 14H and 16H due to the large number of leak sites, the steel liner could still act as a partial barrier to diffusion. Also, the delay in flow through the preferential pathway, due to the assumption of flow being limited to infiltration through the closure cap, may overestimate the amount of time radionuclides can diffuse out of the sand pads if groundwater in-leakage were to occur.

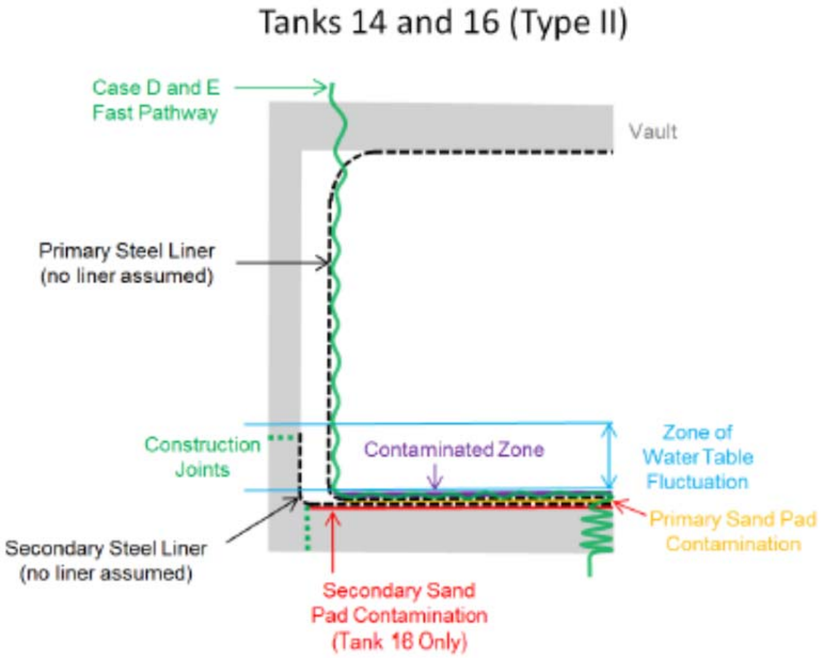


Figure 2. Illustration of the as-modeled sand pad waste and fast pathway in Tanks 14H and 16H

Path Forward

DOE should discuss the extent to which the preferential flow path affects the waste located in the sand pads and its risk significance. This should include the fraction of the inventory of the short-lived radionuclides (e.g., cesium-137 and strontium-90) that decay prior to significant flow occurring in the fast flow path. DOE should also provide discussion regarding the fraction of the highly radioactive radionuclides (e.g., technetium, plutonium, and neptunium) that diffuse out of the sand pads and into the grout or basemat.

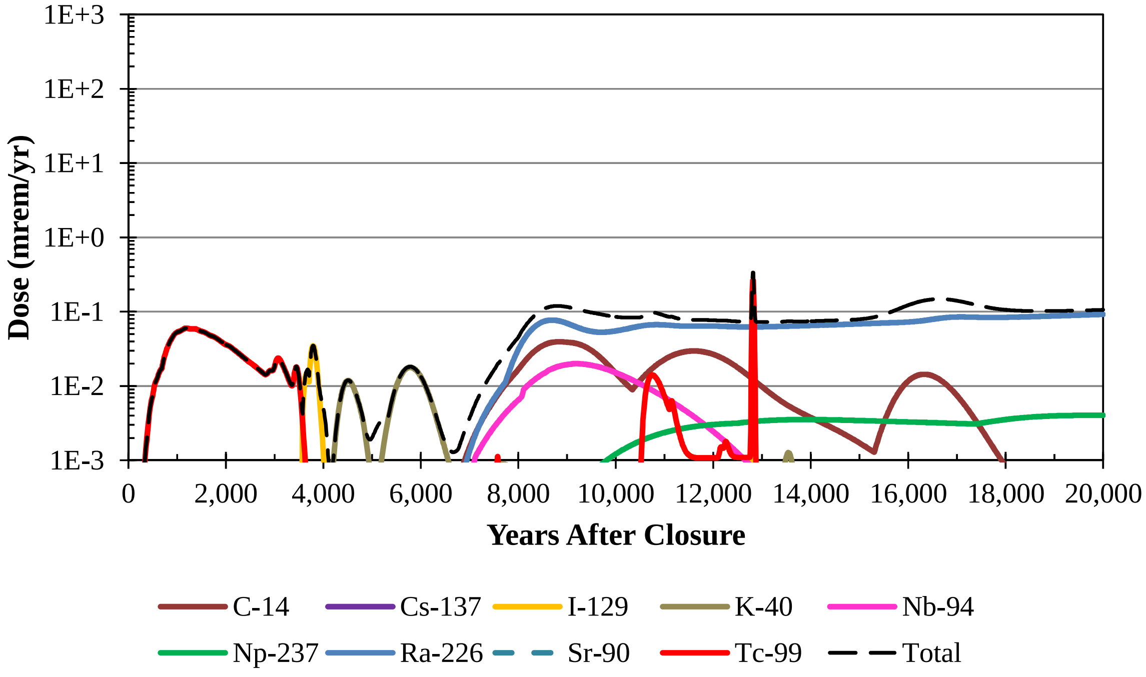
Response RAI-NF-13

Waste inventories assumed within the sand pads are not expected to contribute significantly to risk despite assuming what is believed to be bounding values relative to the assigned inventories within these sand pads. When modeling the annulus and sand pads, no special effort was made to prevent material in these regions from migrating into the contamination zone within the waste tanks (i.e., adding to the in-tank inventories) because this modeling assumption is expected to maximize the overall HTF peak doses.

As discussed in the response to RAI-NF-8, diffusion of contaminants from the annulus and sand pads inventories into annular or waste tank grout is an unintended consequence of a modeling simplification. As discussed in the response to RAI-NF-8, the simplified approach taken facilitates modeling of solubility controls that are appropriate under most conditions.

For greater insight, the GoldSim model was used in deterministic mode to develop a set of additional figures showing the radionuclide-specific dose contributions from the assumed sand pads inventories. These figures show the peak radionuclide doses, regardless of sector. Figure RAI-NF-13.1 shows the radionuclide contributions to the MOP dose from only the sand pads inventories (i.e., Tanks 13-16 sand pads) using Base Case (Case A) modeling assumptions.

Figure RAI-NF-13.1: MOP Radionuclide Dose Contributions (from the Sand Pads) Using Base Case Modeling Assumptions



The peak dose in this figure occurs once the primary liner for Tank 13 fails (around 12,700 years after closure). As an artifact of the modeling approach, more than half (approximately 55 %) of the Tc-99 artificially migrates (via diffusion) from the sand pads, through the primary and secondary liners, and into the contamination zone and remains there until liner failure.

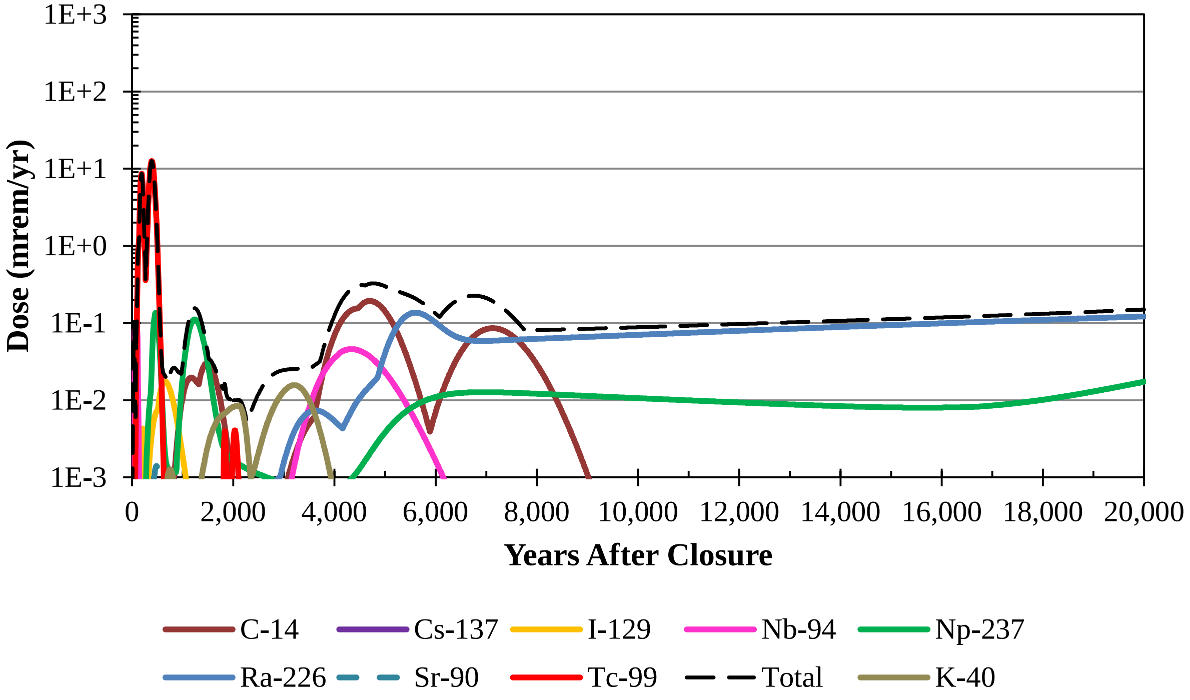
Alternatively, this diffusion effect is minimized for Tanks 14, 15, and 16 because the majority of the available inventory is advectively transported out of the contamination zone (due to assumed liner failure at the start of the simulation). Figure RAI-NF-13.1 shows a low magnitude Tc-99 dose occurring in the first 4,000 years. This dose comes from the sand pads inventories from Tanks 14, 15, and 16, of which 99 % of the Tc-99 is released within the first 4,000 years.

The peak Tc-99 dose from the Tank 13 sand pads is approximately 0.26 mrem/yr; whereas the peak Tc-99 dose from the Tank 16 sand pads is one order of magnitude smaller (0.024 mrem/yr), despite a much earlier release and a Tank 16 Tc-99 sand pads inventory that is approximately one order of magnitude higher than in Tank 13. Therefore, the holdup of Tc-99 that has migrated into the contamination zone is conservative.

Also note that Cs-137 and Sr-90 are included in the legend for Figure RAI-NF-13.1; however, these radionuclides have no dose significance as they decay before they are able to contribute to the peak dose.

Figure RAI-NF-13.2 shows the radionuclide contributions to the MOP dose, regardless of sector, from only the sand pads inventories, using pessimistic modeling assumptions. These assumptions are defined in the response to RAI-NF-8 (for the model identified as “Flow Run 65, No Holdup”). This pessimistic model releases the majority of the contaminants at the start of the simulation and includes assumptions that accelerate flow.

Figure RAI-NF-13.2: MOP Radionuclide Dose Contributions (from the Sand Pads) Using Pessimistic Modeling Assumptions



In this pessimistic modeling case, more than 99 % of the Tc-99 inventories associated with the sand pads are released within the first 300 years. Tank 16 contributes a higher Tc-99 peak dose (approximately 8.2 mrem/yr) than Tank 13 (approximately 0.42 mrem/yr), which is consistent with the higher initial inventory.

Again, note that Cs-137 and Sr-90 are included in the legend for Figure RAI-NF-13.2. A small amount of Sr-90 is visible in the figure (0.0014 mrem/yr at about 500 years after final facility closure). The Cs-137 dose never exceeds 3E-07 mrem/yr. Even under these pessimistic modeling conditions, Cs-137 and Sr-90 significantly decay before these radionuclides can contribute to the MOP peak dose. A review of the inadvertent intruder results confirmed that Cs-137 and Sr-90 also had no significant contribution to the inadvertent intruder peak-dose results. Therefore, Cs-137 and Sr-90 from the sand pads inventories have no dose significance.

Future PA Revisions and the Role of SAs

As required by DOE Manual 435.1-1, maintenance of the HTF PA will include future updates to incorporate new information, update model codes, consider actual residual inventories, etc., as appropriate. Section 8.2 of the HTF PA states that as “additional data becomes available …additional modeling may be required.” Each time additional modeling is performed, DOE will evaluate whether or not an update to the Base Case (Case A) or a revision of the PA is needed. PA maintenance and the potential impact of new information is evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1. New fate and transport modeling will be performed, as required, through the SA process, replacing PA assigned inventories with the final residual characterization data. If other new information has been identified (e.g., updated K_d values developed through research or experimentation), this new data would also be evaluated through inclusion in the SA. The results of the waste tank-specific SA are then evaluated to determine if new information impacts PA-based conclusions. While it typically would not be necessary to replicate all of sensitivity and uncertainty analyses from the PA as part of a SA, it may be appropriate to include additional analyses to evaluate the new or unique waste tank-specific information.

As waste tank-specific SAs are prepared it is anticipated that sensitivity analyses will be performed to address both individual waste tank inventories and unique waste tank conditions explicitly, such as potential preferential pathways (e.g., degraded liners or in-leakage pathways). Because different as-modeled waste tank conditions may prove to be conservative or non-conservative (with respect to timing and/or magnitude of peak doses) depending on other corresponding waste tank conditions, these analyses are more suited to waste tank-specific sensitivity analyses than to changes to the overarching Base Case assumptions. For example, modeling assumptions that cause the annulus inventories to be released relatively slowly (thereby adding to releases from the in-tank inventories) versus relatively rapidly would impact peak doses differently depending on the quantity (total curies) and nature of the inventory remaining in a waste tank annulus (e.g., short-lived versus long-lived radionuclides). In a similar manner, preferential pathways such as failed liners can have very radionuclide-specific dose impacts, with earlier liner failures not necessarily corresponding to greater peak doses, as demonstrated in HTF PA Section 5.6.7.6 (Liner Failure Times Analysis using the PORFLOW Deterministic Model). Additionally, if the closure inventory for a specific radionuclide is determined to be significantly different than previously projected, the impact of the modeling simplification discussed in the RAI would need to be further evaluated.

RAI-NF-14

Although several HTF sources are expected to be located below the water table, tank releases are modeled under unsaturated conditions in the Performance Assessment. DOE should provide additional support for modeling HTF source releases under unsaturated conditions.

Basis

DOE simulates releases from HTF sources that are expected to be located below the water table through use of unsaturated, near-field flow and contaminant transport models. Radionuclide fluxes extracted from the unsaturated zone models are then used to load radioactivity into the HTF/PORFLOW® model that is used to simulate saturated zone transport at HTF.

For submerged and partially submerged tanks (i.e., Type I and II tanks), no vadose zone is expected to be present but inclusion of a vadose zone in the near-field model domain may increase travel times to a potential receptor if the contaminant flux is calculated based on flux out of the near-field model domain (versus flux out of the tank/vault system). Predicted doses may also be sensitive to the manner in which contaminant flux is loaded in the saturated zone model (e.g., number of source cells or source location). Finally, release rates from HTF sources may be higher in the saturated zone compared to the vadose zone in certain cases. Therefore, DOE should provide additional clarification or support for model simplifications to provide assurance that doses are not significantly underestimated in the HTF Performance Assessment.

Path Forward

DOE should clarify if the near-field model fluxes are calculated at the bottom of the HTF tank basemats or at the bottom of the near-field model domain.

DOE should clarify the location of source loading (elevation of source release relative to the water table) and the number of source cells used to represent the source. DOE should provide an estimate of the range in potential dose based on dilution or concentration of the contaminant flux given source loading selections.

DOE should evaluate the impact of simulation of HTF source releases in an unsaturated zone model for submerged and partially submerged tanks. DOE should consider all relevant flow/transport regimes in evaluating whether radionuclide release rates could be potentially underestimated. For example, DOE should consider cases where flow rates through the engineered system may be low and releases limited by diffusion. For example, relatively high flow around the tank/vaults in the saturated zone at early times could lead to higher release rates, if flow rates in the saturated zone maintain a higher concentration gradient. DOE should also consider cases where flow occurs predominately through preferential pathways through the tank/vaults in alternative configurations (i.e., could releases be underestimated at early times in the near-field model if flow were to occur primarily through preferential pathways in the saturated zone prior to significant cementitious material degradation). Finally, DOE should consider the period of time when flow through the grout monolith increases significantly and releases are dominated by advection through the cementitious materials. Details regarding hydraulic head gradients and the magnitude of flow through the engineered system in a saturated system should be provided. Note that additional detail from Portage modeling (PORTAGE-08-022) may be helpful in responding to a portion of this request for additional information.

Response RAI-NF-14

The HTF PA near-field models encompass both unsaturated and saturated conditions depending on the structure being simulated. For near-field model domains extending down to the water table, the aquifer modeling cells are set to the solute flux leaving the model domain (crossing the water table). For near-field model domains that extend into the water table, the aquifer modeling cells are defined to be the solute flux leaving the engineered system (entering surrounding soil) rather than the model domain. Near-field sources that are submerged fully or partially are treated in this manner. That is, near-field models all of the Type I and Type II tanks, and the transfer lines and ancillary structures within the vicinity of the Type I and Type II tanks include a portion of the saturated zone, but the flux leaving the waste tank concrete vault is modeled as aquifer modeling cells.

Aquifer modeling cells for waste tanks are defined in the horizontal plane as those computational cells with centers residing within the physical footprint of the waste tanks. For smaller sources, the nearest cell center is selected. The number of source cells varies between four and six for waste tanks. Vertically, the aquifer modeling cells are placed at the lower of the highest fully saturated cell and the cell coinciding with the basemat elevation. Figures RAI-NF-14.1 and RAI-NF-14.2 show the horizontal and vertical positions, respectively; of the HTF source nodes (note that the vertical height is exaggerated). Among waste tanks, only sources for the fully submerged waste tanks are significantly below the water table. Computational cells at the water table beneath H Area are roughly 20 feet deep and become thinner with depth (e.g., on the order of 7 feet deep in the UTRA-LZ). DOE has not investigated the specific impact of grid resolution in the source zone on solute transport, but has studied general numerical dispersion/dilution within the HTF modeling sectors (see the *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, RAI-FF-3).

Although the near-field models consider saturated conditions for fully and partially submerged waste tanks, the 2-D grids can only accommodate axisymmetric flow past waste tanks. Any lateral flow in the saturated zone (crossflow) is not explicitly represented in the near-field models, and would require a 3-D near-field model. DOE earlier developed a 3-D Cartesian near-field waste tank model but observed significant drawbacks with using the model to investigate 3-D flow effects. [PORTAGE-08-022] First, the Cartesian grid, although refined within the waste tank, could not accurately resolve thin features of a cylindrical waste tank (e.g., steel liner), limiting its usefulness to failed-liner scenarios. Second, large computational times required to complete simulations significantly constrained the number of baseline and sensitivity cases that could be considered. DOE considered the trade-offs between 2-D and 3-D near-field models and concluded that various effects of aquifer crossflow could be neglected. [SRNL-L6200-2010-00026]

The possibility of a “dead-zone” of contamination build-up under the basemat that would hinder diffusive release from a waste tank was acknowledged in the HTF modeling. This effect was neglected in general because diffusional release from the waste tank contamination zone is not significant compared to advection. For waste tanks assumed to have a failed liner at time zero, diffusion-dominated release may occur for a short time. Several factors tend to offset this potential non-conservatism:

- Advection in soil surrounding a waste tank, relatively high diffusion coefficients for soil, and the proximity of the lower boundary limit contamination build-up beneath the basemat;
- The modeling assumption of early liner failure encompassing the entire surface area significantly over-predicts diffusive release; and
- Early releases that are biased low tend to reduce the peak dose (that occurs at a later time), as demonstrated in the liner sensitivity analysis provided in HTF PA Section 5.6.7.6 (Liner Failure Times Analysis using the PORFLOW Deterministic Model).

As discussed in HTF PA Section 4.4.4.1.2 (General Vadose Zone Waste Tank Modeling in PORFLOW), the effect of aquifer crossflow on advective release from the waste layer following liner failure was quantitatively investigated. Contaminant release was found to be relatively insensitive to crossflow. For the range of crossflow rates expected beneath the HTF, the effect was considered small compared to other biases and uncertainties in the overall analysis.

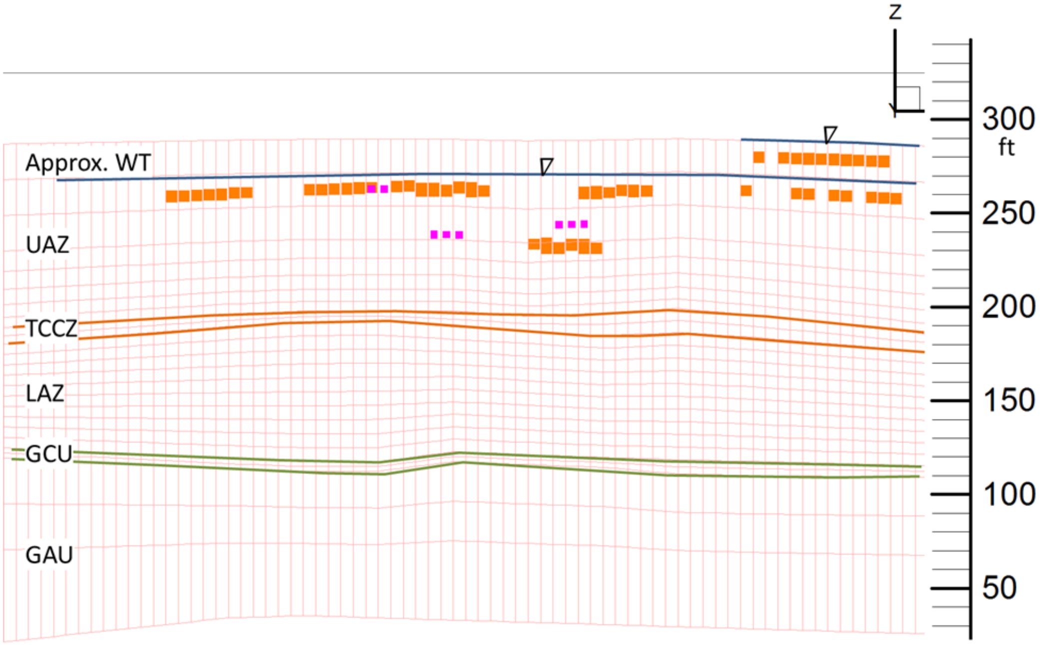
As required by DOE Manual 435.1-1, maintenance of the HTF PA will include future updates to incorporate new information, update model codes, consider actual residual inventories, etc., as appropriate. Section 8.2 of the HTF PA states that as “additional data becomes available …additional modeling may be required.” Each time additional modeling is performed, DOE will evaluate whether or not an update to the Base Case (Case A) or a revision of the PA is needed. Maintenance of the PA and the potential impact of new information is evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1. New fate and transport modeling will be performed, as required, through the SA process, replacing PA assigned inventories with the final residual characterization data. If other new information has been identified (e.g., updated K_d values developed through research or experimentation), it would also be evaluated through inclusion in the SA. The waste tank-specific SA results are then evaluated to determine if new information impacts PA-based conclusions. While it typically would not be necessary to replicate all of sensitivity and uncertainty analyses from the PA as part of a SA, it may be appropriate to include additional analyses to evaluate the new or unique waste tank-specific information.

As waste tank-specific SAs are prepared it is anticipated that sensitivity analyses will be performed to address explicitly both individual waste tank inventories and unique waste tank conditions, such as potential preferential pathways (e.g., degraded liners or in-leakage pathways). Because different as-modeled waste tank conditions may prove to be conservative or non-conservative (with respect to timing and/or magnitude of peak doses) depending on other corresponding waste tank conditions, these analyses are more suited to waste tank-specific sensitivity analyses than to changes to the overarching Base Case assumptions. For example, modeling assumptions that cause the annulus inventories to be released relatively slowly (thereby adding to releases from the in-tank inventories) versus relatively rapidly would impact peak doses differently depending on the quantity (total curies) and nature of the inventory remaining in a waste tank annulus (e.g., short-lived versus long-lived radionuclides). In a similar manner, preferential pathways such as failed liners can have very radionuclide-specific dose impacts, with earlier liner failures not necessarily corresponding to greater peak doses, as demonstrated in HTF PA Section 5.6.7.6 (Liner Failure Times Analysis using the PORFLOW Deterministic Model). Additionally, if the closure inventory for a specific radionuclide is determined to be significantly different than previously projected, the impact of the modeling simplification discussed in the RAI would need to be further evaluated.

Figure RAI-NF-14.1: Horizontal Location of Source Cells for Waste Tanks and Ancillary Structures

Note: Modeled source locations are shown as orange squares for waste tanks and smaller, magenta squares for ancillary structures.

Figure RAI-NF-14.2: Vertical Location of Source Cells for Waste Tanks and Ancillary Structures



Note: Modeled source locations are shown as orange squares for waste tanks and smaller, magenta squares for ancillary structures. Vertical height is exaggerated. Triangles denote the approximate ground level and water table (i.e., Approx. WT) relative to mean sea level.

CC-NF-1

Provide documentation of groundwater in-leakage into the submerged and partially submerged tanks.

Response CC-NF-1

No specific documentation identifying waste tanks with groundwater in-leakage has been created, however, a review of waste tank-specific history reports and anecdotal evidence suggests that groundwater has intruded into the annulus of Tanks 9, 10, 11, and 12. These waste tanks are Type I tanks, the first four waste tanks constructed in HTF. All four of these waste tanks are currently submerged in the water table. In the case of Tank 9, past waste tank annulus visual inspections have indicated the presence of water in the annulus, although liquid has not been seen in the annulus during recent inspections. No visual evidence indicating a potential location of groundwater in-leakage in to the annulus has been identified for Tank 9 so it is not clear if this was groundwater intrusion through the concrete vault wall or surface rain water that ponded on the surface above the waste tank and made its way into the annulus around the riser openings. With respect to Tanks 10, 11, and 12, there has been recent events where groundwater has intruded into the annulus of each of these three waste tanks. In the case of Tank 11, there is visual evidence of the water entering the annulus at the top of the annulus liner.

Identifying groundwater intrusion is complicated by the operation of the annulus ventilation systems. In the recent events of Tanks 10, 11, and 12, the ventilation systems were not operating in normal mode. Such normal operation results in introducing heated air to the annulus that in turn heats the metal surfaces (i.e., liner) of the annulus. Operation of the ventilation system has the demonstrated effect of evaporating the water that enters the annulus before any significant quantity collects. For example, earlier in 2013, the Tank 11 ventilation system became inoperable. In time, over 20 inches of groundwater collected in the annulus. Once the majority of this liquid was removed from the Tank 11 annulus and the ventilation was restored, the annulus was once again dry.

In addition, waste tank-specific history reports indicate the possible intrusion of water into the annulus of Tanks 14 and 16 in the late 1950’s but this water intrusion may have been attributable to rainwater ponding on the top of the waste tanks and entering through risers, as discussed above for the Type I tanks. Tanks 14 and 16 are both Type II tanks that are partially submerged in the water table.

CC-NF-2

DOE should provide additional support for the assumption that the chemical transition from Oxidized Region II to Oxidized Region III is not risk significant. The assumed solubilities for the highly radioactive radionuclides in the HTF do not appear to be sensitive to the pH transition. However, future revisions to the HTF Performance Assessment and updated geochemical modeling may indicate that certain radionuclide solubilities are sensitivity to pH.

The normative mineralogy of the hydrated grout assumed for the Geochemist's Workbench modeling of grout degradation is based on a mass balance calculation using the chemical composition of unhydrated cement. DOE used select phases that were taken from published cement simulations (i.e., Höglund, 2001; Lothenbach and Winnefeld, 2006; Kulik, 2011) to represent the normative mineralogy in the grout degradation modeling. However, the minerals that DOE selected to represent the hydrated grout are inconsistent with the minerals from Höglund (2001) and Lothenbach and Winnefeld (2006). Because the mineralogy used in modeling grout degradation determines the pH evolution of grout pore water, which in turn affects the calculated solubility, using an incorrect mineralogy in the model could lead to non-conservative solubilities and releases of radionuclides from the contaminated zone.

Response CC-NF-2

It should be noted that the conclusion that the chemical transition from Oxidized Region II to Oxidized Region III was not risk significant was not an inherent assumption. While it is true that there are numerous underlying assumptions made that can affect the solubility difference between Oxidized Regions II and III, the fact that the final derived Oxidized Region II and Oxidized Region III solubilities were often very similar was the result of calculations using widely accepted thermodynamic data, and was not an *a priori* assumption.

Normative mineralogy of grout assumed in most modeling efforts, including the HTF PA, was calculated from the chemical composition of the cementitious materials. The chemical composition of the cementitious materials for HTF reducing grout was different from that used in *Project SAFE: Modeling of Long-term Concrete Degradation Processes in the Swedish SFR Repository,* SKB Report R-01-08 and *Thermodynamic Modeling of the Hydration of Portland Cement,* DOI: 10.106/J, studies. Table CC-NF-2.1 lists the composition of the cementitious material used in each modeling study.

Table CC-NF-2.1: Chemical Compositions of Cementitious Materials used in Modeling Studies

	Project SAFE Study	Thermodynamic Modeling Study	SRS Waste Tank Reducing Grout
CaO (wt %)	64.0	63.2	23.6
SiO_2	21.0	19.7	44.3
Al_2O_3	3.5	4.7	19.2
Fe_2O_3	4.6	2.7	5.0
MgO	0.7	1.9	4.6
K_2O	0.6	1.1	1.7
Na_2O	0.1	0.1	0.3
SO_3	2.2	3.4	1.2

The cementitious materials used in the Project SAFE study and the Thermodynamic Modeling study are simple Portland cements, whereas the composition of the reducing grout used in the waste tanks at SRS contains large fractions of fly ash and blast furnace slag. These components are much richer in silica, alumina, and magnesia than Portland cement and this is reflected in the chemical composition of the reducing grout used in the waste tanks at SRS.

The differences in chemical composition translate into differences in normative mineralogy. The mineralogy used by the *Project SAFE: Modeling of Long-term Concrete Degradation Processes in the Swedish SFR Repository,* SKB Report R-01-08, study is dominated by the calcium silicate hydrate (CSH) phase with a high calcium / silicon ratio and Portlandite. In this study, sulfate was represented as a monosulfate phase and magnesium as brucite. Portlandite is not stable at complete hydration of SRS reducing grout because the abundant silica reacts with it to form CSH. The low calcium / silicon ratio in the chemical composition of the cementitious materials of SRS reducing grout means that the dominant CSH phase has a low calcium / silicon ratio relative to the CSH phase in the Project SAFE study. Initially, gypsum was used to represent sulfate in the SRS reducing grout, but starting mineralogy recalculated by the Geochemist's Workbench placed the sulfate in the more thermodynamically stable phase ettringite (see Table 3 from the *Evolution of Chemical Conditions and Estimated Solubility Controls on Radionuclides in the Residual Waste Layer During Post-Closure Aging of High-Level Waste Tanks,* SRNL-STI-2012-00404). The abundant alumina in SRS reducing grout makes hydrotalcite a more stable phase for magnesium than brucite. The difference in the initial mineralogy of these two modeling studies illustrates how chemical composition of cementitious materials affects normative mineralogy. Therefore, the initial normative mineralogy used to model degradation of SRS reducing grout is not in error, but rather is consistent with the chemical composition of the cementitious materials.

The initial normative mineralogy used in the *Thermodynamic Modeling of the Hydration of Portland Cement,* DOI: 10.106/J, study is not comparable to the initial normative mineralogy of the SRS waste tank reducing grout. The Thermodynamic Modeling study modeled hydration of Portland cement and their initial normative mineralogy is unhydrated. A basic assumption in the modeling of degradation of SRS waste tank reducing grout in the *Evolution of Chemical Conditions and Estimated Solubility Controls on Radionuclides in the Residual Waste Layer During Post-Closure Aging of High-Level Waste Tanks,* SRNL-STI-2012-00404, report is that the starting point of the model is a fully hydrated grout. Therefore, different initial normative mineralogy between the two studies is expected. Interestingly, the mineralogy at the end of the hydration period (10,000 hours) modeled in the Thermodynamic Modeling study is similar to the initial normative mineralogy used in the Evolution of Chemical Conditions report. They are both dominated by CSH, ettringite, and hydrotalcite. At the end-point for the model of the Thermodynamic Modeling study, the cement also contains Portlandite and a calcium monocarboaluminate phase. Again, Portlandite does not exist in the normative mineralogy of the SRS waste tank reducing grout because of the abundance of silica. Hence, the only difference is the presence of the calcium monocarboaluminate phase in the final mineralogy of the Thermodynamic Modeling study.

CC-NF-3

DOE should provide a basis that the use of Hanford sediments to develop the cement-leachate impacted K_ds (Table 4.2-25 of the HTF Performance Assessment) for HTF vadose zone soil is appropriate. Further, DOE should clarify why the Hanford derived cement leachate factor for plutonium (a factor of two) was not applied to derive the cement-leachate impacted from the non-impacted K_d in the HTF Performance Assessment. In its clarification, DOE should also more clearly describe how the factor of two was derived from PNNL-16663, which resulted in a factor of 0.25.

Response CC-NF-3

The cement-leachate impact factors were taken from the *Geochemical Data Package for Performance Assessment Calculations Related to the Savannah River Site,* SRNL-STI-2009-00473. As explained in this document, these factors represent literature values, which were used in the PA because site-specific values are not available. The Hanford Site cementitious radionuclide K_d value data set recorded in *Geochemical Processes Data Package for the Vadose Zone in the Single-Shell Tank Waste Management Areas at the Hanford Site,* PNNL-13421, was one of the few extensive studies identified in the literature, and as such, were used to determine impact factors on the Hanford Site sediment. These impact factors were then used (taking into consideration SRS soil conditions) to adjust the expected K_d values in a cement-leachate SRS sediment.

Regarding the question about the selection of the plutonium impact factor, all the plutonium impact factors are based on the Hanford Site PA database, which in turn are based on experimental data specific to the *Geochemical Processes Data Package for the Vadose Zone in the Single-Shell Tank Waste Management Areas at the Hanford Site,* PNNL-13421, data set. This report includes "best values" for plutonium sorption in groundwater (PNNL-13421, Page B.3, Table B1; for intermediate impacted sand) of 600 mL/g and for cementitious waste impacted K_d values of 150 mL/g (PNNL-13421, Page B.4, Table B1). The impact factor is the ratio of these two values, or 0.25. The experimental data to support this value of 150 mL/g was not noted in the Hanford Site data set. The reported Hanford Site cement-impacted K_d value was inconsistent with thermodynamic considerations of plutonium. Plutonium solubility has been measured to decrease sharply as the pH increases from 7 ($K_{solubility}$ = 1E-05 mol) to pH > 9 ($K_{solubility}$ = 1E-10 mol), as documented in *The Solubility of Actinides in a Cementitious Near-Field Environment*, 0956-053X(92)90051-J. To capture this geochemical phenomenon of reduced aqueous plutonium concentrations as the pH increases, it was elected in the *Geochemical Data Package for Performance Assessment Calculations Related to the Savannah River Site,* SRNL-STI-2009-00473, to depart from the best estimates used at Hanford Site and to rely on known thermodynamic trends. Furthermore, under SRS site-specific conditions, it has been recently shown in *Iodine, Neptunium, Plutonium, and Technetium Sorption to Saltstone and Cement Formulations under Oxidizing and Reducing Conditions*, SRNL-STI-2009-00636 (Table 9.1 and 9.2), that in cement leachate simulants at pH values between 11 and 12, that plutonium rapidly precipitates out of solution in the absence of any solids, resulting in solubility values in the order of 1E-09 to 1E-10 mol. The Hanford Site K_d data suggest that the plutonium aqueous concentrations increase at higher pH values, contrary to thermodynamic considerations and experimental results. [0956-053X(92)90051-J, SRNL-STI-2009-00636] To capture the pH-Pu_{aq} trend, DOE elected to employ a cement impact factor of two, to capture the expected trend of Pu_{aq} concentrations decreasing in the presence of the cementitious leachate compared to pH 5.5 groundwater. The factor of two was based on expert judgment considering the available data and the other impact factors.

CC-NF-4

DOE should clarify whether the piping that enters the tanks will be grouted.

Response CC-NF-4

As each waste tank is filled with grout, grout material will flow into the open end of transfer line piping that penetrates either the waste tank risers or walls, thereby sealing and effectively isolating the transfer line piping. Though the waste tank fill-grout will seal the transfer line piping at the waste tank penetrations, there are no current plans to specifically grout or fill the HTF transfer lines.

CC-NF-5

The analytic solution to the diffusion equation to compute the chloride concentration on page 32 in SRNL-STI-2010-00047 appears to be incorrect to use at long times. It appears to only be valid to use at short times, when the depth of the chloride penetration is small compared to the vault thickness. The analytical solution assumes a fixed concentration at $x=0$, zero concentration at $x=\infty$, and zero initial concentration. Instead, a correct solution for long times should consider zero flux at the concrete/liner interface, to keep all of the chloride within the vault thickness. Because of the incorrect use of the analytic equation, chloride concentrations at the concrete/steel interface in Figures 18 and 19 may be underestimated. Such figures are only provided to derive a notion of times for chloride to diffuse. These results do not appear relevant to time estimates in the stochastic methodology. Confirm that these results are not relevant to estimates of liner failure.

Response CC-NF-5

DOE acknowledges that the diffusion equation used may not be appropriate at long times, but the comment is correct that the results of the diffusion equation were not used in the stochastic methodology. The equations in Sections 3.3.2 and 6.2 of *Life Estimation of High Level Waste Tank Steel for H-Tank Farm Closure Performance Assessment,* SRNL-STI-2010-00047, were used to determine the time of corrosion initiation due to chlorides.

Hydrology and Far-Field Transport (FF)

RAI-FF-1

The HTF/PORFLOW® model may not be well calibrated. DOE should provide more detail regarding model calibration at HTF.

Basis

GSA/PORFLOW® (and GSA/FACT2) documentation suggests that the GSA model is not well calibrated local to HTF. For example, WSRC-TR-96-00399, Rev. 1, Volume 2 indicates that:

- there are unexpected high residuals east of HTF (Page 23);
- relatively larger residuals are found in and east of HTF (Page 24);
- 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
- additional work is needed to better define uncertainty in model predictions (Page 25).

Further, WSRC-TR-2004-00106, Rev. 0 indicates on page 23 that GSA/PORFLOW head residuals are generally relatively large compared to GSA/FACT and that the artificial recharge zone in the GSA/FACT model was more effective at reducing head residuals at HTF but was considered less realistic. Page 24 of WSRC-TR-2004-00106, Rev. 0 goes on to state that more extensive model calibration would improve the GSA/PORFLOW® model.

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 Performance Assessment. If the HTF/PORFLOW® model is not well-calibrated, the dose predictions may be over- or under-estimated depending on such factors as source location and radionuclide.

Path Forward

DOE should provide additional information regarding the goodness of fit of the model to calibration targets (e.g., water levels) local to the area of interest at HTF. This information should include residuals and calibration statistics for calibration targets available at the time of GSA/PORFLOW® modeling. More recent information could also be used to evaluate model agreement to measured values, if calibration targets used at the time of modeling are not thought to be representative of post-closure conditions (see RAI-FF-2). Environmental monitoring data could also be used to help validate the HTF/PORFLOW® model and demonstrate the sufficiency of the model in predicting contaminant fate and transport at HTF. For example, DOE could perform backwards particle tracking to identify the source of observed Gordon aquifer contamination. If corroborating source release information is available, validation exercises may provide additional support for the predictive capability of the HTF/PORFLOW® model.

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[2] 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.

Response RAI-FF-1

Use of HTF Environmental Monitoring Data to Validate the Model

A preliminary review of HTF environmental monitoring data was performed to identify any areas in which the data might be used to help validate the HTF PORFLOW modeling. An informal review of the HTF monitoring well sampling results contained within the ERDMS database was conducted to see if any trends or occurrences of interest were noted. The review concentrated on non-volatile beta sample events at wells close to the HTF Tank 16 (i.e., wells HTF-5 through HTF-8), based on the reasonable assumption that radionuclides of primary interest for this exercise (e.g., Tc-99) would show up in the non-volatile beta data at these wells. While this review did identify some non-volatile beta detection results of interest (e.g., high with respect to other results around the same time frame), no obvious trends with respect to location, timing, or magnitude of the events was apparent. While it is possible that a more in depth review of the ERDMS data might result in data tendencies that might be used to help validate the HTF PORFLOW modeling, this possibility was not apparent from the preliminary review.

The following complications muddle the use of HTF well sampling data in HTF PORFLOW Model validation:

1. Well sampling data for HTF before the late 1970s is not available in ERDMS.
2. The HTF well sampling locations with data available vary over time. For example, wells HTF-1 through HTF-17 were drilled in June through September of 1973, while wells HTF-18 through HTF-34 were not drilled until December 1984 through August 1985. In addition, many of the older wells (such as HTF-5 through HTF-9) that had supplied data of interest have been abandoned over the years for various reasons.
3. There are no precisely quantified releases of record against which sampling data could be associated since the few historical releases of note in HTF (e.g., the Tank 16 leak) have material release ranges (versus precise characterization) associated with them and no source-term quantities useful for contaminant tracking.
4. The factors with the most impact on infiltration rates (e.g., rainfall, waste tank asphalting) will have a corresponding impact on travel time, and these factors have varied greatly over time in degrees not easily quantified.

Calibration Target Study

In order to investigate “the goodness of fit of the GSA model to calibration targets (e.g., water levels) local to the area of interest at HTF,” a review of the wells utilized in the GSA model calibration was first performed. First, the original set of GSA model well calibration targets was replaced with a more recent, existing set (comprising data from 2002 through 2003) for GSA water table wells (see *An Updated Regional Water Table of the Savannah River Site and Related Coverages,* WSRC-TR-2003-00250). Second, calibration statistics were recomputed. Further study was then concentrated on 52 of the wells across the GSA that were identified to have a > 6-foot residual when compared to the GSA steady-state flow field heads. This threshold was adopted to reflect what is thought to be approximately double, what the water level fluctuation is in most SRS wells, from year to year, due to differences in local and regional precipitation and subsequent deep recharge to the aquifers. Additional data associated with these 52 wells were extracted from the ERDMS database to evaluate each well further as part of this study. Many of the SRS “water table” wells have additional measurements recorded since the 2002 to 2003 period when the Median or Adjusted Median was computed; however, some of the 52 wells have had no additional measurements since that time.

The findings of the present well study are documented in Table RAI-FF-1.1. This Table RAI-FF-1.1 contains a listing of the 52 wells evaluated and a status disposition for each well (each well is generally indicated as "Not Credible" or "Dependable water level estimate"). In most "Not Credible" cases, a reason for the “Not Credible” designation is provided in the comments. Overall, 21 of the 52 wells studied were designated as "Not Credible" while 31 were found to be wells with a dependable estimate of the water level.

Table RAI-FF-1.1: Summary List of High Residual Wells with Status

Well	Median WT	Model Head	Residual	Absolute Residual	Comments
FC4D	**151**	**178.05**	**27.05**	**27.05**	Not credible - screen or borehole likely penetrates into Gordon Aquifer.
HCA1 (a)	269.1	253.95	-15.15	15.15	Dependable water level estimate - No obvious reason to remove this well from target list, 37 measurements.
BGO12CX	**218.5**	**232.66**	**14.16**	**14.16**	Not credible - surrounded by wells whose water levels are 11-13 ft higher, screen zone is situated ~ 10 ft lower in aquifer unit.
HCA2 (a)	269.8	256.3	-13.5	13.5	Dependable water level estimate.
HTF30	**270**	**283.37**	**13.37**	**13.37**	Not credible - limited record, only 3 measurements, last sampled in 1986, surrounded by wells that conform more closely to model.
HAA3D (b)	263.9	250.93	-12.97	12.97	Dependable water level estimate.
HAA5D	**275.3**	**288.22**	**12.92**	**12.92**	Not credible - There are discrepancies in ERDMS database, suggesting the screen zones and water levels of HAA-5D and -5C are mixed.
HCA4 (a)	269	256.14	-12.86	12.86	Dependable water level estimate.
BRR6D	**207.3**	**219.9**	**12.6**	**12.6**	Not credible - ambiguous information in database. Unexplained 11-13 foot drop in water levels after the first 4 measurements.
FBP46D	**166.2**	**178.67**	**12.47**	**12.47**	Not credible - only 2 measurements, possible borehole penetration into Gordon Aquifer
HSL4D (b)	260.9	249.55	-11.35	11.35	Dependable water level estimate
NWP1D	**210.9**	**222.16**	**11.26**	**11.26**	Not credible - anomalous readings. Nearby well has more reasonable water level elevation.
HAA15D (a)	269.8	258.68	-11.12	11.12	Dependable water level estimate
HET1D (b)	268.8	257.77	-11.03	11.03	Dependable water level estimate
HAA2D (b)	276.4	265.73	-10.67	10.67	Dependable water level estimate
FBP44D	**167.5**	**178.15**	**10.65**	**10.65**	Not credible - only 2 measurements, possible borehole into Gordon Aquifer.
HET4D (b)	259.6	249.5	-10.1	10.1	Dependable water level estimate
HET2D (b)	258.7	248.75	-9.95	9.95	Dependable water level estimate
ZW7 (b)	266.7	256.8	-9.9	9.9	Dependable water level estimate
FAL2	**217.3**	**227.16**	**9.86**	**9.86**	Not credible - very close to another well that has much more credible water levels. Possible preferential recharge paths might be influencing well levels.
FCA19D	**217.1**	**226.92**	**9.82**	**9.82**	Not credible - anomalous readings. Nearby well has more reasonable water level elevation.
HHP1D (b)	271.3	261.48	-9.82	9.82	Dependable water level estimate
FNB4	213.9	204.14	-9.76	9.76	Probably dependable - but no new measurements since 1999, after which water levels declined. Other wells nearby have lower water levels since then. Modeled water levels are generally a little low in this area.
HET3D (b)	259.1	249.36	-9.74	9.74	Dependable water level estimate
HSL5D (b)	266.3	256.59	-9.71	9.71	Dependable water level estimate
HC8C	**198**	**207.47**	**9.47**	**9.47**	Not credible - only 1 measurement (in 1988) other suitable calibration targets nearby.

Table RAI-FF-1.1: Summary List of High Residual Wells with Status (Continued)

Well	Median WT	Model Head	Residual	Absolute Residual	Comments
MGC32	**245.2**	**235.95**	**-9.25**	**9.25**	Not credible - anomalous water levels. Surrounded by wells that all have lower median water levels.
BGX8DR	204.9	213.95	9.05	9.05	Dependable water level estimate. New measurements, now 99 total, new median = 203.2 ft; new residual = 10.75 ft, model may be too high here.
FAL1	**218.4**	**227.2**	**8.8**	**8.8**	Not credible - very close to another well that has much more credible water levels. Possible preferential recharge paths might be influencing well levels.
H19	**228.5**	**219.72**	**-8.78**	**8.78**	Not credible - surrounded by wells with more credible water levels. Screen zone reported to be 1.5 ft. in length.
HSB141D	237.9	229.4	-8.5	8.5	Dependable water level estimate
FBP47D	**170.4**	**178.78**	**8.38**	**8.38**	Not credible - only 2 measurements, possible borehole into Gordon Aquifer.
BG29	**244**	**235.87**	**-8.13**	**8.13**	Not credible
HAA8D	266.5	258.9	-7.6	7.6	Dependable water level estimate
NWP3D	225	232.52	7.52	7.52	Dependable water level estimate - conforms to neighbors BGO-13DR and -14DR closely.
HR811	246.8	239.38	-7.42	7.42	Dependable water level estimate
HHP2D	274.5	267.11	-7.39	7.39	Dependable water level estimate
NEP4D	191.6	198.9	7.3	7.3	Dependable water level estimate - about 10 ft higher than nearby flowing stream. Model needs to honor flowing stream elevations as = to the water table.
FSL4D	216.8	224.01	7.21	7.21	Dependable water level estimate - new median water table for full record is 214.7 ft, so new residual is ~ 9.3 ft.
BGX3D	214.5	221.49	6.99	6.99	Dependable water level estimate - many new measurements, general drop in water levels since 2003; new full record median = 212.4 ft.
BG115	**215.8**	**222.76**	**6.96**	**6.96**	Not credible; only 1 measurements in 1988; other suitable calibration targets nearby.
Z1	**218.7**	**225.59**	**6.89**	**6.89**	Not credible; only 1 measurements in 1986; other suitable calibration targets nearby.
NWP2D	200.2	207.02	6.82	6.82	Dependable water level estimate. very similar to HMD-2D and HMD-1D during the period of record overlap.
BGX5D	208.2	215	6.8	6.8	Dependable water level estimate. Now 115 measurements. Median water table = 202.6 ft, new residual = 8.8 ft, no obvious reason to eliminate.
ZW2	**207.5**	**200.77**	**-6.73**	**6.73**	Not credible. No new measurements since 1995. Nearby well FNB-5 has same screen interval, longer record of measurements - complete overlap. Its residual is -1.2 ft.
FAC2	**235.9**	**229.31**	**-6.59**	**6.59**	Not credible -- older construction, screen too high, may partially reflect perching. More reliable sources close-by.
NBG3	217.5	223.99	6.49	6.49	Dependable WL estimate.
FNB3	208.7	202.31	-6.39	6.39	Dependable water level estimate. New median computed - 208.0 ft, new Residual = -5.69 ft; < 6 ft residual now.
FAC6	220.7	227.01	6.31	6.31	Dependable water level estimate. Even though slightly out of acceptable range, several other similar wells near, with median water levels less than the 6-ft threshold.
BG30	**241.1**	**234.86**	**-6.24**	**6.24**	Not credible.
BG94	191.9	185.69	-6.21	6.21	Dependable water level estimate. Water levels have dropped since the last measurement in this well and probably the residual is < 6 ft, now.
NWP101D	225.5	231.56	6.06	6.06	Dependable water level estimate. Conforms to neighbors BGO-13DR and -14DR closely.

Wells shown in **Bold** were designated “Not Credible”
a: Wells in zone near H-Canyon in H Area where model seems to be low in computing water table elevations
b: Wells southwest of HTF where model seems to be low in computing water table elevations

Two relatively small areas were identified over which the GSA model could be improved to make the computed water levels conform somewhat better to well measurements. Both areas are near HTF and they both have the model predicting water level a little low. The wells associated with these two areas are noted in Table RAI-FF-1.1. Specifically, the two areas where the GSA model seems to be low in computing heads (residuals > 6 foot) are a zone just southwest of HTF and a zone near H-Canyon. An image of each area (shaded blue in the figures) is provided in Figures RAI-FF-1.1 and RAI-FF-1.2. Model refinements in these areas would produce modest perturbations to hydraulic gradients and groundwater velocities. Based on the location of these two relatively small areas, any potential model improvements would not be expected to significantly impact near field (i.e., 100 meter) releases from the HTF. The risk significance of groundwater velocity variability is investigated in the response to RAI-FF-3 through two aquifer flow field sensitivity cases.

In general, the GSA model "goodness-of-fit" improved with consideration of the updated and screened calibration targets from 2002 to 2003. Additional information regarding the Goodness of Fit of the Model to Calibration Targets is provided in the response to RAI-FF-2, including calibration statistics local to H Area.

Figure RAI-FF-1.1: Low Zone Outside H Area Southwest of HTF (Wells with Residuals > 6 foot)

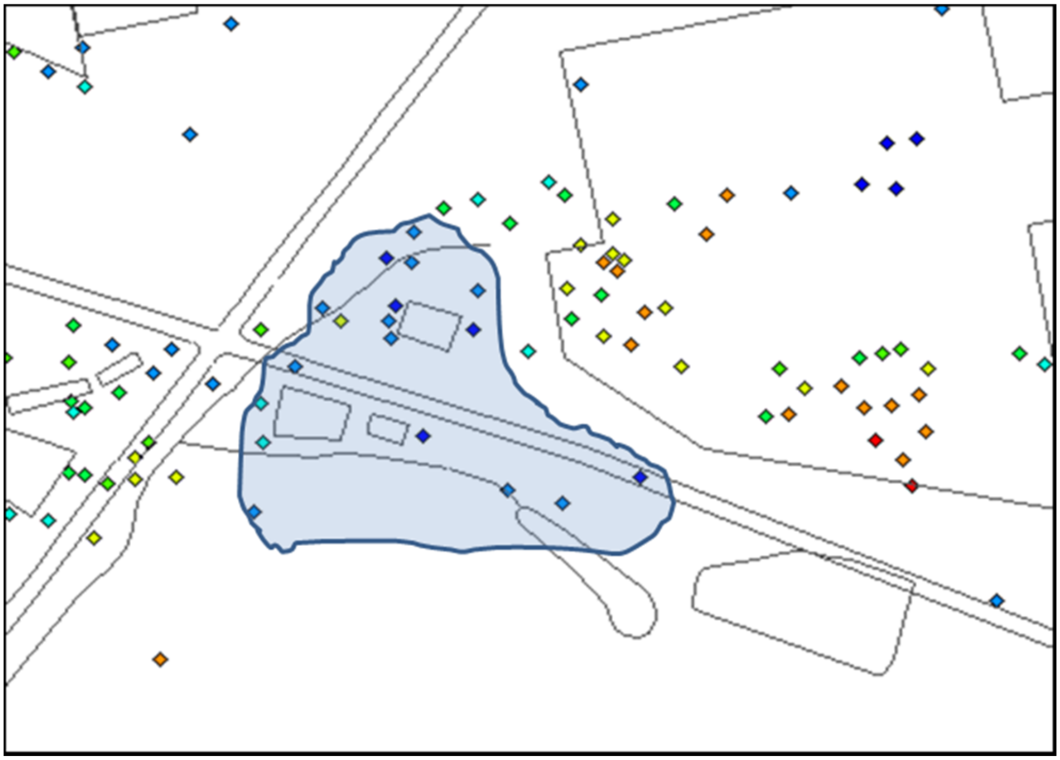
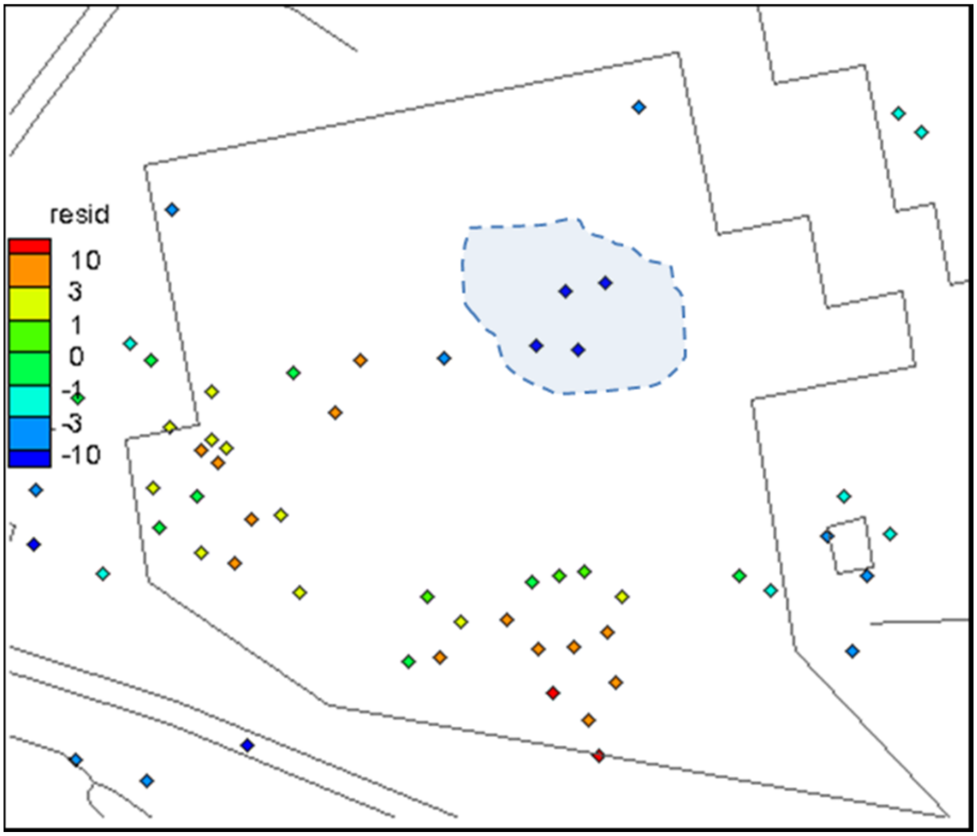


Figure RAI-FF-1.2: Low Zone within H Area near H-Canyon (Wells with Residuals > 6 foot)



RAI-FF-2

HTF calibration targets developed during GSA model development may not represent post-closure conditions. DOE should evaluate the representativeness of HTF calibration targets to long-term conditions.

Basis

The GSA/FACT and GSA/PORFLOW® models, upon which the HTF/PORFLOW® model is based, 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 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 time interval over which water level measurements were averaged. If the HTF/PORFLOW® model is not well calibrated to calibration targets representative of post-closure conditions, it is unclear if the HTF/PORFLOW® model is adequate for the purposes of simulating post-closure contaminant flow and transport at HTF.

Path Forward

GSA/FACT model documentation lists a number of potential sources local to HTF. For example, WSRC-TR-96-00399, Rev. 1, Volume 2 (page 21) lists a number of water leaks or potential sources to the model and indicates that undoubtedly unknown leaks exist at HTF. DOE should evaluate the potential for GSA/PORFLOW® calibration targets to have been influenced by potential sources and sinks, including the sources listed in the GSA/FACT model documentation.

Since the GSA/FACT and GSA/PORFLOW® models were developed, additional information has been collected at HTF that could also be used to evaluate the representativeness of the calibration targets. DOE could perform the following types of activities related to consideration of new information:

- Develop new calibration targets based on a longer or more representative period of record.
- Develop uncertainty ranges for calibration targets.
- Evaluate the goodness of fit of the HTF/PORFLOW® model to new calibration targets.
- If necessary, recalibrate the GSA/PORFLOW® model[3].

Finally, DOE could provide arguments as to why the HTF/PORLOW® model is adequate for the purposes of making long-term dose predictions for the HTF Performance Assessment (e.g., sufficient accuracy or biased towards higher dose predictions).

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[3] Model recalibration is a long-term effort that is not expected to be accomplished during the RAI resolution period.

Response RAI-FF-2

The HTF PA presents in Table 4.2-17 the GSA/PORFLOW calibration statistics for well water level data through 1995, which was used to develop the original GSA model, and updated monitoring data through 2006. These data comparisons are reproduced here as Tables RAI-FF-2.1 and RAI-FF-2.2, respectively. Both data sets indicate the presence of double-digit residuals as evidenced by the minimum and maximum values. These large residuals could be a concern to the extent that the underlying well targets are representative of long-term average conditions and had an influence on model calibration. To investigate the former, well water levels were reevaluated for reliability as model calibration targets, focusing on shallow wells used to define the water table in 2003 as discussed in the Calibration Target Study section of RAI-FF-1.

Table RAI-FF-2.1: Summary Statistics for GSA/PORFLOW Hydraulic Head Residuals for Well Targets (through 1995)

Aquifer Zone	Number of Wells	Residual Median	Residual Average	Residual RMS	Residual min	Residual max
Gordon	79	-0.0	-0.5	+1.7	-4.7	+2.5
lower UTR	173	+0.8	+0.6	+4.6	-9.4	+27.0
upper UTR	386	-0.1	-0.5	+3.4	-15.2	+10.0

RMS = Root-Mean-Square

Table RAI-FF-2.2: Summary Statistics for GSA/PORFLOW Hydraulic Head Residuals for Well Targets (updates through 2006)

Aquifer	Number of Wells	Residual Median	Residual Average	Residual RMS	Residual min	Residual max
Gordon	94	+0.3	-0.0	+1.5	-3.8	+2.6
UTRA-LZ	272	+1.1	+1.0	+4.7	-11.9	+27.0
UTRA-UZ	551	+0.8	+0.1	+3.5	-16.8	+14.5

RMS = Root-Mean-Square

As discussed in the response to RAI-FF-1, a review of ERDMS well data across the GSA was performed as a first step toward investigating the “goodness of fit” of the GSA model to calibration targets (e.g., water levels) local to the area of interest at HTF. The results of this study are documented in Table RAI-FF-1.1 of the response to RAI-FF-1. Out of wells available for calibration (data was not available for all of the wells within the model domain), 21 wells were determined or suspected to be unreliable model calibration targets, as discussed in the response to RAI-FF-1. Table RAI-FF-2.3 presents summary statistics for hydraulic head omitting the 21 unreliable wells. The minimum and maximum residuals for the 2003 well targets documented in *An Updated Regional Water Table of the Savannah River Site and Related Coverages,* WSRC-TR-2003-00250, are observed in Table RAI-FF-2.3 to be much improved over Table RAI-FF-2.2. Figure RAI-FF-2.1 shows the spatial distribution of residuals corresponding to Table RAI-FF-2.3. Focusing on the HTF and surroundings, Figure RAI-FF-2.2 shows residuals within a 3,500-foot radius of H Area, and Table RAI-FF-2.4 summarizes statistics for this subgroup using the 2003 well targets.

The GSA model reasonably fits the underlying well data and the fit with available well data is improved when unreliable well data is removed. DOE believes the GSA model is adequate for the purposes of fate and transport modeling in the HTF PA, considering the risk significance of hydraulic conductivity and flow field variability, as discussed in the response to RAI-FF-3.

Table RAI-FF-2.3: Summary Statistics for GSA/PORFLOW Hydraulic Head Residuals for 2003 Well Targets

Aquifer Zone	Number of Wells	Residual Median	Residual Average	Residual RMS	Residual min	Residual max
Gordon	-	-	-	-	-	-
UTRA-LZ	52	+2.7	+1.6	+4.3	-9.8	+9.0
UTRA-UZ	406	+0.4	-0.0	+3.4	-15.2	+7.5

RMS = Root-Mean-Square

Table RAI-FF-2.4: Summary Statistics for GSA/PORFLOW Hydraulic Head Residuals for 2003 Well Targets near H Area

Aquifer Zone	Number of Wells	Residual Median	Residual Average	Residual RMS	Residual min	Residual max
Gordon	-	-	-	-	-	-
UTRA-LZ	-	-	-	-	-	-
UTRA-UZ	85	-0.8	-2.0	+5.4	-15.2	+5.9

RMS = Root-Mean-Square

Figure RAI-FF-2.1: GSA/PORFLOW Hydraulic Head Residuals for 2003 Well Targets

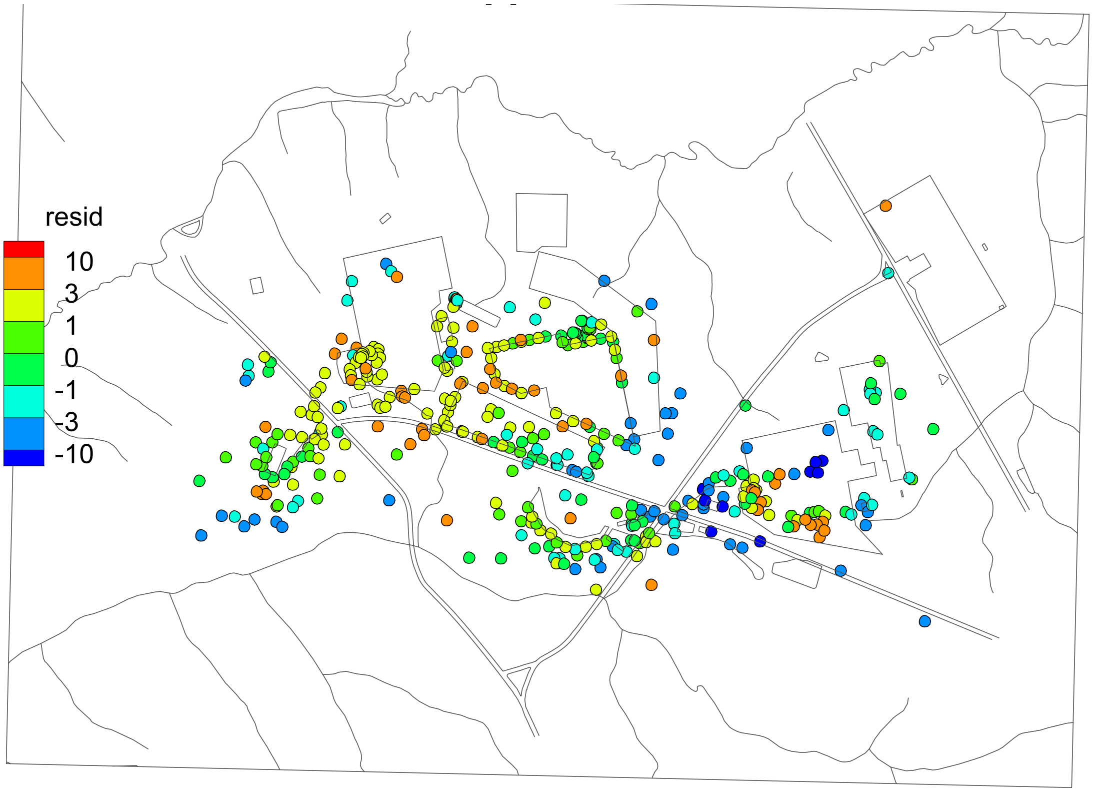
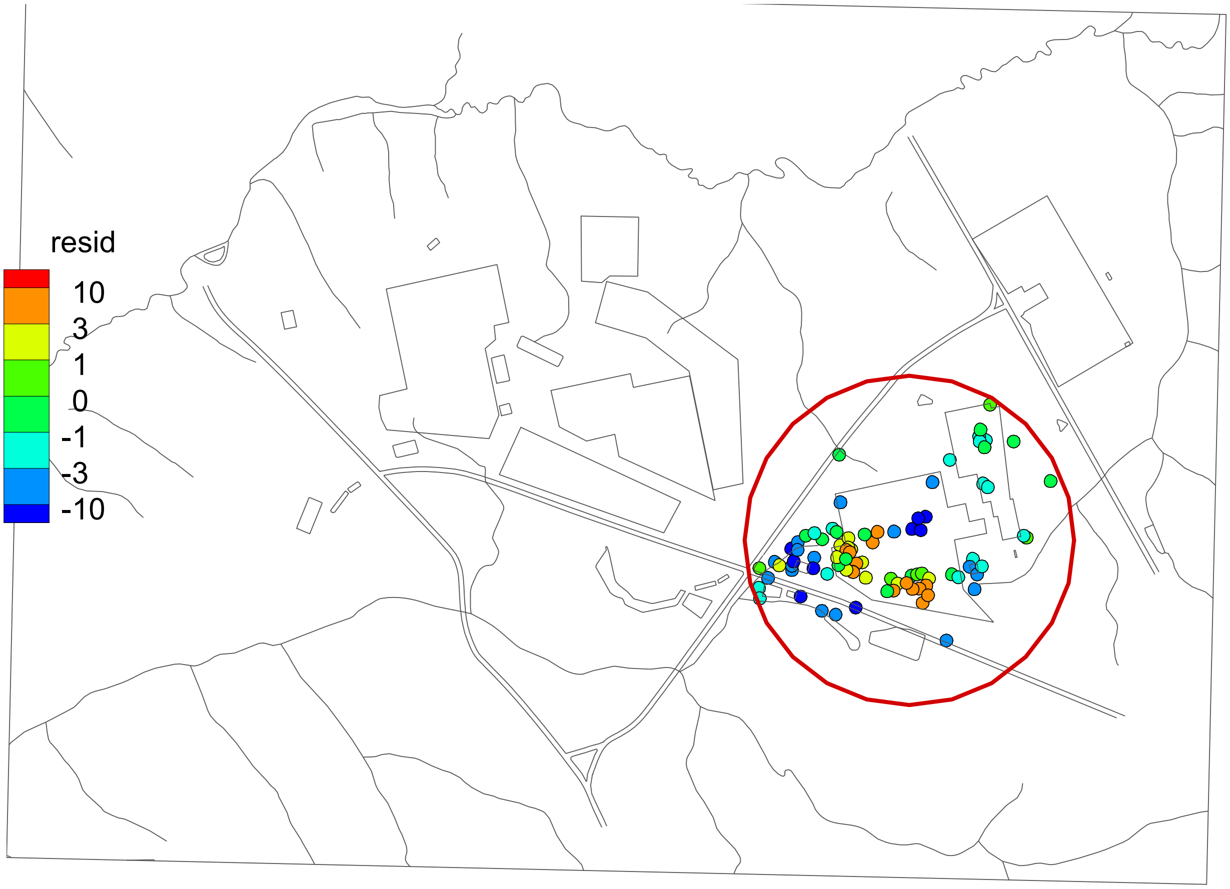


Figure RAI-FF-2.2: GSA/PORFLOW Hydraulic Head Residuals for 2003 Well Targets near H Area



RAI-FF-3

A strong physical basis for adjustments to the Upper Three Runs Aquifer hydraulic conductivity at HTF during GSA/PORFLOW® model calibration was not provided. DOE should provide additional support for hydraulic conductivity assignments at HTF.

Basis

Adjustments to hydraulic conductivity during the GSA/PORFLOW® model calibration process may not be adequately supported and could lead to significant impacts to the flow field at HTF. Changes in hydraulic conductivity could lead to increased or decreased dilution factors and travel times. Dose predictions could be under- or over-estimated depending on such factors as source location and radionuclide.

GSA/PORFLOW® model documentation (WSRC-TR-2004-00106, Rev. 0) indicates that during model recalibration, hydraulic conductivity was lowered and artificial recharge sources[4] omitted at HTF. The documentation indicates that a low permeability confining zone or generally lower hydraulic conductivity was thought to exist at HTF but supporting details were lacking. More recently, DOE indicated that there may be evidence of low permeability zones and perched water at HTF but upon further investigation stated that there appears to be a lack of corroborating evidence for perched zones at HTF (ML13126A127; ML13154A327).

Path Forward

DOE should perform the following activities to clarify and provide additional support for the hydraulic conductivity assignments at HTF:

- Clarify the horizontal and vertical extent of hydraulic conductivity adjustments at HTF.
- Provide additional support for the hydraulic conductivities assumed for HTF.
- Evaluate the impact of hydraulic conductivity adjustments on key radionuclide concentrations and dose at the compliance boundaries.

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[4] Artificial recharge sources at HTF were added during predecessor model, GSA/FACT, calibration.

Response RAI-FF-3

Clarification of Hydraulic Conductivity Adjustments

Section 2.9 of *Pre- And Post-Processing Software Associated with the GSA/FACT Groundwater Flow Model,* WSRC-TR-99-00106, describes the overall process for developing the GSA/PORFLOW model conductivity fields, and Section 2.9.5 of the document describes the tools available for making hydraulic conductivity adjustments to match calibration targets. Figure RAI-FF-3.1 provides a full listing of the hydraulic conductivity adjustments made in the GSA/PORFLOW model to the initial conductivity field based primarily on mud fraction data from sediment cores, as requested in the RAI-FF-3 *Path Forward* (see WSRC-TR-99-00106, Section 2.9.5 for further explanation of the adjustment process).

Figure RAI-FF-3.1: Contents of “Cal.dat” Defining Adjustments to the Initial GSA/PORFLOW Hydraulic Conductivity Field

```
!
! GLOBAL ADJUSTMENTS TO BASELINE
./Polygons/All.ply
'../Hydrostrat/surf1.out',0.0
'../Hydrostrat/surf2.out',0.0
                             1.    38.     38.       1.    0.005   1.e+20   1.e+20
1.   'au'  .true.
./Polygons/All.ply
'../Hydrostrat/surf2.out',0.0
'../Hydrostrat/surf3.out',0.0
                             1.    1.e-3   1.e-3     1.    1.e-5   1.e-5    1.e+20
1.   'cu'  .true.
./Polygons/All.ply
'../Hydrostrat/surf3.out',0.0
'../Hydrostrat/surf4.out',0.0
                             1.15  1.e-20  1.e+20    1.    0.05    1.e+20   1.e+20
1.   'au'  .true.
./Polygons/All.ply
'../Hydrostrat/surf4.out',0.0
'../Hydrostrat/surf5.out',0.0
                             1.    1.e-20  1.e+20    1.    0.005   0.05     1.e+20
1.   'cu'  .true.
./Polygons/All.ply
'../Hydrostrat/surf5.out',0.0
'../Hydrostrat/surf7.out',0.0
                             1.15  1.e-20  1.e+20    1.    0.05    1.e+20   1.e+20
1.   'au'  .true.
!
! PERTURBATIONS ON BASELINE
./Polygons/LA_1.ply
'../Hydrostrat/surf3.out',0.0
'../Hydrostrat/surf4.out',0.0
                             1.3   1.e-20  1.e+20    1.    0.05    1.e+20   1.e+20
1.   'au'  .true.
./Polygons/LA_2.ply
'../Hydrostrat/surf3.out',0.0
'../Hydrostrat/surf4.out',0.0
                             1.    1.e-20  5.        1.    0.05    1.e+20   1.e+20
1.   'au'  .true.
./Polygons/TC_1.ply
'../Hydrostrat/surf4.out',0.0
'../Hydrostrat/surf5.out',0.0
                             1.    1.e-20  1.e+20    1.    0.001   0.01     1.e+20
1.   'cu'  .true.
./Polygons/TC_2.ply
'../Hydrostrat/surf4.out',0.0
'../Hydrostrat/surf5.out',0.0
                             1.    1.e-20  1.e+20    1.    0.01    0.05     1.e+20
1.   'cu'  .true.
./Polygons/TC_3.ply
'../Hydrostrat/surf4.out',0.0
'../Hydrostrat/surf5.out',0.0
                             1.    1.e-20  1.e+20    1.    0.0001  0.001    1.e+20
1.   'cu'  .true.
./Polygons/UA_1.ply
'../Hydrostrat/surf5.out',0.0
'../Hydrostrat/surf7.out',0.0
                             0.8   1.e-20  1.e+20    1.    0.05    1.e+20   1.e+20
1.   'au'  .true.
./Polygons/UA_2.ply
'../Hydrostrat/surf5.out',0.0
'../Hydrostrat/surf7.out',0.0
                             1.    1.e-20  4.        1.    0.05    1.e+20   1.e+20
```

The adjustment process involved multiple steps. In the first step, global adjustments were made to the five-hydrostratigraphic zones. These perturbations are summarized, by zone, as follows:

- Gordon Aquifer: set horizontal hydraulic conductivity (K_h) = 38 ft/d and require vertical hydraulic conductivity (K_v) > 0.005 ft/d (i.e., values less than 0.005 ft/d are set to 0.005 ft/d)
- Gordon Confining Unit: set K_h = 1.0E-03 ft/d and K_v = 1.0E-05 ft/d
- UTRA-LZ: increase K_h by 1.15 × and require K_v > 0.05 ft/d
- TCCZ, UTR Aquifer: Require $0.005 < K_v < 0.05$ ft/d
- UTRA-UZ: increase K_h by 1.15 × and require K_v > 0.05 ft/d

In the second step of the adjustment process, adjustments were made to specific zones based on more local calibration needs. Figures RAI-FF-3.2, RAI-FF-3.3, and RAI-FF-3.4 illustrate the selection polygons defining the targeted zones (the Gordon Confining Unit and Gordon Aquifer did not need adjustments).

Figure RAI-FF-3.2: Key Model Calibration Selection Polygons for the UTRA-UZ

Figure RAI-FF-3.3: Key Model Calibration Selection Polygons for the TCCZ

Figure RAI-FF-3.4: Key Model Calibration Selection Polygons for the UTRA-LZ

The hydraulic conductivity adjustments made near H Area were based on calibration to well water levels and were summarized by the following:

- LA_2 polygon: Require $K_h < 5$ ft/d and $K_v > 0.05$ ft/d
- TC_3 polygon: Require $0.0001 < K_v < 0.001$ ft/d
- UA_2 polygon: Require $K_h < 4$ ft/d and $K_v > 0.05$ ft/d
- UA_4 polygon: Require $K_h < 1.5$ ft/d and $K_v > 0.005$ ft/d
- In the third step of the adjustment process, global adjustments were made to accommodate porting the GSA model from the FACT code to the PORFLOW code. [WSRC-TR-2004-00106] These adjustments, by zone, were:
- UTRA-LZ: Increase K_h by 1.35×
- TCCZ: Decrease K_v by 0.5×
- UTRA-UZ: Increase K_h by 1.25×

The remaining adjustments impose conductivity values specific to alluvium and pumping test areas. The resulting conductivity field is partially depicted by the vertically averaged values within the UTR Aquifer shown in Figures RAI-FF-3.5 through RAI-FF-3.7. The UTRA-UZ average includes values above the water table.

Figure RAI-FF-3.5: Vertically Averaged Horizontal Conductivity in the UTRA- UZ

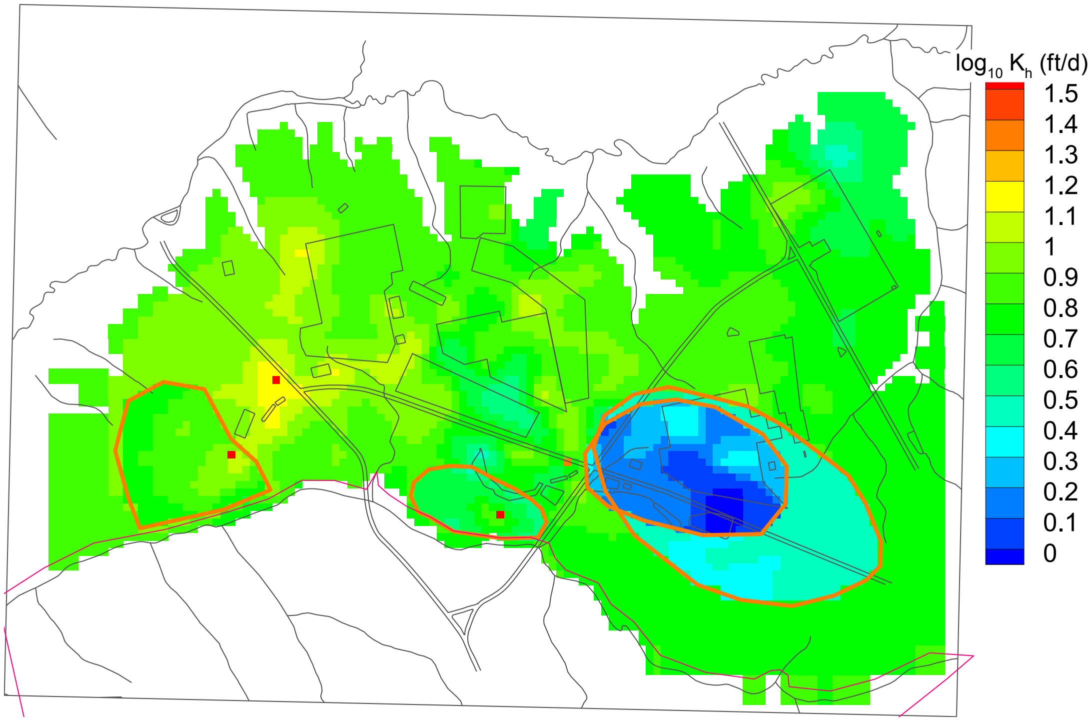


Figure RAI-FF-3.6: Vertically Averaged Vertical Conductivity in the UTR Aquifer TCCZ

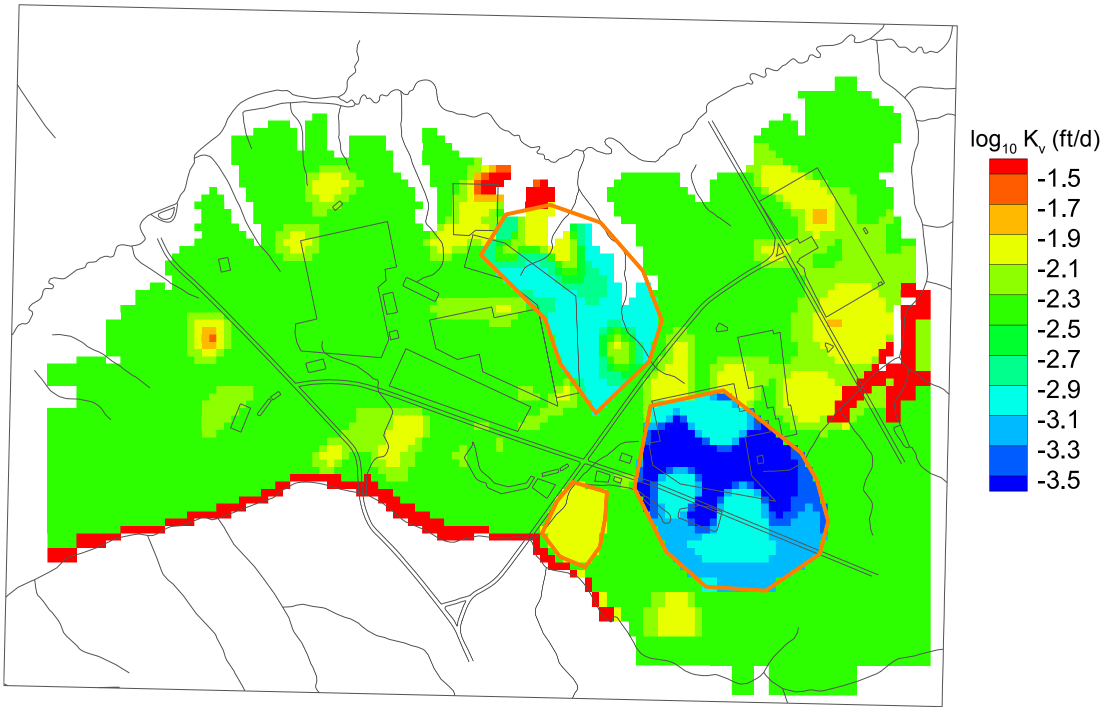
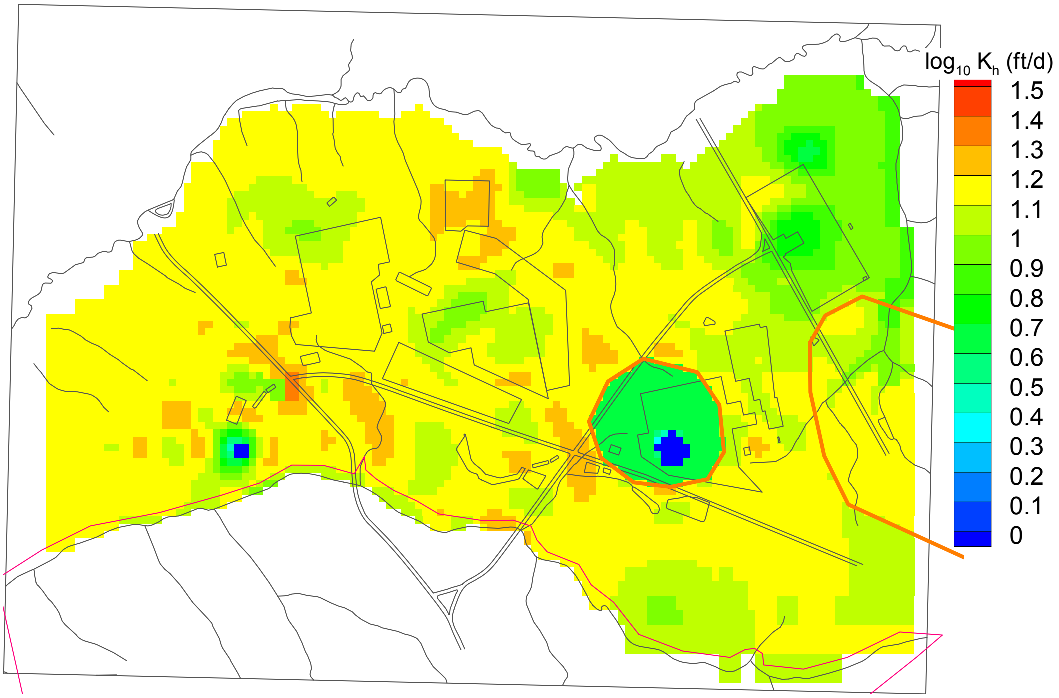


Figure RAI-FF-3.7: Vertically Averaged Horizontal Conductivity in the UTRA-LZ



Impact of Far Field Flow Variability

The risk significance of hydraulic conductivity adjustments and other settings in the GSA/PORFLOW far field flow model can be assessed by considering the sensitivity of solute transport simulations to potential variations in the aquifer velocity field, which affects plume travel time and distance. The potential variability in aquifer flow rate in the GSA was considered in the response to RAI-47 of *Response To Additional Information Request on Draft Section 3116 Determination For Salt Waste Disposal at Savannah River Site*, CBU-PIT-2005-00131. In the RAI-47 response, "Fast" and "Slow" velocity fields were generated to complement the "Nominal" GSA/PORFLOW simulation by adjusting 1) recharge, 2) leakage through the Gordon Confining Unit, and 3) effective porosity. For the off-nominal scenarios, each of the three parameters was adjusted to its plus or minus 95 % confidence level value to produce the fastest and slowest possible solute travel times. The UTR hydraulic conductivity field was then re-adjusted to maintain approximate calibration to hydraulic head targets. Particle tracking for the HTF and the Nominal, Fast, and Slow aquifer flow fields are shown in Figures RAI-FF-3.8 through RAI-FF-3.10. The three alternatives exhibit a wide variation in travel time to a point 100 meters from the HTF facility boundary corresponding to a significant variation in the average UTR Aquifer flow rate.

Figure RAI-FF-3.8: Groundwater Pathlines and 10-Year Time Markers for the Nominal GSA/PORFLOW Flow Field

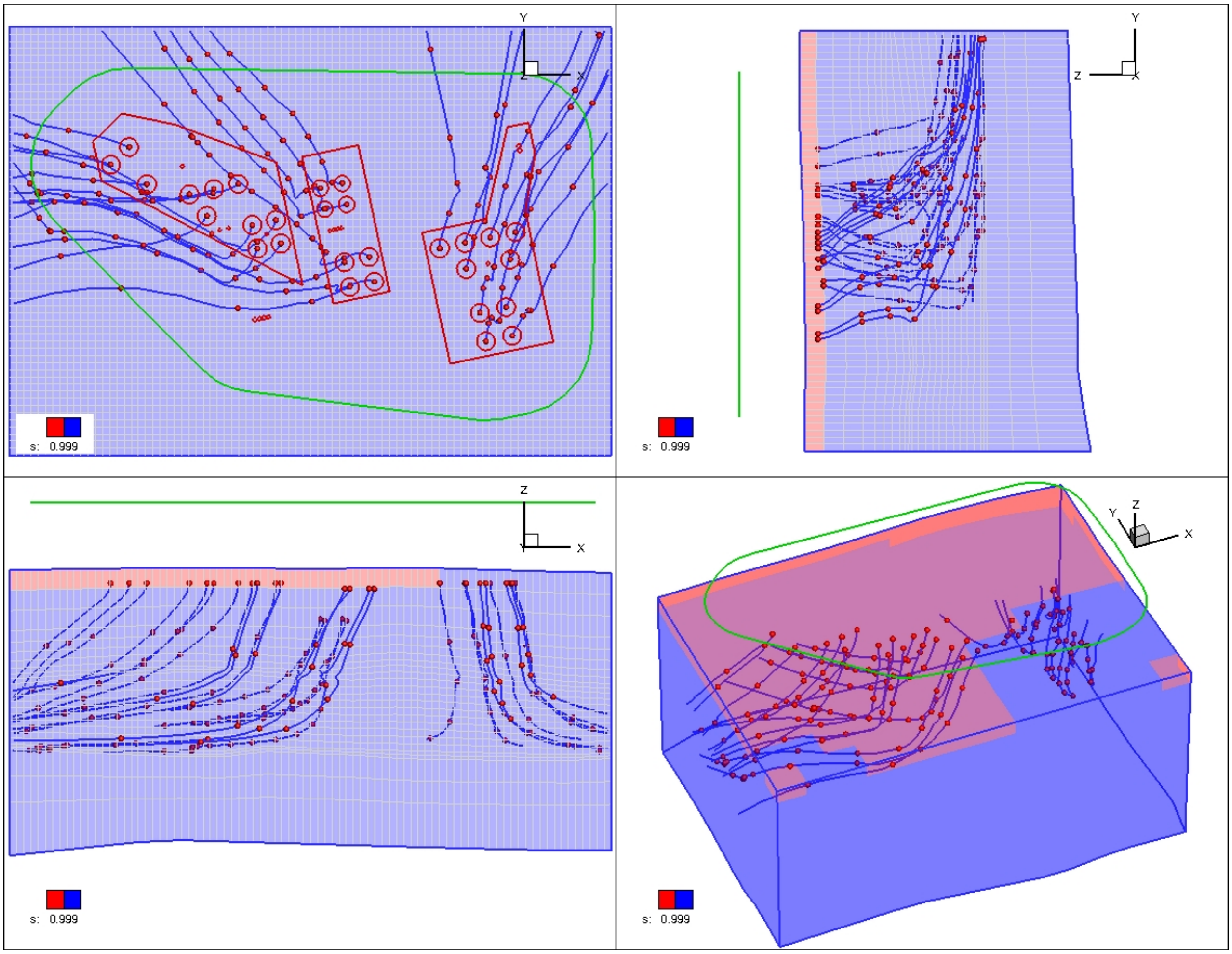


Figure RAI-FF-3.9: Groundwater Pathlines and 10-Year Time Markers for the Alternative Fast GSA/PORFLOW Flow Field

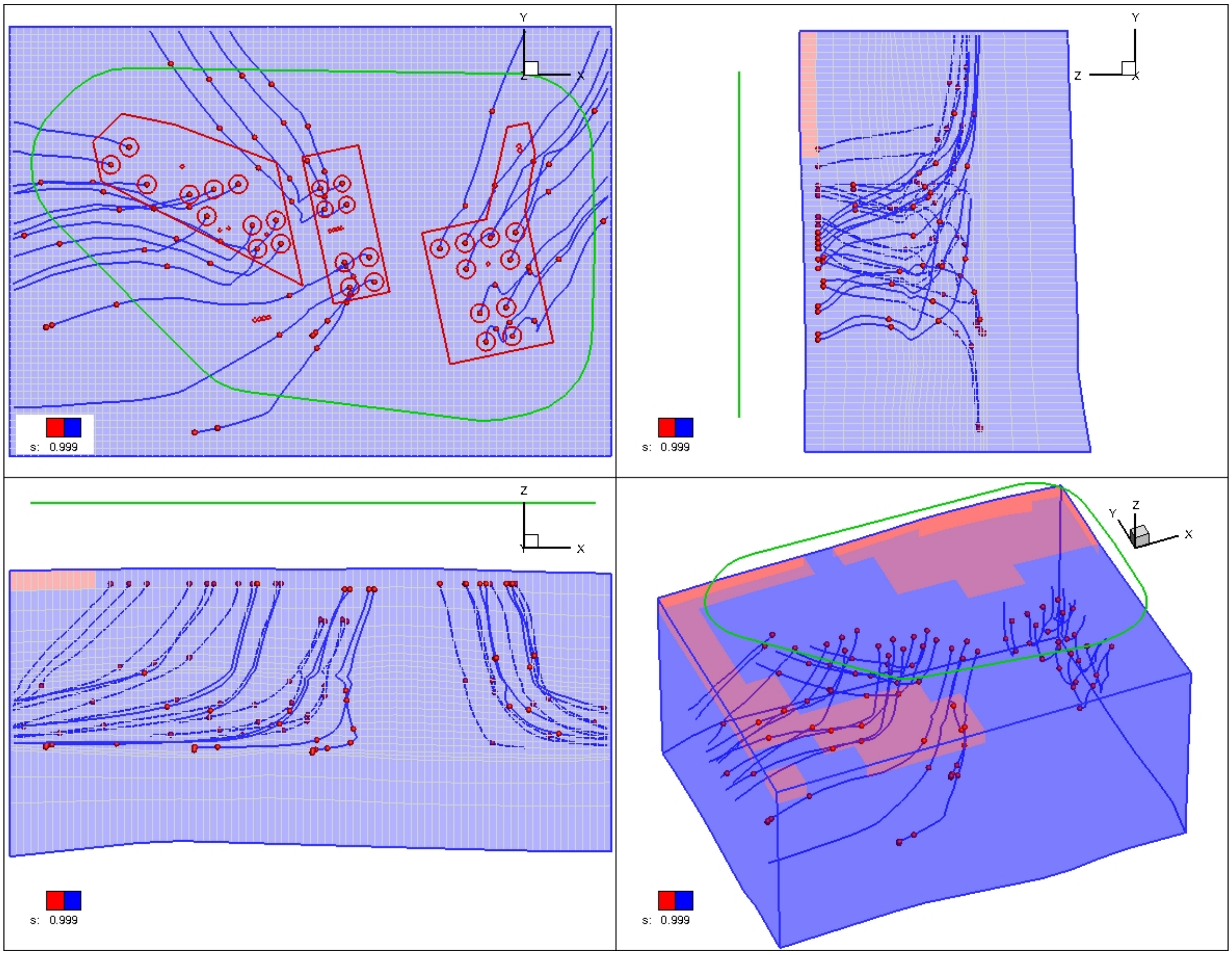
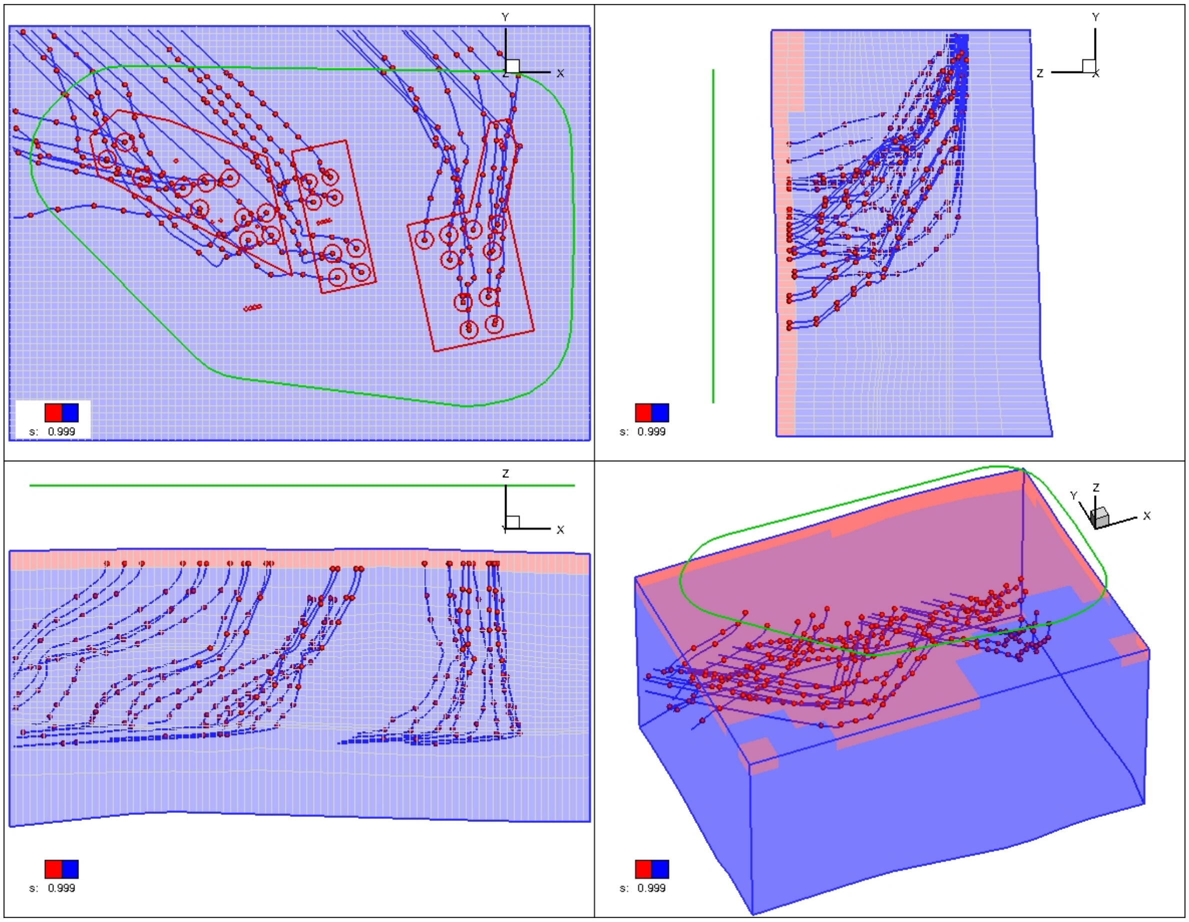


Figure RAI-FF-3.10: Groundwater Pathlines and 10-Year Time Markers for the Alternative Slow GSA/PORFLOW Flow Field



The HTF PA GoldSim Model was used to perform additional deterministic sensitivity runs using flow variability. These new modeling runs provide insights to the sensitivity of peak doses relative to variability in aquifer flow rate. For this analysis, all of the Base Case (Case A) assumptions were used except that the Darcy velocities along the pathlines were varied for each run. The objective of this analysis was to evaluate, non-mechanistically, the flow variability independent of cause, to understand the relative risk importance of various inputs to hydrogeology (hydraulic conductivities, recharge rates, barrier degradation rates) better.

The HTF GoldSim Model was deterministically executed three times, each time using a different set of pathline specific Darcy velocities (displayed in Table RAI-FF-3.1). The three sets of Darcy velocities represent the nominal set used in the PA, a fast set, and a slow set. The Darcy velocities were based on the times that breakthrough curve peaks, generated by instantaneous releases of conservative tracers at the individual waste tanks, reached the 100-meter distance from the HTF (the 100-meter distance from HTF is shown in Figure 5.2-1 of the HTF PA). The Darcy velocities were in turn, generated by dividing the pathline distance to the 100-meter distance from the HTF by the peak breakthrough times to generate a pore velocity and multiplying the pore velocity by the porosity. Note that because the nominal pathline distances were used to generate all three velocity fields, the sets represent approximations, which can be (and are) used only to examine specific effects on processes such as dilution.

Table RAI-FF-3.1: Darcy Velocities Used in the Far Field Flow Variability Study

Tank or Ancillary Structure	Slow Darcy Velocity Rates (ft/yr)	Nominal Darcy Velocity Rates (ft/yr)	Fast Darcy Velocity Rates (ft/yr)
T9	2.6	4.3	4.4
T10	2.4	3.7	4.0
T11	3.0	4.4	4.4
T12	2.8	3.9	3.7
T13	6.4	10.8	14.3
T14	3.8	4.2	5.9
T15	6.3	9.8	11.4
T16	7.0	13.1	14.1
T21	4.3	10.7	10.7
T22	3.7	10.0	11.7
T23	4.0	9.0	9.8
T24	4.0	10.5	11.6
T29	3.8	6.2	7.9
T30	5.1	6.2	7.7
T31	4.0	6.5	7.9
T32	3.8	5.9	7.6
T35	7.1	8.1	9.4
T36	7.6	8.3	9.9
T37	7.6	8.8	9.5
T38	4.9	5.9	5.9
T39	5.3	6.8	6.8
T40	5.8	8.2	8.5
T41	5.6	9.2	10.3
T42	4.6	5.5	6.5
T43	4.4	7.3	9.7
T48	4.3	5.0	7.1
T49	3.5	5.2	7.3
T50	4.1	4.5	6.0
T51	2.7	4.2	6.1
HPT2	3.4	4.1	4.3
HPT3	3.3	4.0	4.2
HPT4	3.3	3.8	4.0
HPT5	6.8	11.9	13.6
HPT6	7.0	11.4	13.0
HPT7	7.7	11.2	12.4
HPT8	7.3	10.8	12.9
HPT9	7.1	11.3	13.6
HPT10	7.9	11.0	11.9
E242_H	4.3	11.3	12.5
E242_16H	4.8	6.7	8.4
E242_25H	6.4	6.7	7.7
HTF_T_Line1	10.9	19.1	29.9
HTF_T_Line2	3.2	5.1	5.8
HTF_T_Line3	12.5	15.7	27.4
HTF_T_Line4	6.9	12.0	13.0
CTSO	4.2	7.1	8.9
CTSN	4.9	8.4	10.1

Figure RAI-FF-3.11 depicts peak dose curves (from any sector) for the three runs. Comparing the dose curves presented in Figure RAI-FF-3.11 it can be seen that there is relatively little difference between the three dose curves. In early years, the slow-flow results have the highest peaks of the three runs, mainly reflecting a decrease in Tc-99 dilution, due to lower flow rates. In later years, the lower flow-rate run generates lower doses, reflecting the dose contribution of Ra-226, a radionuclide with a relatively high decay rate, that is moderately to strongly sorbed in the soil. For Ra-226, the influence of the slower flow rates on decay dominates over the influence of decreased dilution. Even when other species control the shape of the peak, such as Tc-99 at 10,000 years, the lower Ra-226 dose contributions lower the peak values. Comparing the nominal flow-rate results with the fast flow-rate results in Figure RAI-FF-3.11 shows that the faster-flow tends to produce slightly lower dose levels at early times and then increases the dose levels later on. This once again reflects the importance of dilution (which increases in the faster-flow runs) on the species controlling the early time doses, and the influence of flow rate on decay of the species controlling the later time doses.

Figure RAI-FF-3.11: Far Field Flow Variability Study Peak Doses

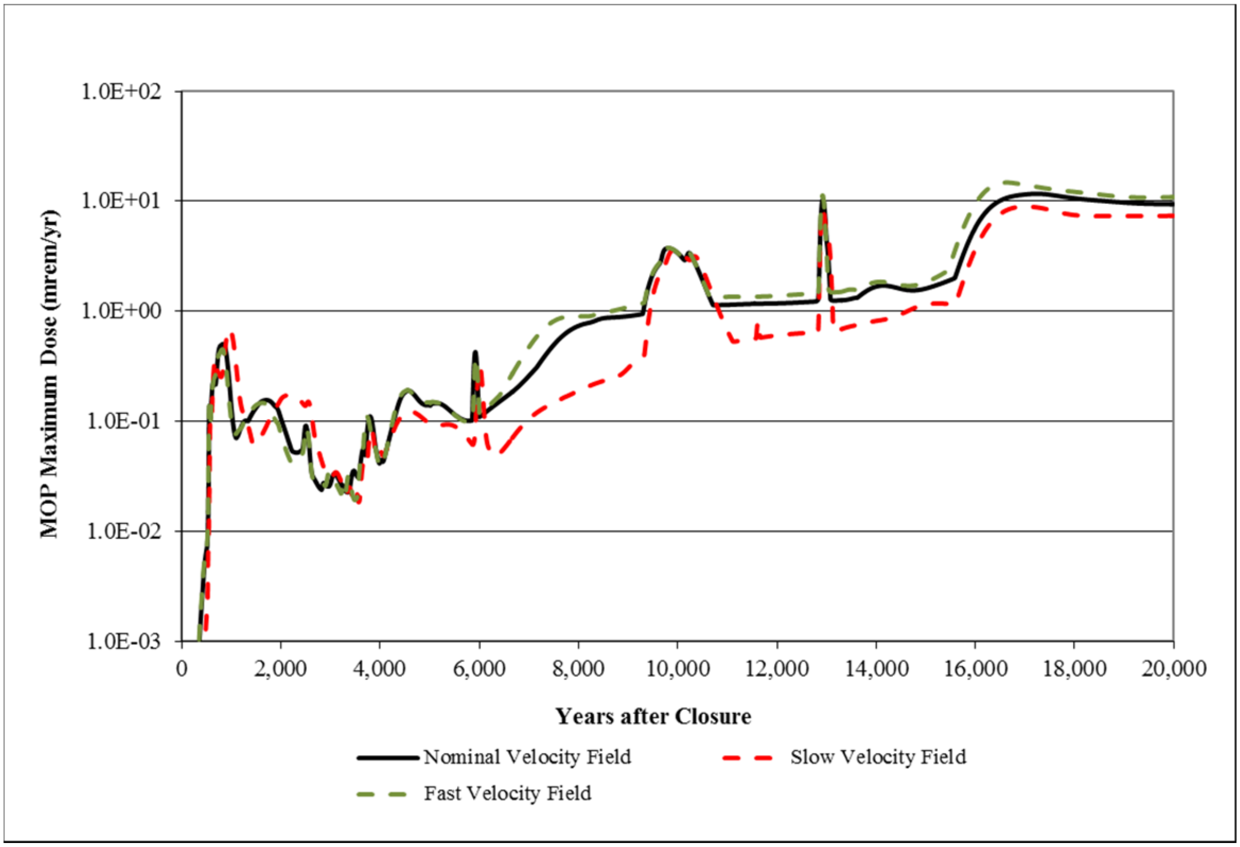


Table RAI-FF-3.2 shows the peak doses over 10,000 years and over 20,000 years from all three runs as a function of the Darcy velocity. Over a 10,000-year period, as flow rates increase from slow to nominal to fast, the peak values show a slight increase, reflecting a peak controlled by both Ra-226 and Tc-99. Over 20,000 years, increased flow rates result in a greater increase peak values reflecting the dominance of Ra-226 in dose contributions. Considering the three sets of Darcy velocities described above, the peak dose contribution does not reach 4 mrem/yr until after 10,000 years and does not exceed 15 mrem/yr within 20,000 years after HTF final facility closure.

Table RAI-FF-3.2: Summary of Far Field Flow Variability Study Results

Darcy Velocity Field	Peak Dose Over 10,000 Years (mrem/yr)	Time of Peak Dose Over 10,000 Years (yrs)	Peak Dose Over 20,000 Years (mrem/yr)	Time of Peak Dose Over 20,000 Years (yrs)
Slow	3.6	9,930	9.0	17,010
Nominal	3.8	9,810	11.7	17,230
Fast	3.8	9,800	14.8	16,630

These runs provide insight into the relationship between Darcy velocity and corresponding peak doses. The DOE recognizes that only modifying the Darcy velocity within the HTF GoldSim Model is a simplified representation of the far field flow changes represented in the Nominal, Fast, and Slow aquifer flow fields shown in Figures RAI-FF-3.8 through RAI-FF-3.10. The flow variability study performed is most informative in cases where the flow path lines did not change significantly.

DOE maintains that the nominal flow characterization currently embodied in the HTF models represents a reasonable estimate of hydrogeological behavior within the post-closure environment. As discussed in the responses to RAI-FF-1 and RAI-FF-2, the GSA model reasonably fits the underlying data and is adequate for the purposes of fate and transport modeling in the HTF PA, considering the risk significance of hydraulic conductivity and flow field variability discussed in this response.

As required by DOE Manual 435.1-1, maintenance of the HTF PA will include future updates to incorporate new information, update model codes, consider actual residual inventories, etc., as appropriate. The potential impact of new information will be evaluated using established site practices and procedures including preparation of an SA, as appropriate, consistent with DOE Manual 435.1-1 and DOE Guide 435.1-1. As waste tank-specific SAs are prepared it is anticipated that sensitivity analyses will be performed that explicitly address both individual waste tank inventories and unique waste tank conditions, such flow variability potential (e.g., waste tank-specific water level or potential for increased recharge). Because different as-modeled waste tank conditions may prove to be conservative or non-conservative (with respect to timing and/or magnitude of peak doses) depending on other corresponding waste tank conditions, these analyses are more suited to waste tank-specific sensitivity analyses than to changes to the overarching Base Case assumptions. For example, modeling assumptions that slow the far field flow may cause waste tank releases to reach the Gordon Aquifer sooner, which may increase dilution. The impact of a change in far field flow will vary depending on the quantity (total curies) and nature (short-lived versus long-lived radionuclides) of the released inventory.

RAI-FF-4

Time variant recharge rates and flow are not considered in the HTF/PORFLOW® model but may be risk-significant. DOE should evaluate the impact of time-variant recharge rates and flow on HTF Performance Assessment predictions.

Basis

HTF flow fields may be variable over time due to climatic variability or engineered barrier degradation; however, DOE uses a long-term, steady state (saturated zone) model to predict contaminant fate and transport at the HTF. Changes in flow rates and directions at HTF over time may have a significant impact on dose predictions.

While the GSA/PORFLOW® model uses a recharge rate of 19 in/yr over most areas of the model domain (WSRC-TR-2004-00106, Rev. 0), the long-term infiltration rate is assumed to be approximately 12 in/yr after degradation of the engineered closure cap. Additionally, the engineered closure cap at HTF is assumed to be effective at reducing recharge to relatively low rates for hundreds to thousands of years following HTF closure. Yet, the impact of the closure cap on recharge rates following facility closure is not considered in the far-field model.

While the closure cap is generally expected to reduce infiltration, the area between the west and east closure caps may represent an area of increased infiltration due to runoff from the caps. The impact of increased runoff from the caps was evaluated in Portage (PORTAGE-08-022, Rev. 0), but infiltration was limited in the drainage area between the west and east caps and a more detailed evaluation of the effect of the cap on HTF performance would be beneficial.

While the FTF Performance Assessment (SRS-REG-2007-00002, Rev. 1) did not consider time-variant recharge rates, in most cases releases from the tanks were not assumed to occur until after the closure cap and cementitious materials were degraded and recharge rates were near long-term, steady-state values. However, time-variant recharge rates may be more risk-significant for HTF sources due to the fact that some tank liners are assumed to be initially failed and releases could occur much earlier in time prior to closure cap and cementitious material degradation (for submerged and partially submerged tank sources).

Path Forward

DOE could perform the following activities to evaluate the impact of time-variant recharge and flow at HTF. Note that some of the activities have been partially evaluated in PORTAGE-08-022, Rev. 0. This report can be used as a starting point in addressing this request for additional information but additional detail would be helpful.

- Compare modeled or hand-contoured potentiometric surfaces at various points in time to evaluate the potential for climatic variability to effect flow rates and directions at HTF. Note that observed flow field variability may be influenced by operations as discussed in RAI-FF- 2 and would not be necessarily indicative of long-term natural variability relevant to the HTF Performance Assessment.
- If found or thought to be significant, evaluate the potential impact of climatic variability on the HTF flow field. This would include evaluation of the impact of variability on dilution, dispersion, and cumulative impacts due to changes in flow rates and directions.
- Evaluate the impact of lower recharge rates due to the presence of an engineered closure cap on HTF water levels and the HTF flow field.

- Evaluate the impact of increased recharge in drainage areas, particularly the area between the west and east engineered closure caps, on HTF water levels and the HTF flow field.
- Evaluate the impact of engineered barrier degradation (e.g., closure cap and tank cement/grout) on HTF releases and the HTF flow field over time.

Response RAI-FF-4

HTF Potentiometric Mapping

In the majority of the Atlantic Coastal Plain, groundwater flow in the water table is driven by surface recharge and tends to be a muted expression of topography. The elevation of the water table varies with recharge, but the relative shape of the surface is generally stable. In the GSA, the major discharge points for the water table remain relatively constant in elevation and location. Thus, in response to moderate changes in climate and topography, the shape of the water table, location of the groundwater divide, and the associated flow vectors are expected to remain similar to present conditions.

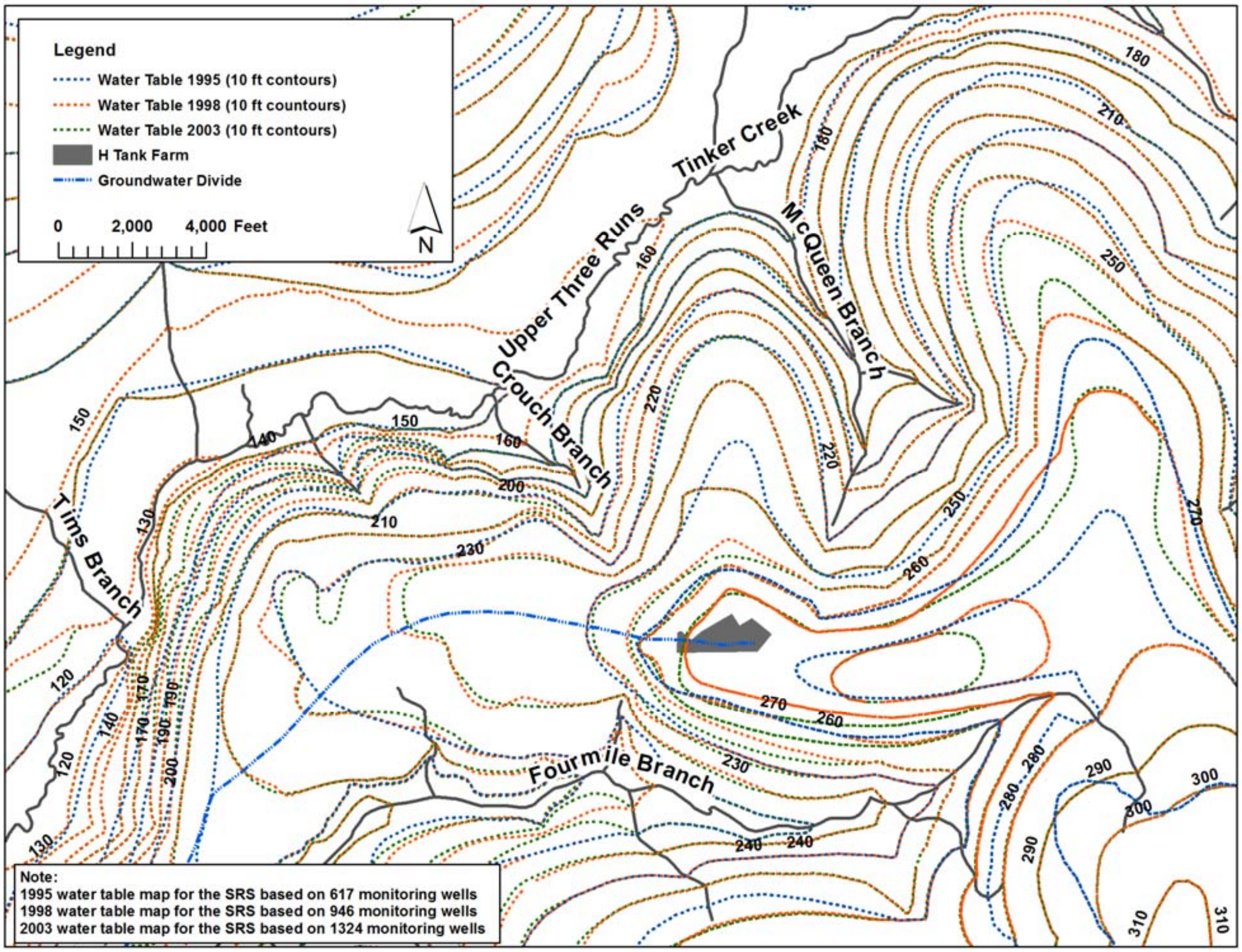
It is noteworthy that a groundwater contour map can vary slightly based on incomplete and uncertain input data used to create the surface. However, in the GSA, there is sufficient historical groundwater information to support the shape of the water table surface used to create the potentiometric surfaces and flow fields used in the PORFLOW modeling.

As stated above, groundwater elevations can vary vertically over time based on recharge. An evaluation of HTF monitoring well water levels in the water table (UTRA-UZ) over time versus HTF annual moving-average rainfall shows a strong correlation. In addition, these effects on the water table due to recharge are similar at all of the monitoring wells in HTF, indicating that as the water table rises and falls, the shape of the surface remains relatively unchanged. This strong correlation between rainfall and recharge would indicate that existing localized engineered barriers have very little effect on the overall water table shape and flow fields.

Contour maps of water tables and potentiometric surfaces can vary also, based on the contouring method used to create them. Most computerized contouring programs have difficulty interpreting the effects of topography, especially within the vicinity of discharge points (HTF PA Figure 4.2-21). In addition, determining contour shapes in areas of minimal data can result in a higher degree of uncertainty. However, the basic shape, orientation, and elevation of contoured surfaces in areas with abundant historical water level data, such as the GSA, can be determined with a higher degree of confidence.

A comparison of historic water table data over time is presented in Figure RAI-FF-4.1. Contour surfaces for three years (1995, 1998, and 2003) are overlain for comparison. For reference, the location of HTF and the associated groundwater divide have been denoted. For the HTF, groundwater discharge from the water table aquifer is to Crouch Branch and McQueen Branch, and ultimately to UTR, to the north and to Fourmile Branch to the south. As indicated by the contouring, slight variations exist between localized individual contours, but the overall shape (including groundwater divides), direction of flow, and discharge points remain constant. These contours are consistent with each other, indicating that there has not been a significant change in our understanding of long-term average water table conditions in the GSA since the mid-1990s. In 1995, 617 monitoring wells were used in contour mapping. In 1998, the number of wells increased to 946. Finally in 2003, the number of wells for contouring increased to 1,324. As indicated by this figure, varying the number of wells used to create the maps does not result in significant variation in shape of the overall contour map and the location, orientation and relative intensity of the groundwater divide within the vicinity of the HTF. Therefore, it is highly unlikely that modest changes in recharge to the water table, such as those experienced over the past two decades, or additional data will result in a significant shift in the groundwater divide that would alter discharge to surface water.

Figure RAI-FF-4.1: Water Table Contour Maps for GSA



Note:
1995 water table map for the SRS based on 617 monitoring wells
1998 water table map for the SRS based on 946 monitoring wells
2003 water table map for the SRS based on 1324 monitoring wells

Shapes and elevations of potentiometric surfaces of the UTRA-LZ in HTF are similar to the water table (see *Hydrogeologic Data Summary in Support of the H-Area Tank Farm Performance Assessment*, SRNL-STI-2010-00148, Figures 23 and 24). This is because the TCCZ, separating the UTRA-UZ from the UTRA-LZ, is considered a semi-confining zone and is not generally considered major impedance to the vertical movement of groundwater. The PORFLOW modeling, and associated flow fields, reflects this strong correlation between the two aquifer zones.

Finally, the Gordon Aquifer is separated from the UTR Aquifer by the Gordon Confining Unit. Within the vicinity of HTF, the Gordon Aquifer only discharges to UTR. The Gordon Confining Unit acts as a significant barrier to vertical movement of groundwater in the GSA and, as such, the aquifer reflects little variation in potentiometric surface shape and discharge over time due to minimal vertical recharge (see *Hydrogeologic Data Summary in Support of the H-Area Tank Farm Performance Assessment*, SRNL-STI-2010-00148, Figure 25).

To assess the potential impact of a low-permeability cover system over the HTF after final facility closure, a sensitivity study on aquifer flow was performed. Figure RAI-FF-4.2 shows groundwater pathlines emanating from HTF waste tanks for nominal conditions. Figure RAI-FF-4.3 shows the corresponding pathlines when the surface recharge is reduced by 100× over the area coinciding with the 1-meter facility boundary used for dose calculations. The general impact of the cap is to push groundwater deeper and toward the north.

Figure RAI-FF-4.2: Groundwater Pathlines for Nominal Conditions

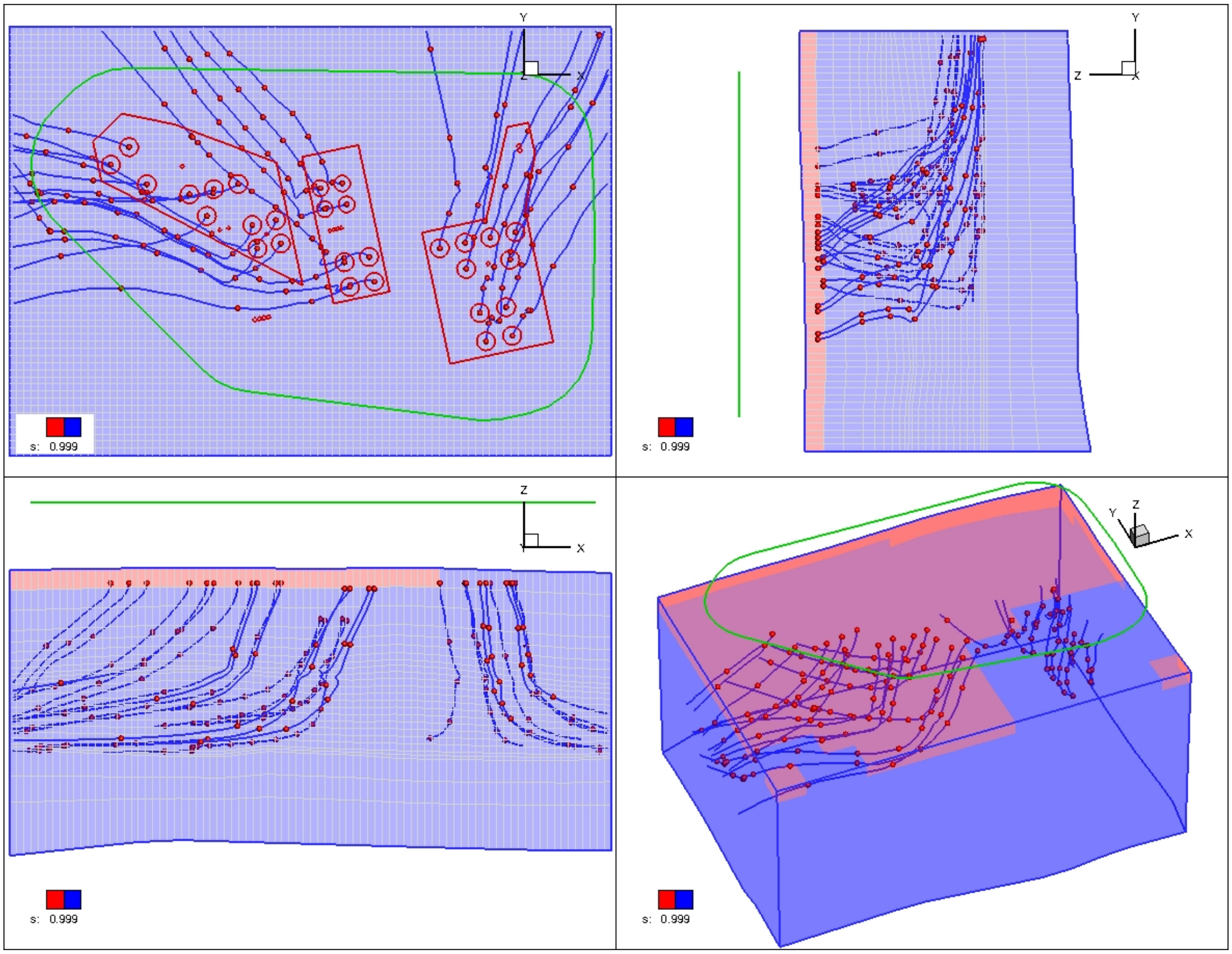
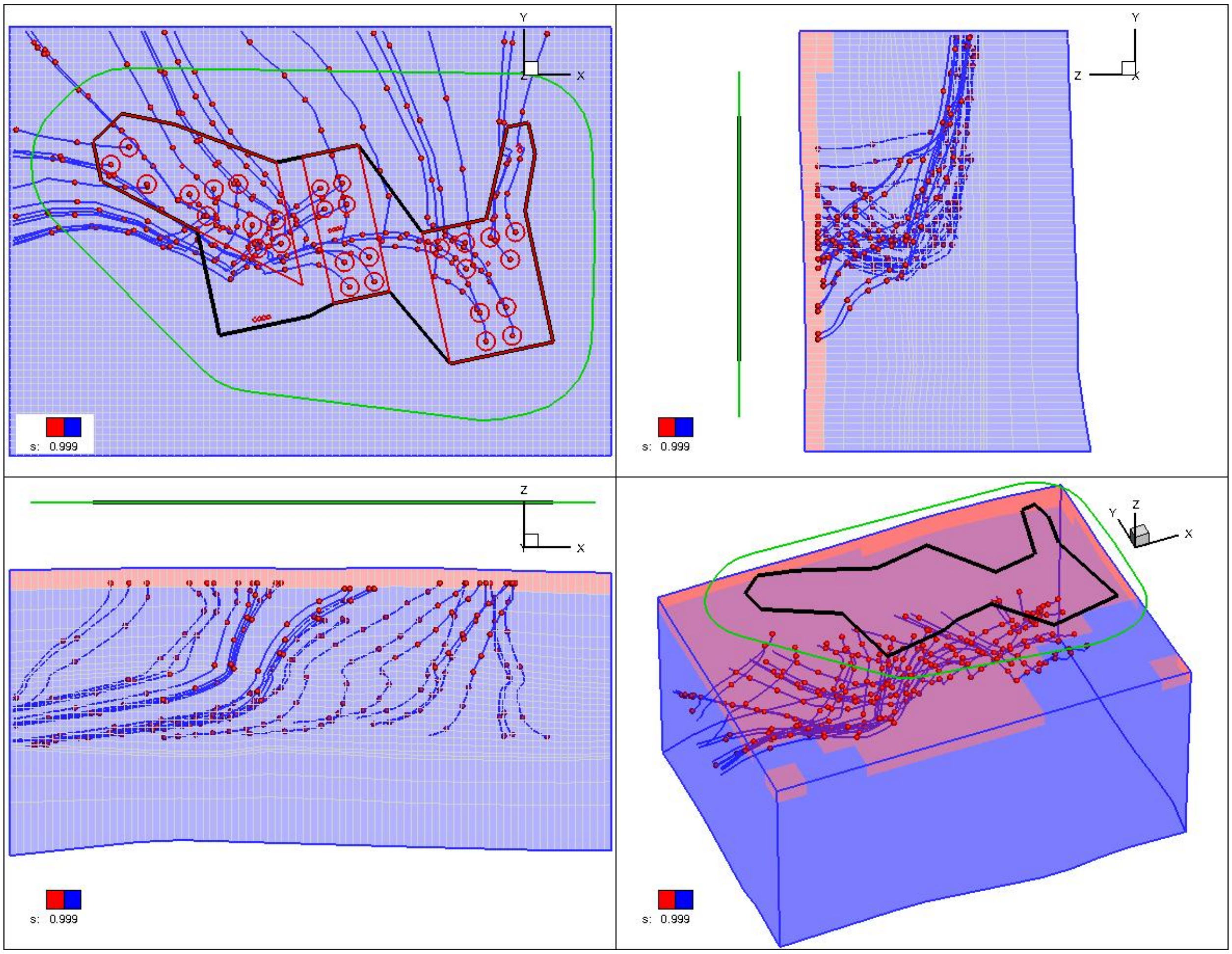


Figure RAI-FF-4.3: Groundwater Pathlines for Reduced Recharge Conditions



Impact of Far Field Flow Variability

As discussed in detail in the response to RAI-FF-3, the risk significance of hydraulic conductivity adjustments and other settings in the GSA/PORFLOW far field flow model can be assessed by considering the sensitivity of solute transport simulations to potential variations in the aquifer velocity field, which affects plume travel time and distance. Figure RAI-FF-3.11 depicts peak dose curves for three HTF sensitivity runs with the different runs reflecting far field flow variability (each run used a different set of pathline specific Darcy velocities). The three modeling runs represent a nominal flow set, a fast flow set, and a slow flow set. By comparing the dose curves associated with the three different sets (presented in Figure RAI-FF-3.11), it can be seen that there is relatively little difference between the dose results.

CC-FF-1

Page 59 of SRR-CWDA-2010-00093, Rev. 2 indicates that some HTF plumes are spread over both aquifers and that higher vertical dispersivities are generally needed for the eastern plumes. Clarify what tank sources are spread over both aquifers and the differences between vertical dispersion for western versus eastern sources in GoldSim® probabilistic modeling.

Response CC-FF-1

To help clarify which waste tank sources are spread over both the UTRA-UZ and UTRA-LZ, a complete series of plume plots from PORFLOW waste tank-specific simulations was generated (see Figures CC-FF-1.1 through CC-FF-1.10). The plumes were generated from initial emplacements of a conservative constituent (i.e., constituent that is not sorbed, is not solubility controlled, and does not decay) and present concentrations for the time period when the center-line concentrations would peak at a point 100 meters from the HTF boundary. Both areal and vertical plots are presented. The concentrations in the plots are in moles per liter (mol/L) and represent maximum values perpendicular to the plane being viewed. The Y-Z and X-Z plots in the upper right hand and lower left hand corners, respectively, of the figures can be used to evaluate the vertical extent of the plume at the 100-meter point. In the vertical profiles, the top eight cells represent the UTRA-UZ, the next two cells represent the TCCZ of the UTR Aquifer, the next 10 cells represent the UTRA-LZ, the following two cells represent the Gordon confining zone, and the bottom three cells represent the Gordon Aquifer.

These new plume figures (Figures CC-FF-1.1 through CC-FF-1.10) show that most of the contaminant resides in the UTRA-LZ at the 100-meter point. Low concentrations (< 3E-10 mol/L) of contaminant occur in the UTRA-UZ in plumes emanating from Tanks 39, 40, 49, and 51 as shown in Figures CC-FF-1.2, CC-FF-1.3, CC-FF-1.8, and CC-FF-1.10, respectively. Higher concentrations (> 3E-10 mol/L) of contaminant occur in the UTRA-UZ in plumes emanating from Tanks 41 and 43 as shown in Figures CC-FF-1.4 and CC-FF-1.6, respectively.

Peak dose results from benchmarking (shown in Section 5.6.2.2. of the HTF PA) validate the modification of vertical dispersivity to improve the match between the PORFLOW and GoldSim models. The eastern waste tank releases followed relatively straight pathlines, and the waste tank release plumes did not show the same high degree of transverse spreading as found in the western waste tank releases, therefore the horizontal transverse dispersivity was not adjusted during the benchmarking process.

Figure CC-FF-1.1: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 38

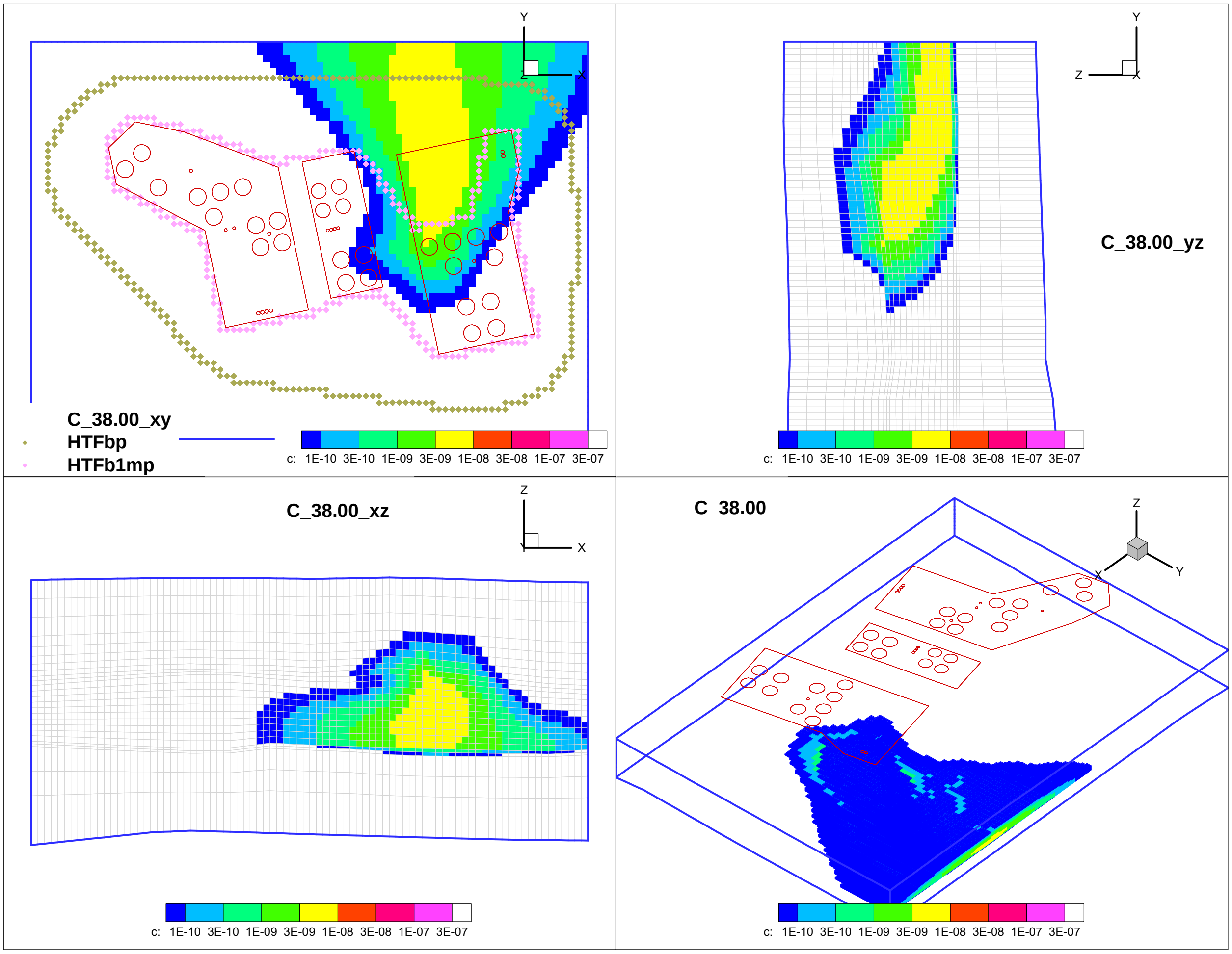


Figure CC-FF-1.2: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 39

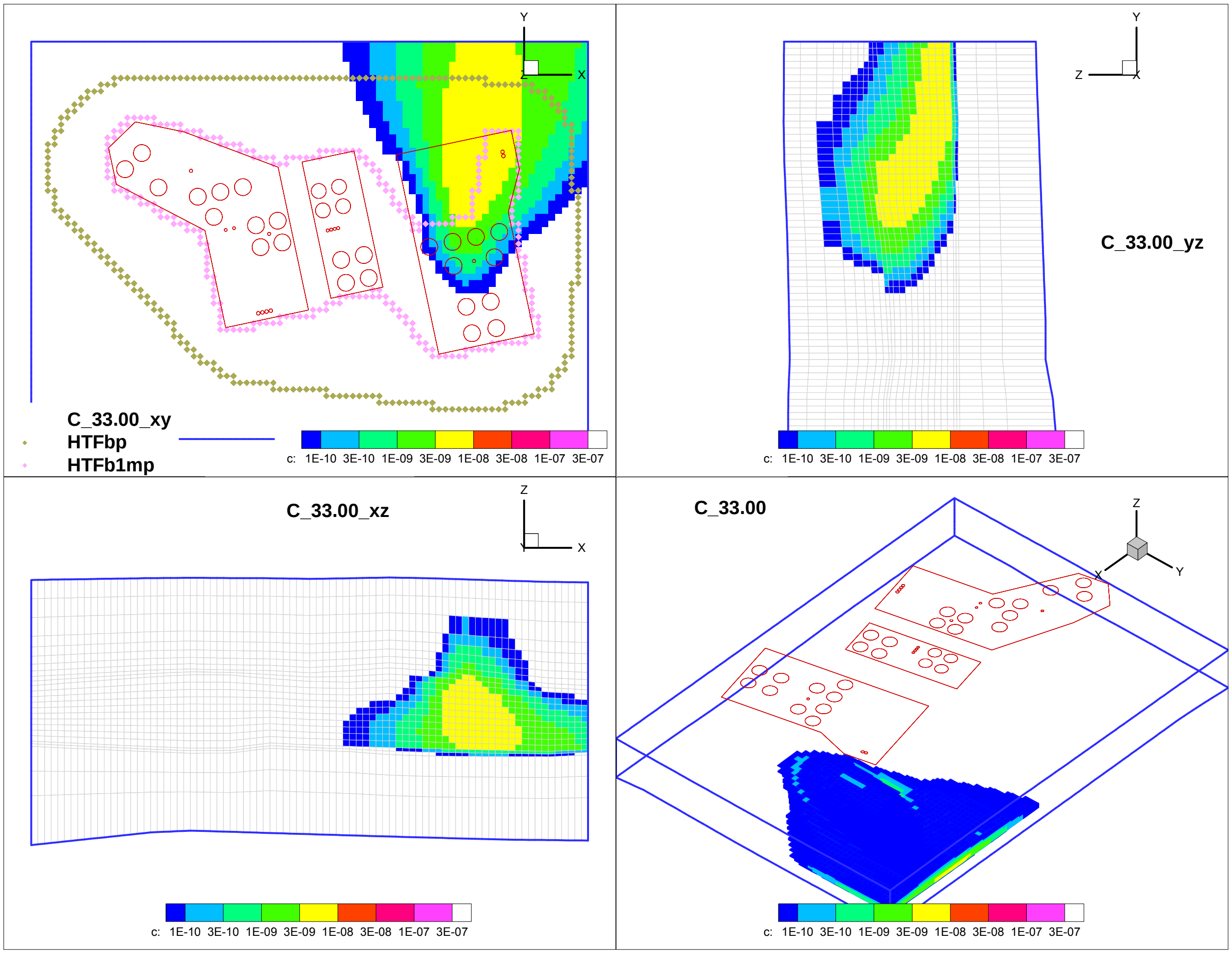


Figure CC-FF-1.3: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 40

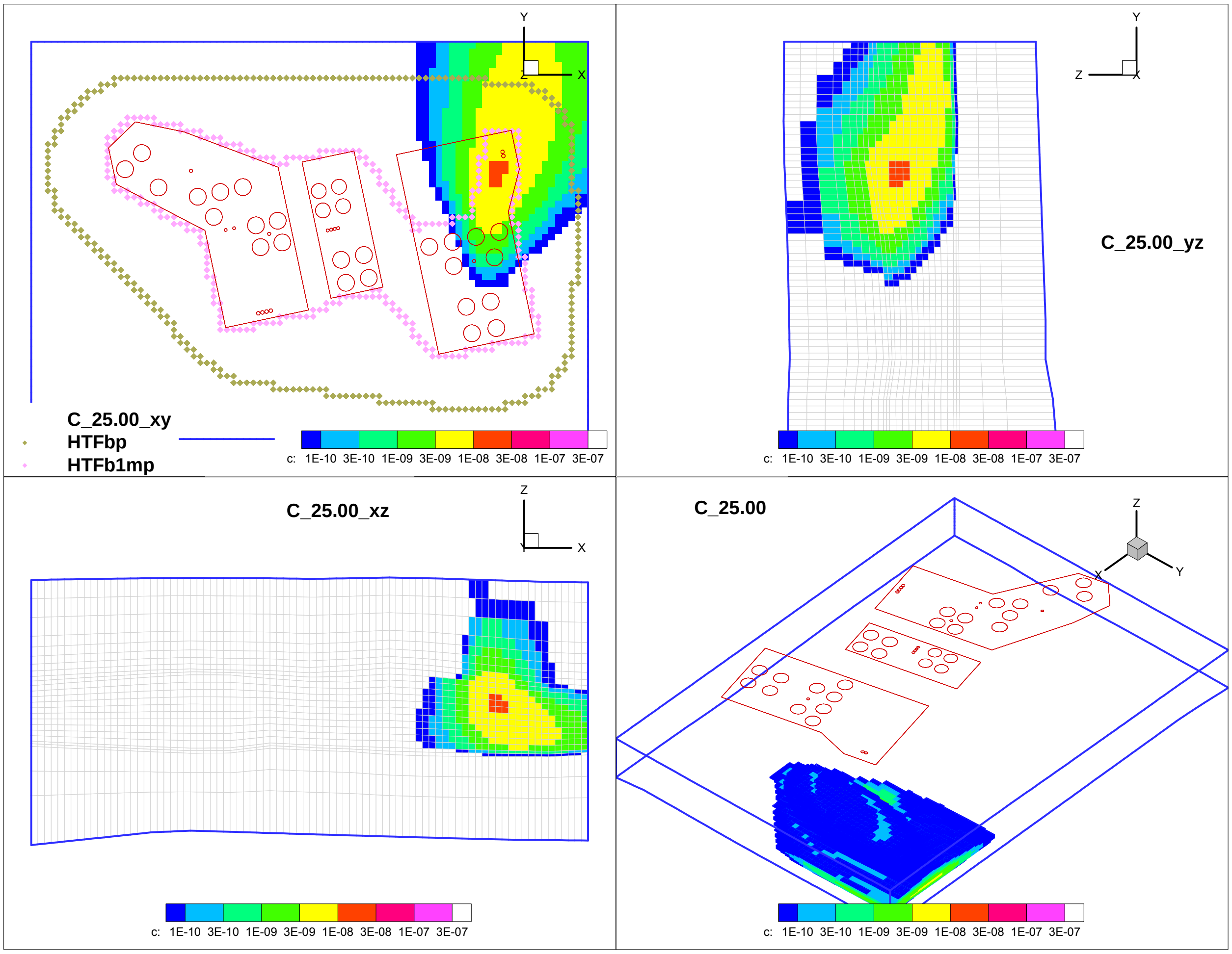


Figure CC-FF-1.4: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 41

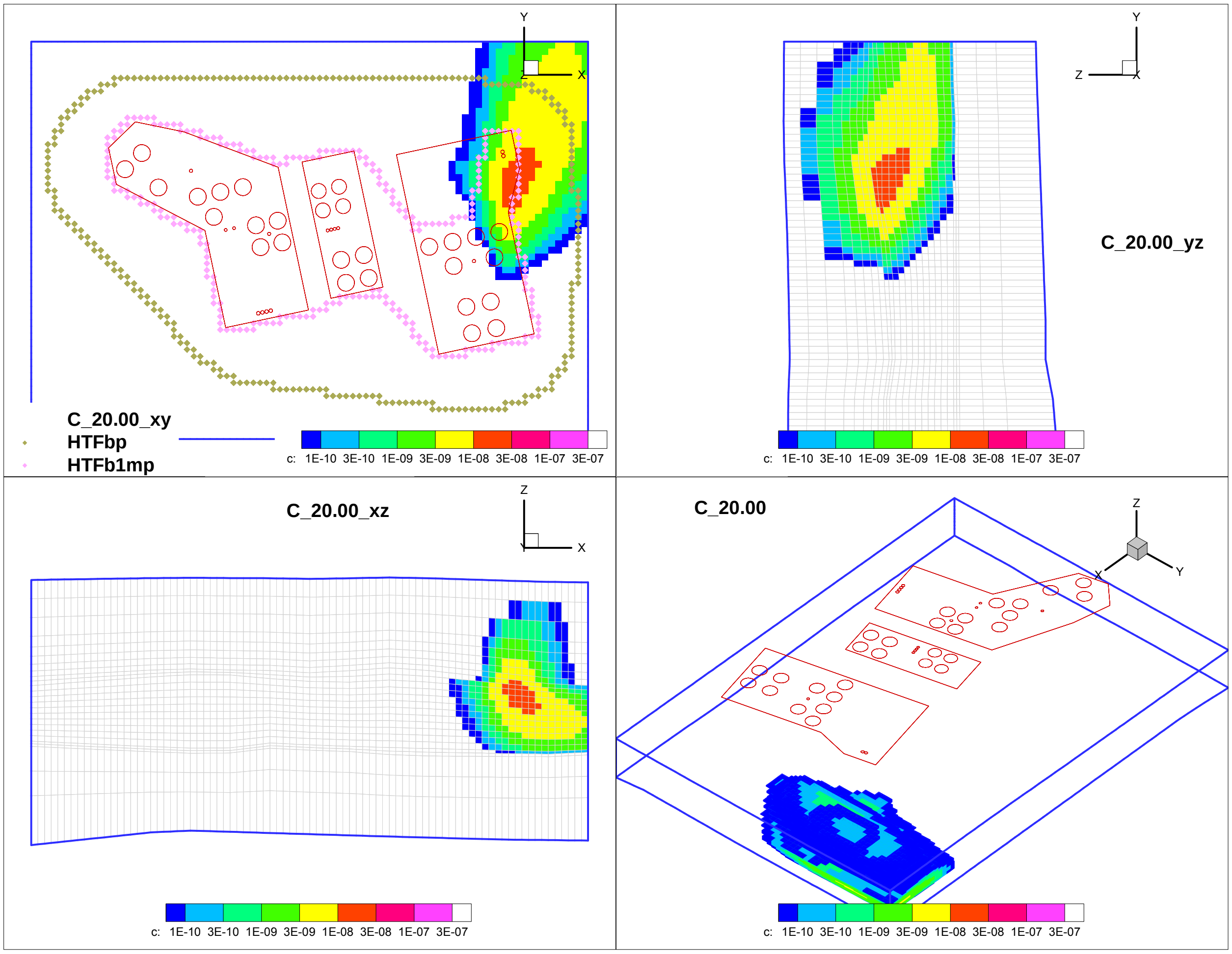


Figure CC-FF-1.5: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 42

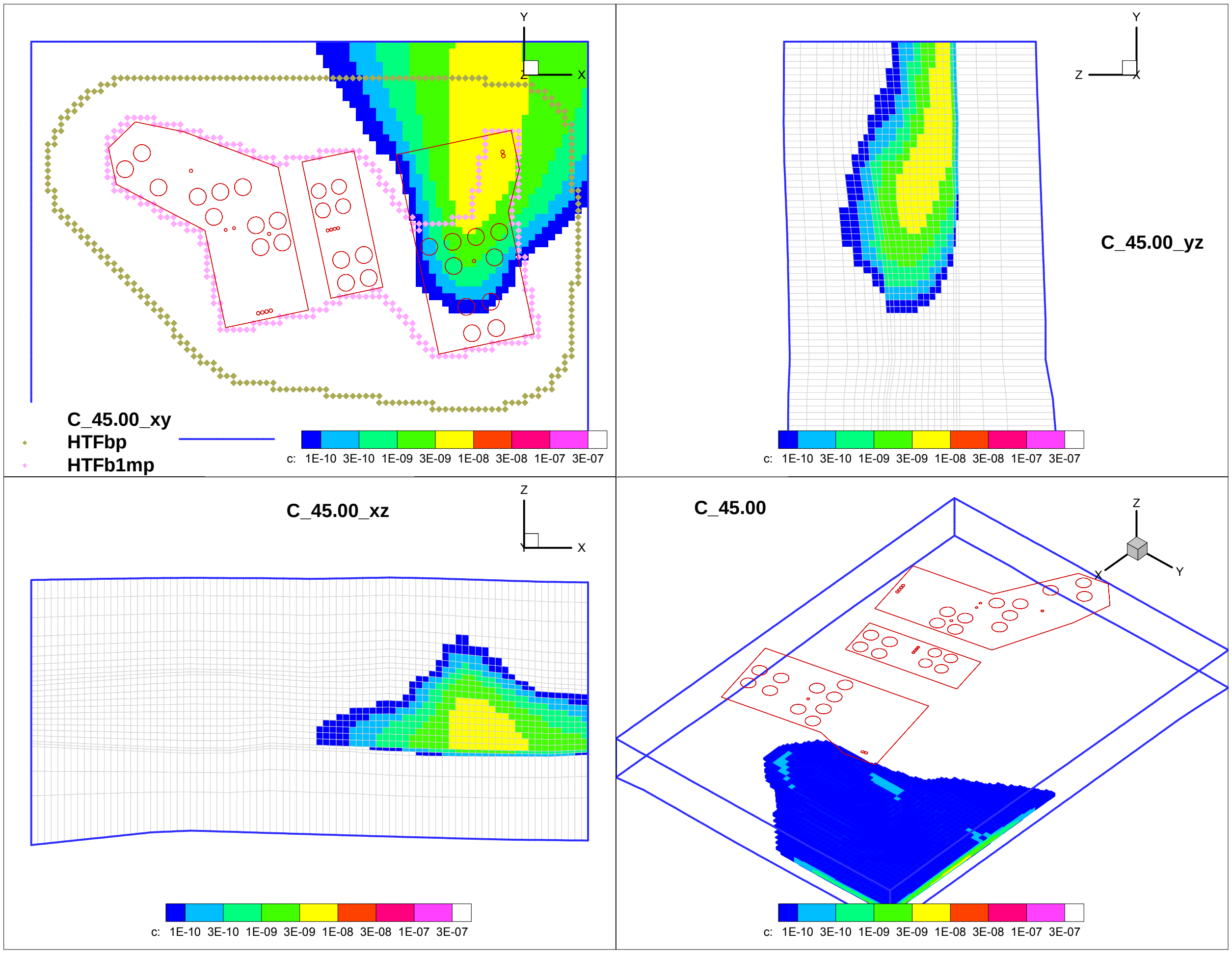


Figure CC-FF-1.6: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 43

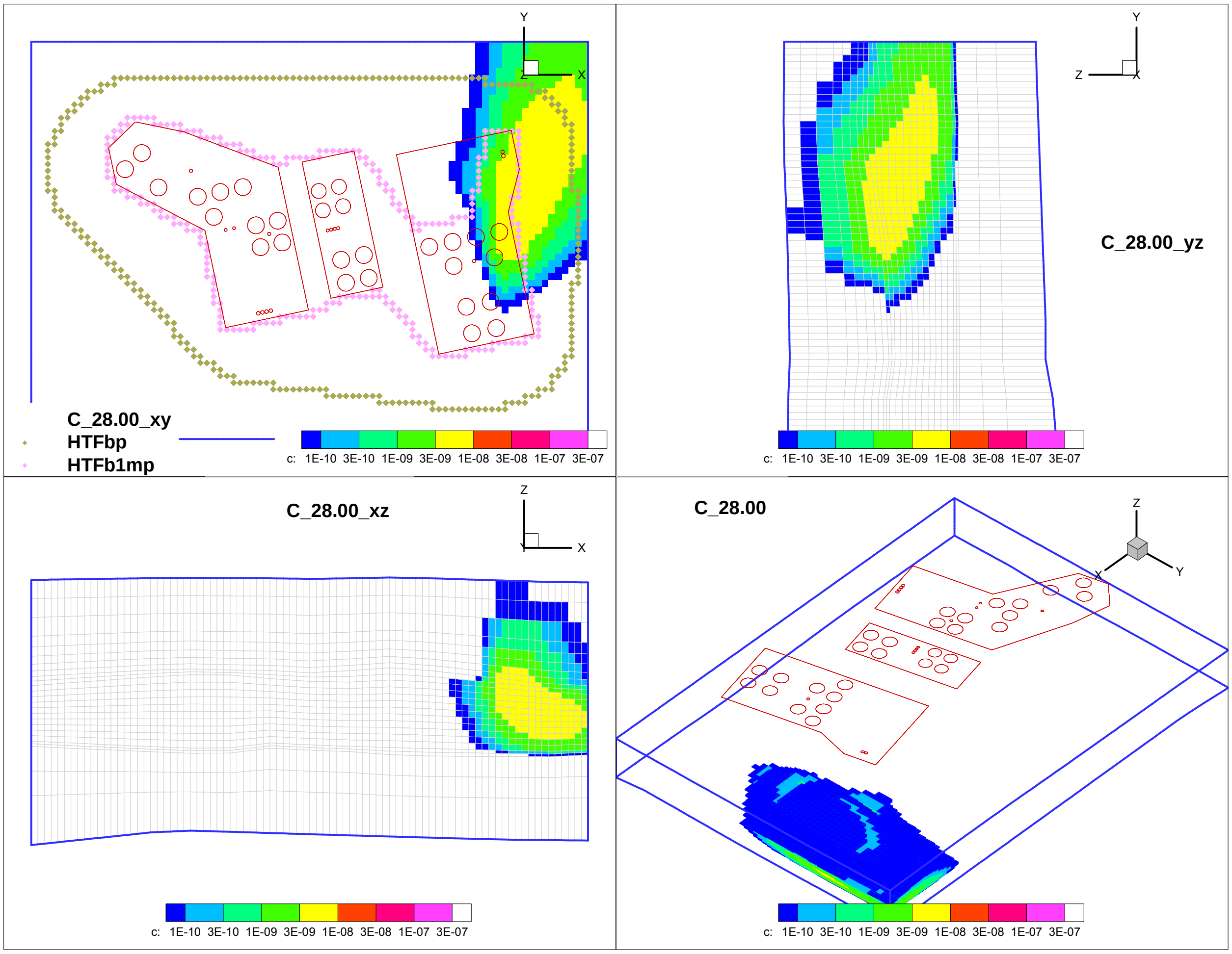


Figure CC-FF-1.7: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 48

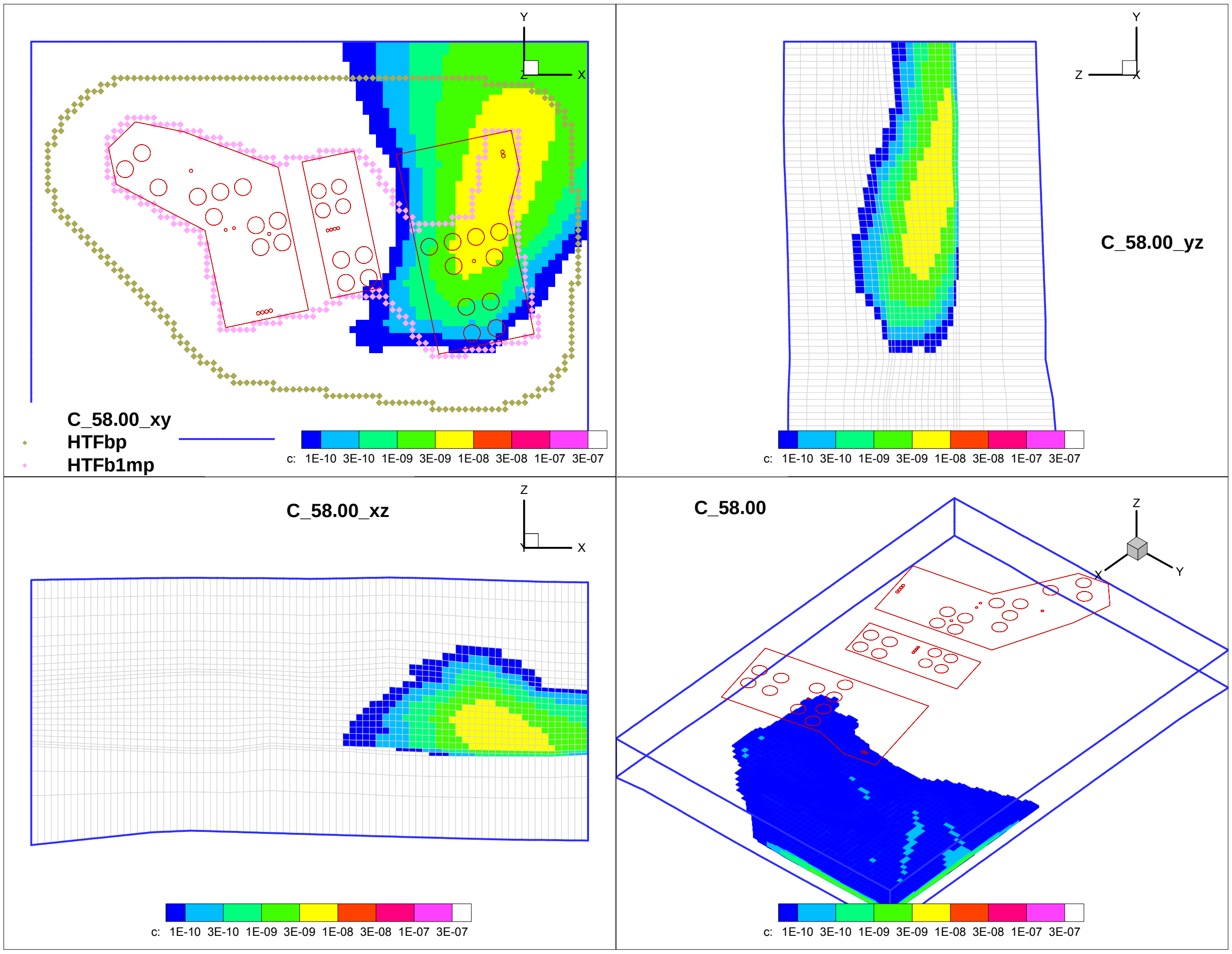


Figure CC-FF-1.8: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 49

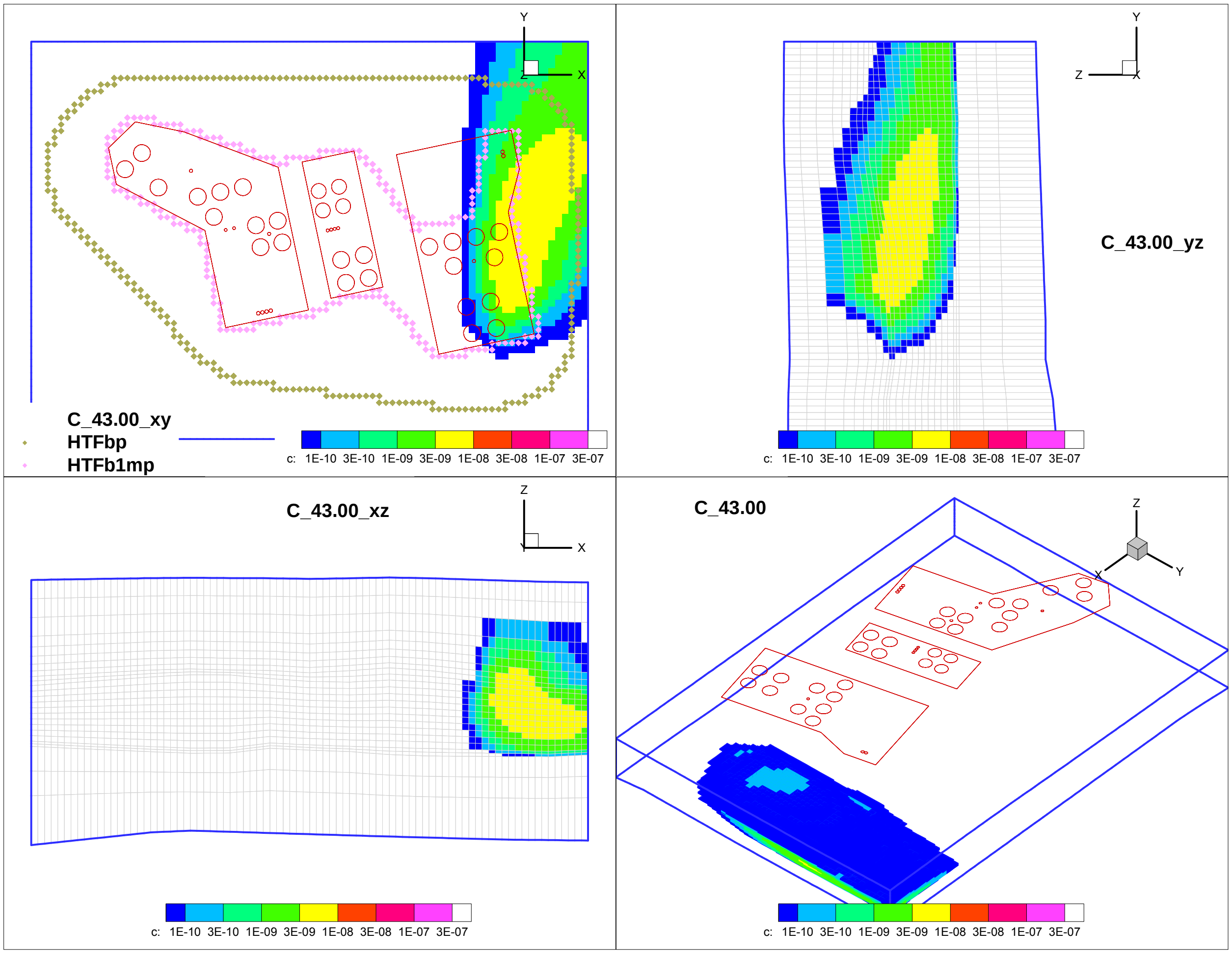


Figure CC-FF-1.9: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 50

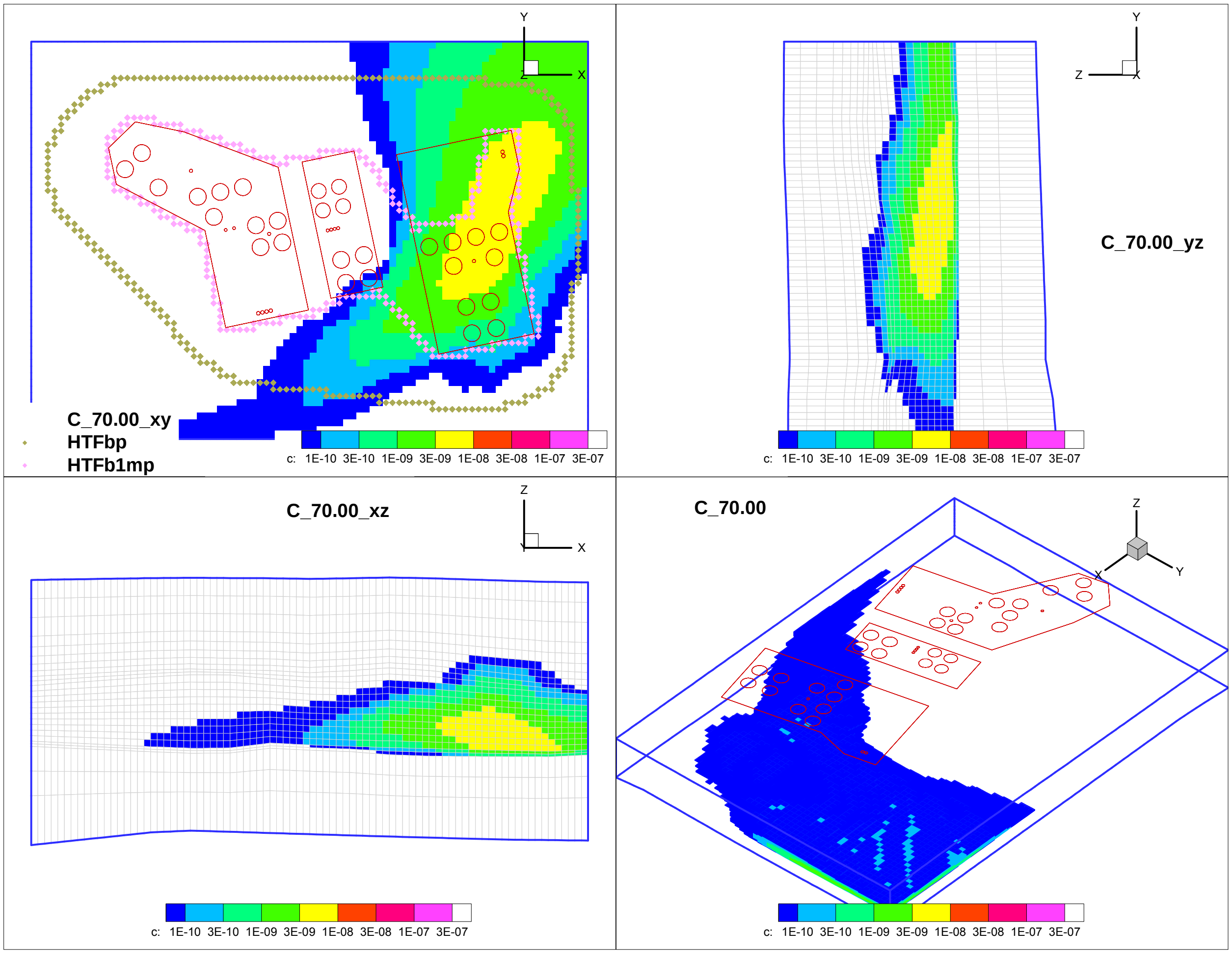
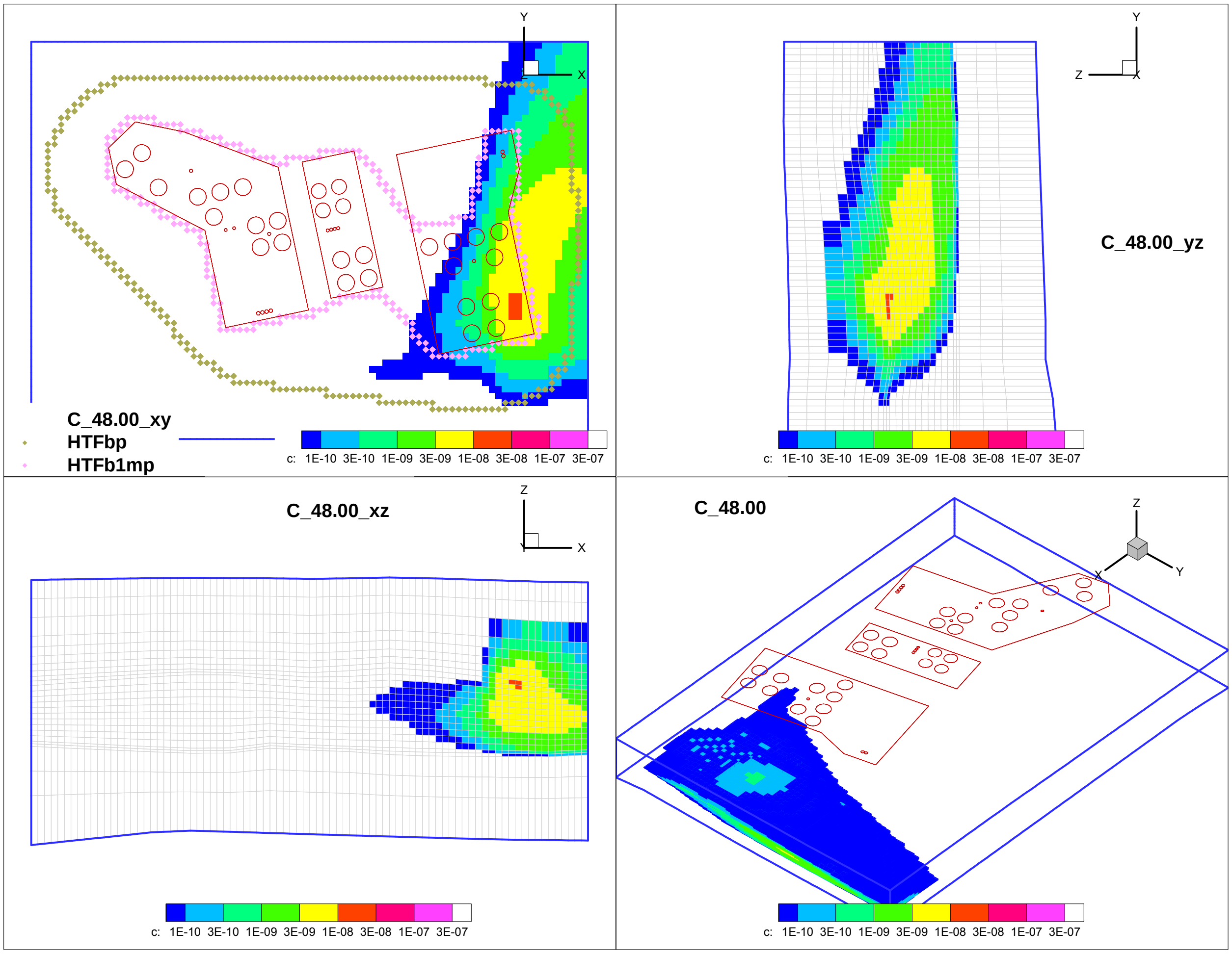


Figure CC-FF-1.10: Concentration Plume (mol/L) Formed by the Release of a Conservative Constituent from Tank 51



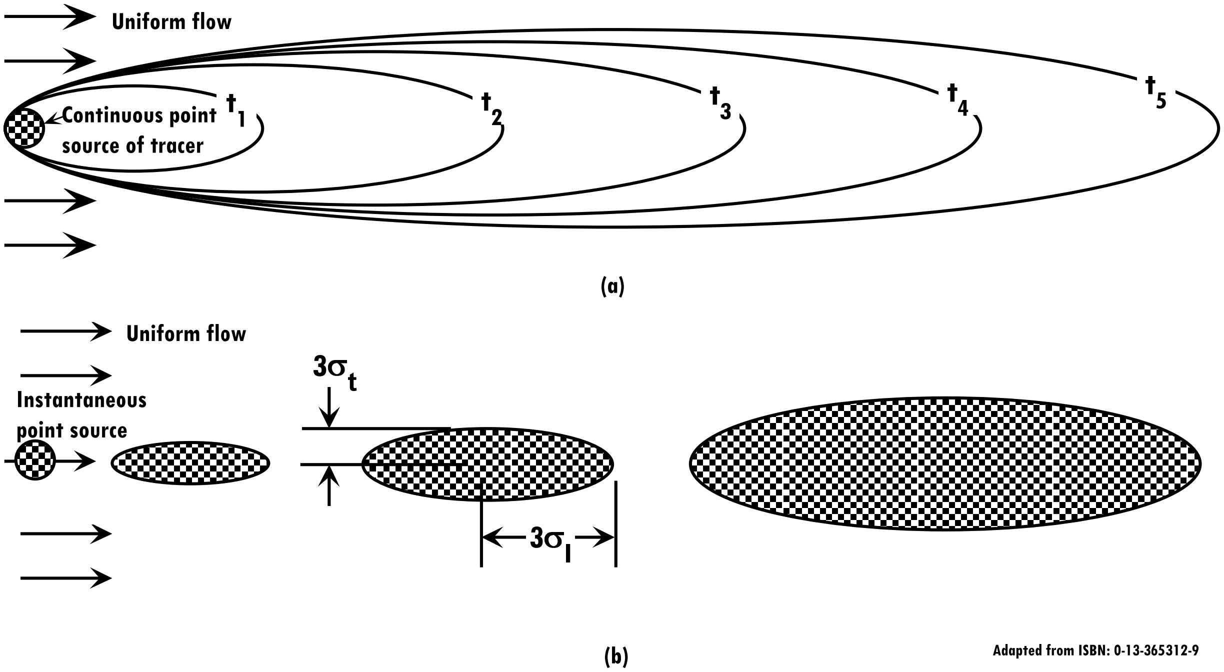
CC-FF-2

Page 60 of SRR-CWDA-2010-00093, Rev. 2 indicates that differences in flow directions were more significant for western sources, leading to the need for higher transverse dispersivities for western sources. Clarify the degree of transverse spreading for various sources at HTF and how changes in transverse dispersivity in GoldSim® probabilistic modeling are used to simulate the effect of changing flow directions.

Response CC-FF-2

When carrying out HTF modeling, it is difficult to match the degree of plume spreading and associated increase in dilution generated by the fully, 3-D PORFLOW model, with the quasi-3-D GoldSim abstraction of the PORFLOW model. On the western side of the HTF, a groundwater divide, spatially divergent flow, and changes in flow direction enhance the lateral spreading of the plumes generated by waste tank releases. The PORFLOW model solves the advective-dispersive transport equations in a fully, 3-D framework, and thus considers both lateral dispersion and the divergence of the flow field in its calculations. The GoldSim model is a quasi-3-D abstraction of the PORFLOW model that solves for 1-D advective-dispersive transport along a pathline, and approximates the influences of horizontal and vertical transverse dispersion with Green's function solutions. In this configuration all the effects of the flow field are not fully captured. Confined to evaluating advective-dispersive transport in a unidirectional flow field, the GoldSim model generates typical cigar shaped plumes for steady and instantaneous releases (Figure CC-FF-2.1 (a) and (b)).

Figure CC-FF-2.1: Plume Generation in Uniform Flow Fields for (a) Steady and (b) Instantaneous Sources



Adapted from ISBN: 0-13-365312-9

The waste tanks located on the western side of the HTF include the Type I tanks (9, 10, 11, and 12), the Type II tanks (13, 14, 15, and 16), the Type IV tanks (21, 22, 23, and 24), and some of the Type III and IIIA tanks (29, 30, 31, 32, 35, 36, and 37). Releases from these waste tanks tend to form plumes that are wider than would be expected for a system with a longitudinal to horizontal-transverse dispersivity ratio of 10:1 as used in the PORFLOW model. The influence of the spatially variant aspects of the flow field on plume spreading can be observed in waste tank-specific plume plots. These PORFLOW generated plots (see Figures CC-FF-2.2 through CC-FF-2.20) show plumes formed by a pulse release of a conservative constituent (i.e., constituent that is not sorbed, is not solubility controlled, and does not decay). The GoldSim pipe element, which assumes a steady unidirectional flow field, used in conjunction with the plume function will generate a cigar shaped plume as depicted in Figure CC-FF-2.1 (a) and (b) depending on whether the source is steady (a) or instantaneous (b). For each source release, the flow-fields increase spreading of the plumes (as modeled in PORFLOW), which decreases the radionuclide concentrations (and associated dose contributions) at a point 100 meters from the HTF boundary.

Figure CC-FF-2.2: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 9

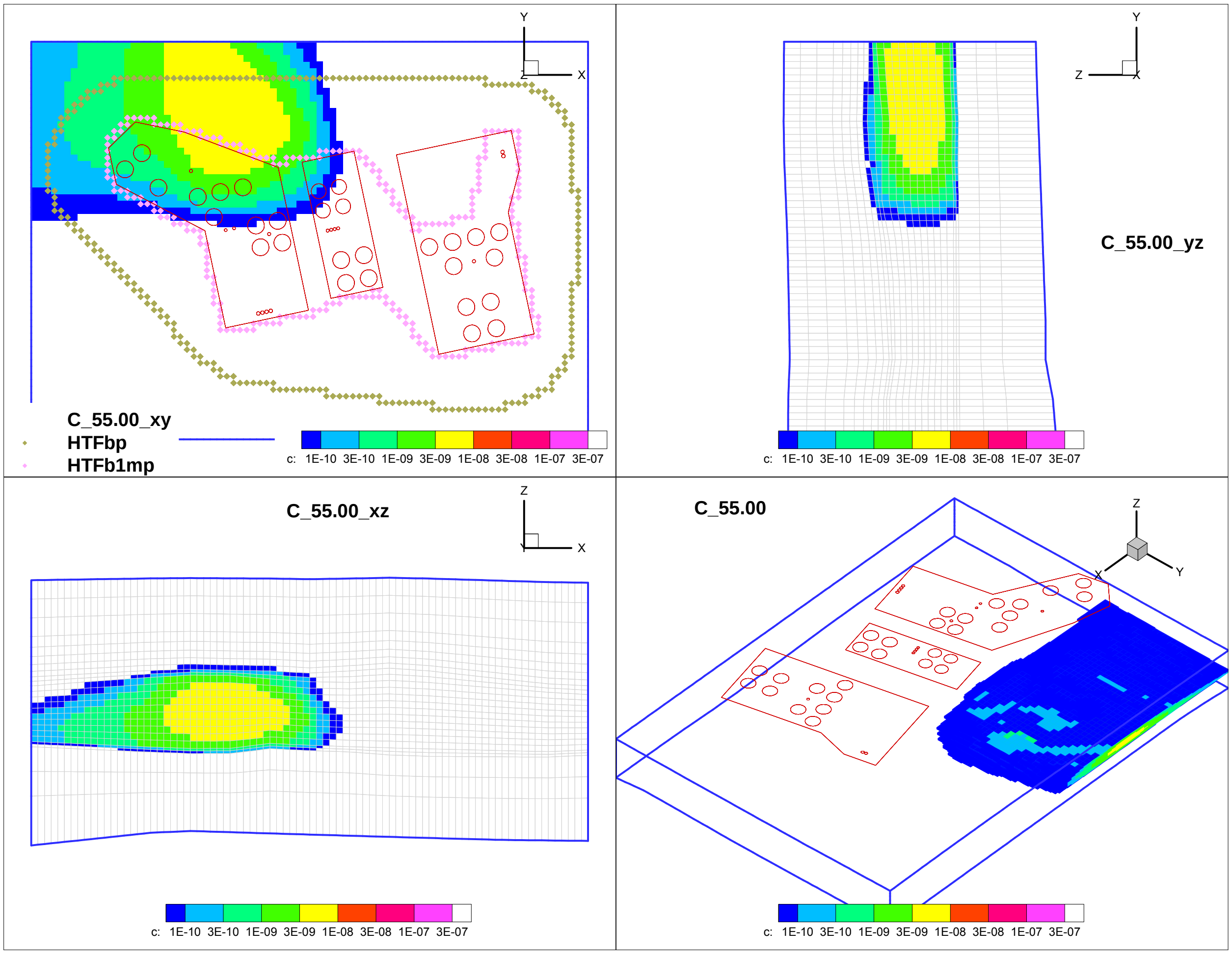


Figure CC-FF-2.3: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 10

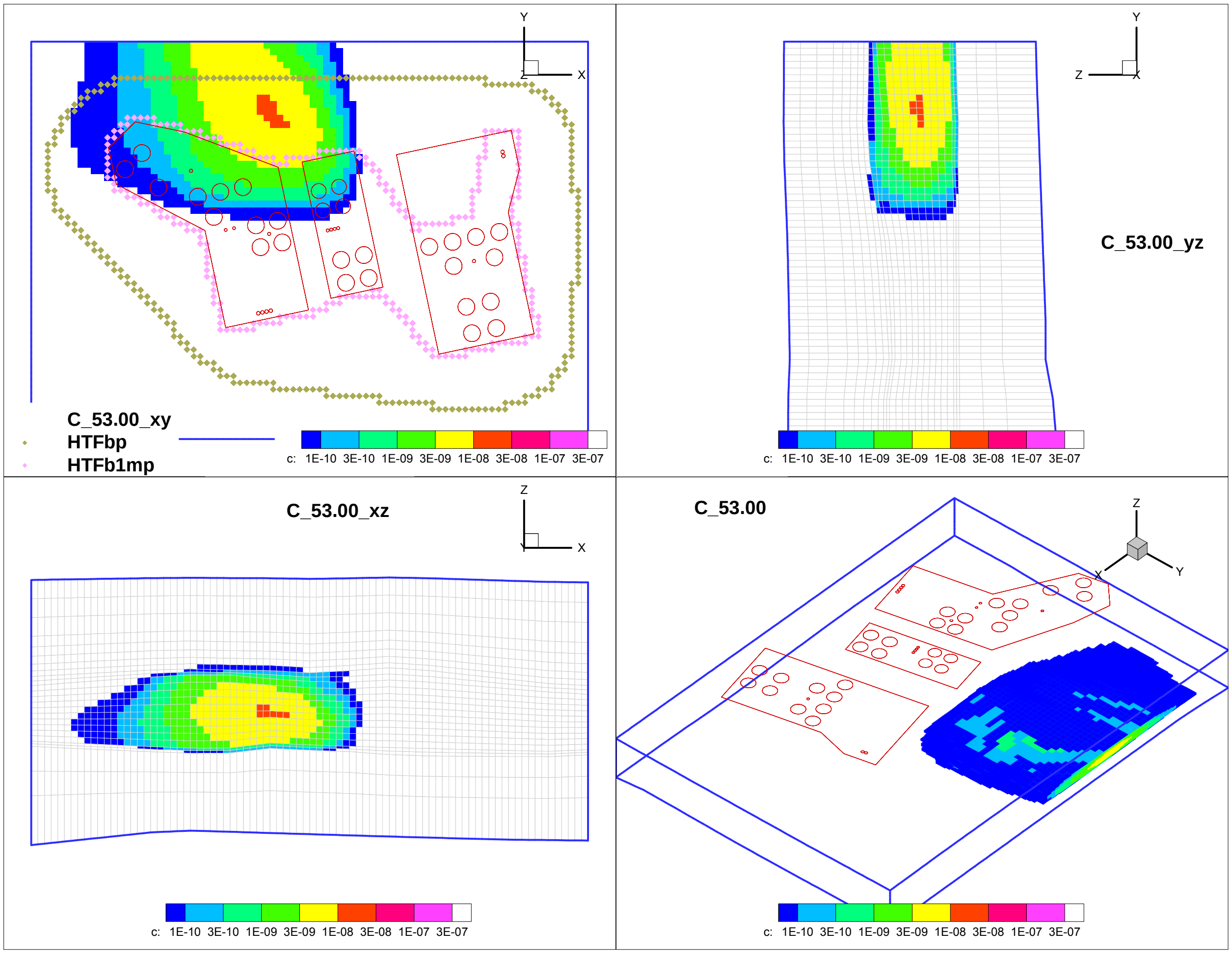


Figure CC-FF-2.4: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 11

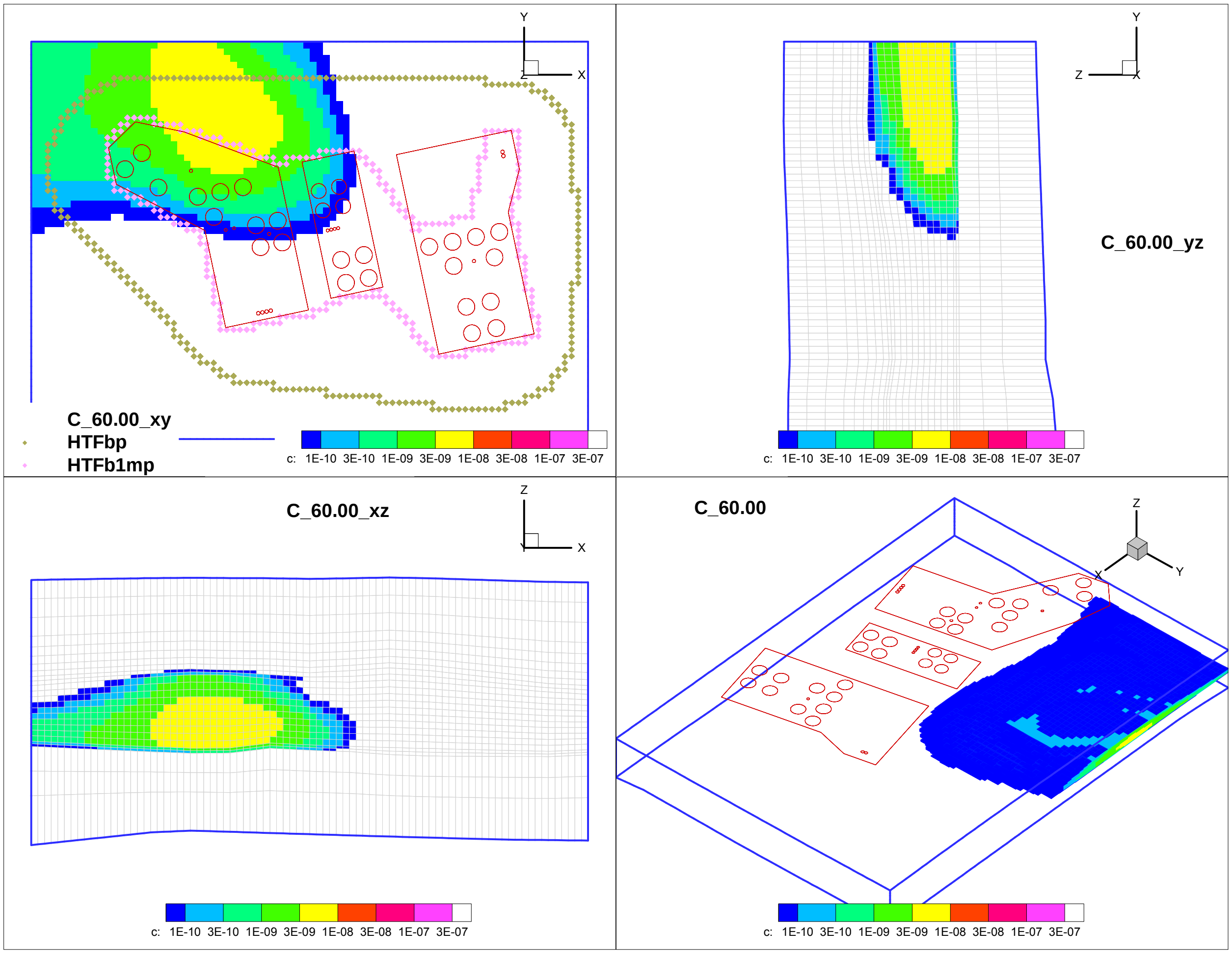


Figure CC-FF-2.5: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 12

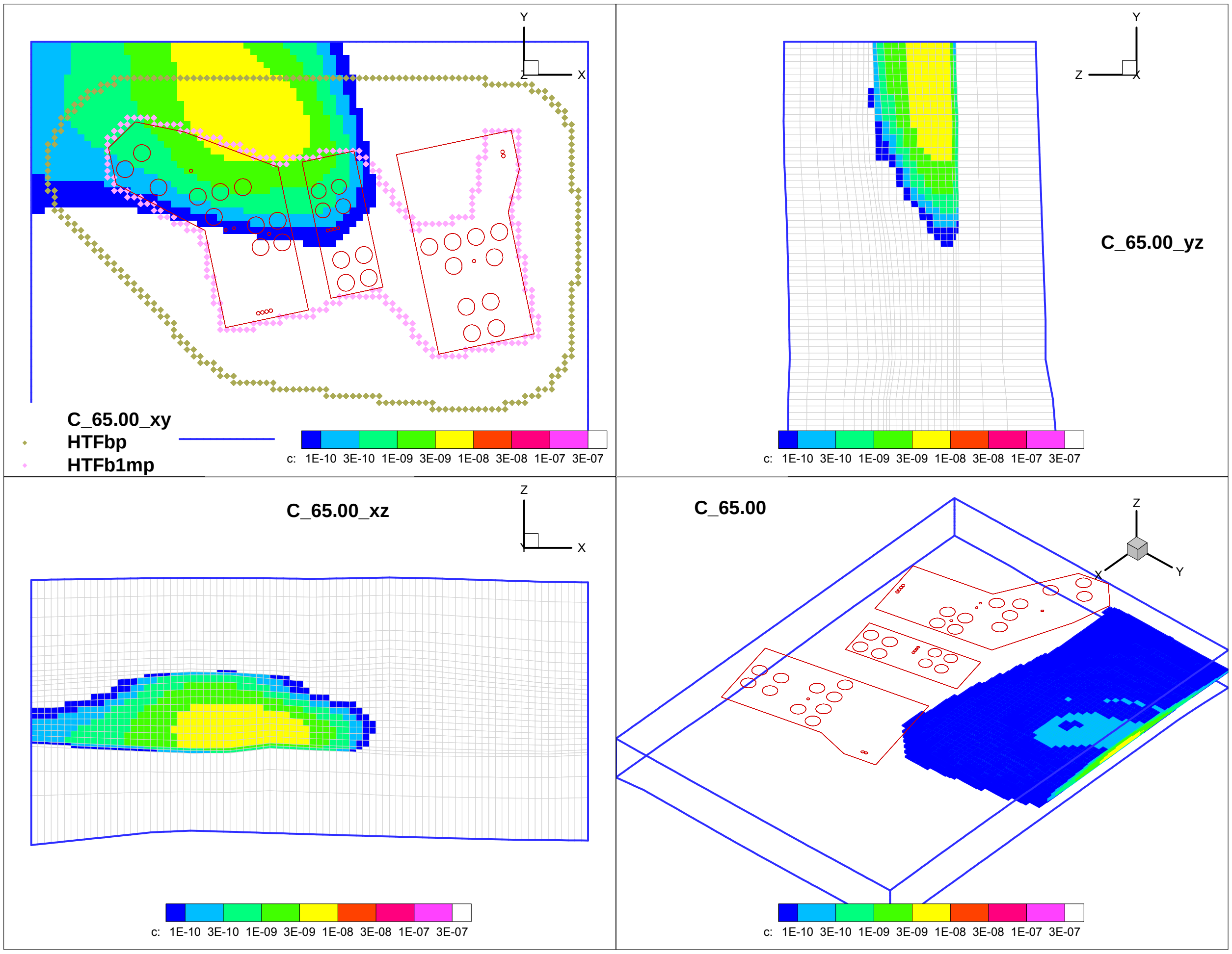


Figure CC-FF-2.6: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 13

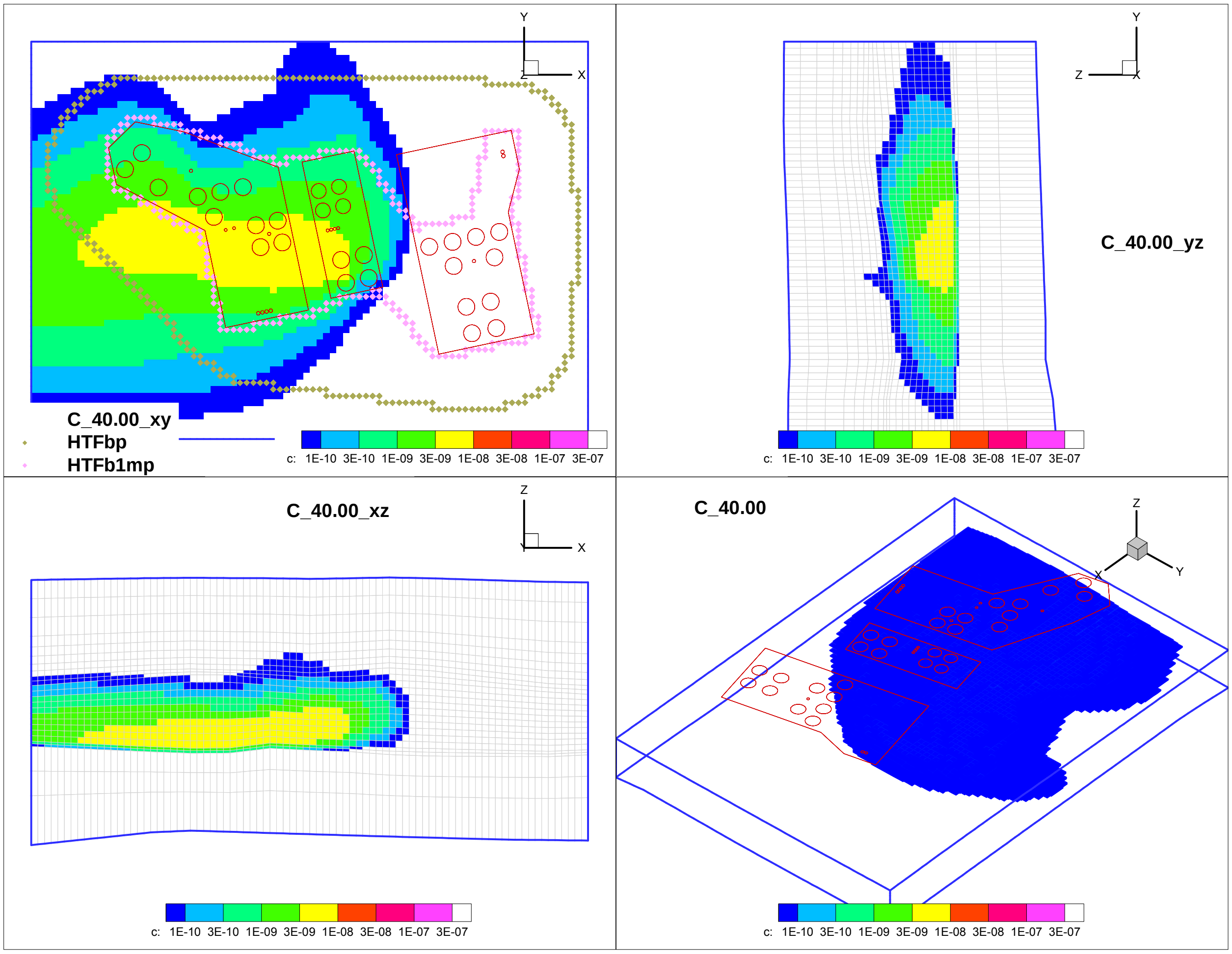


Figure CC-FF-2.7: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 14

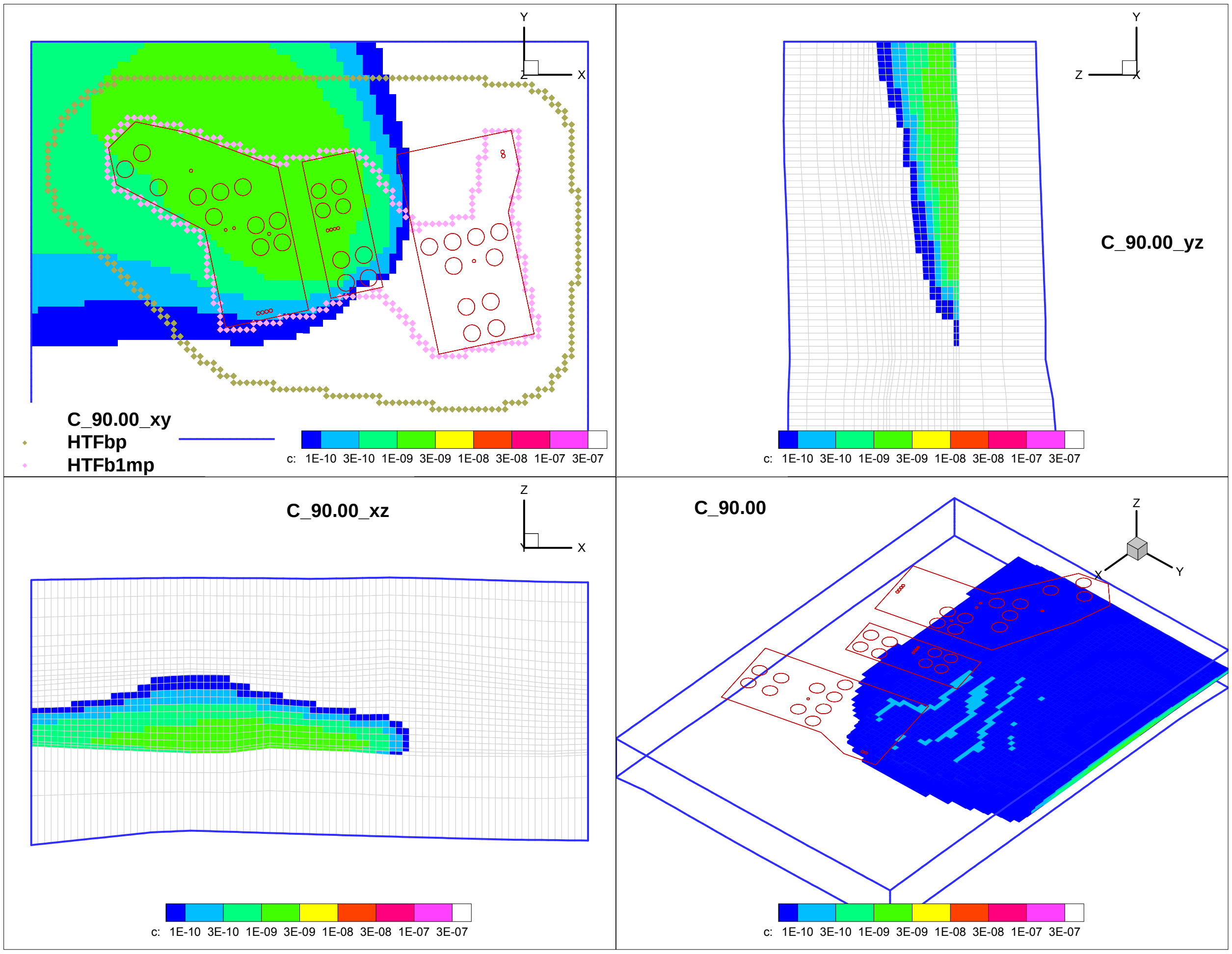


Figure CC-FF-2.8: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 15

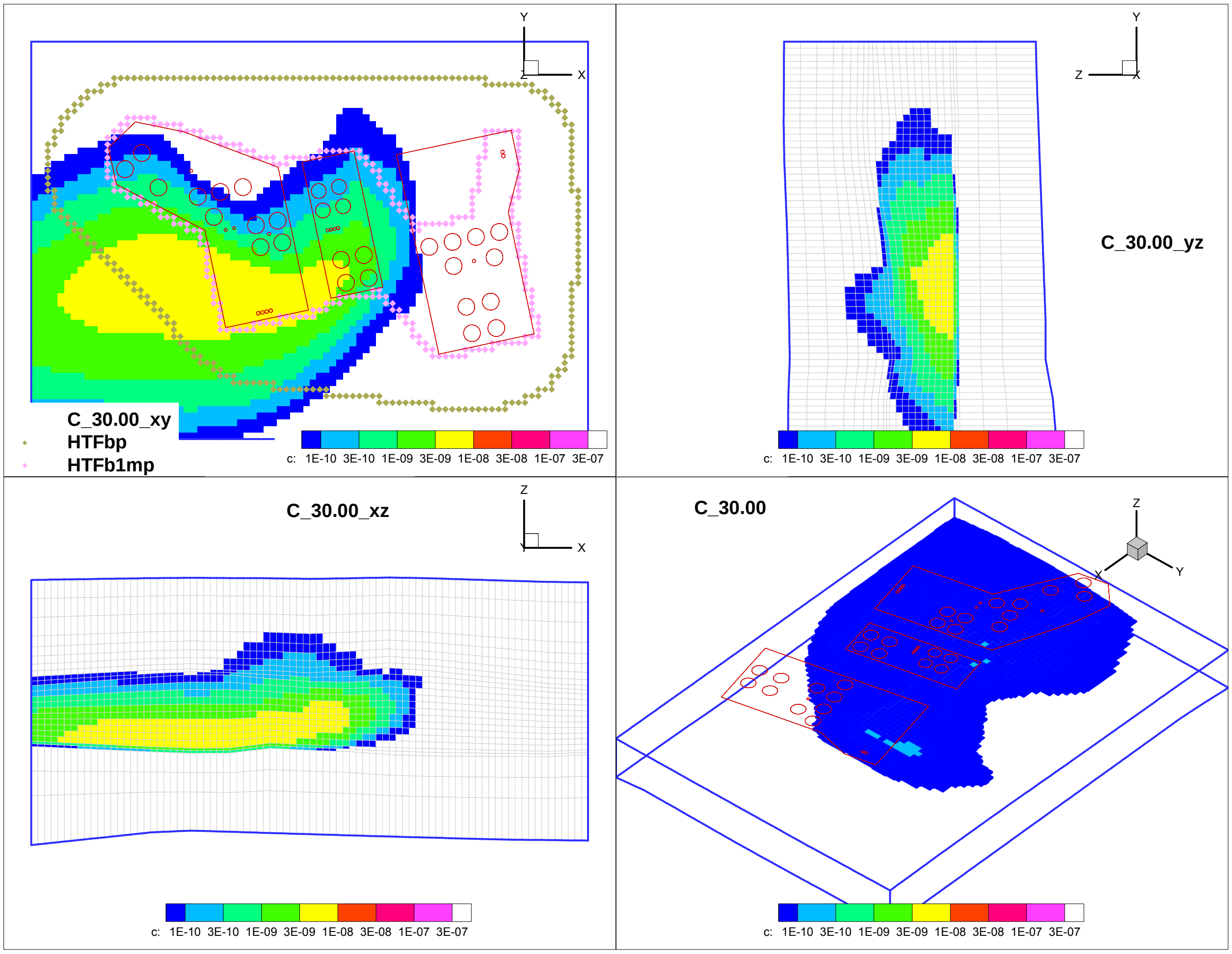


Figure CC-FF-2.9: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 16

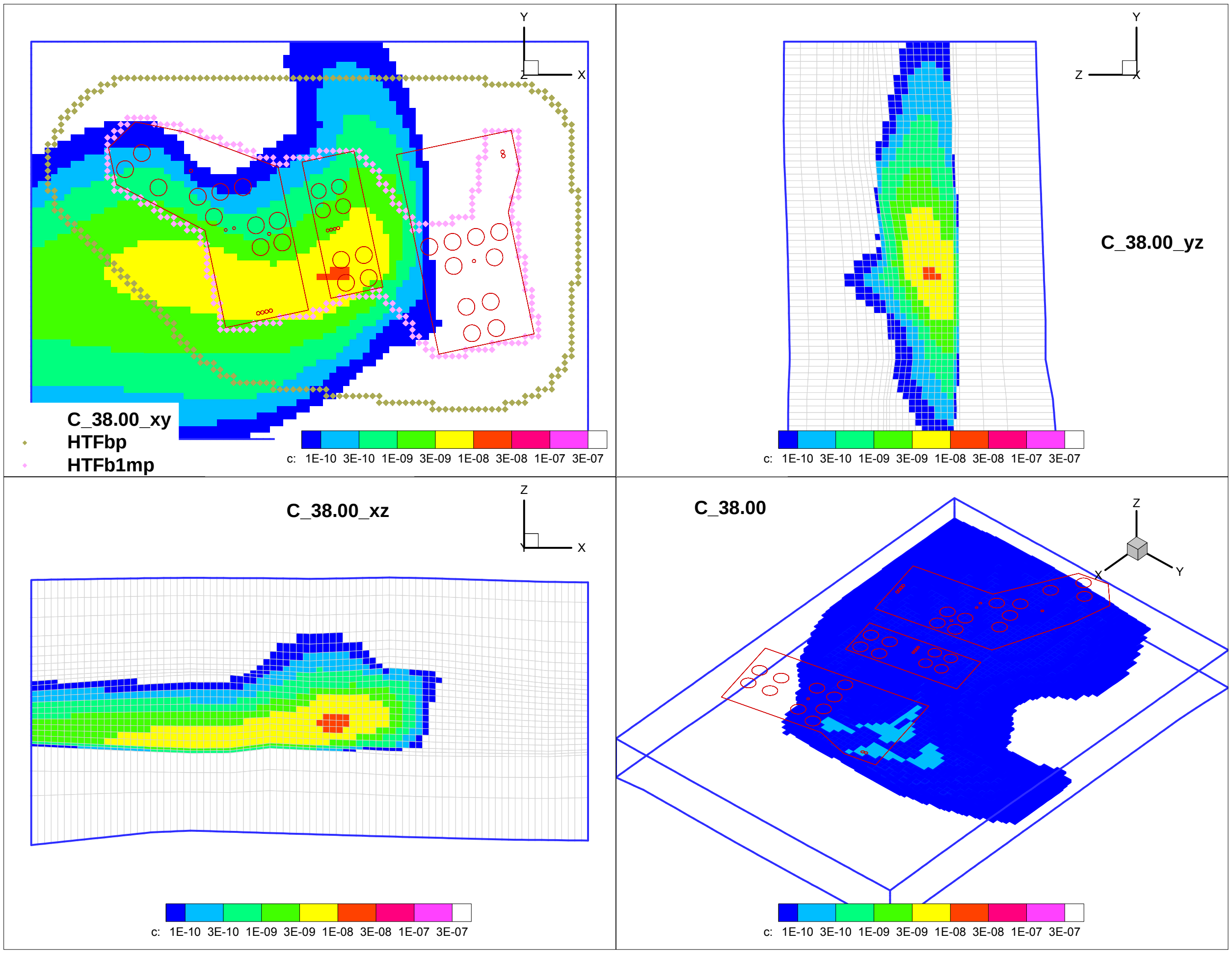


Figure CC-FF-2.10: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 21

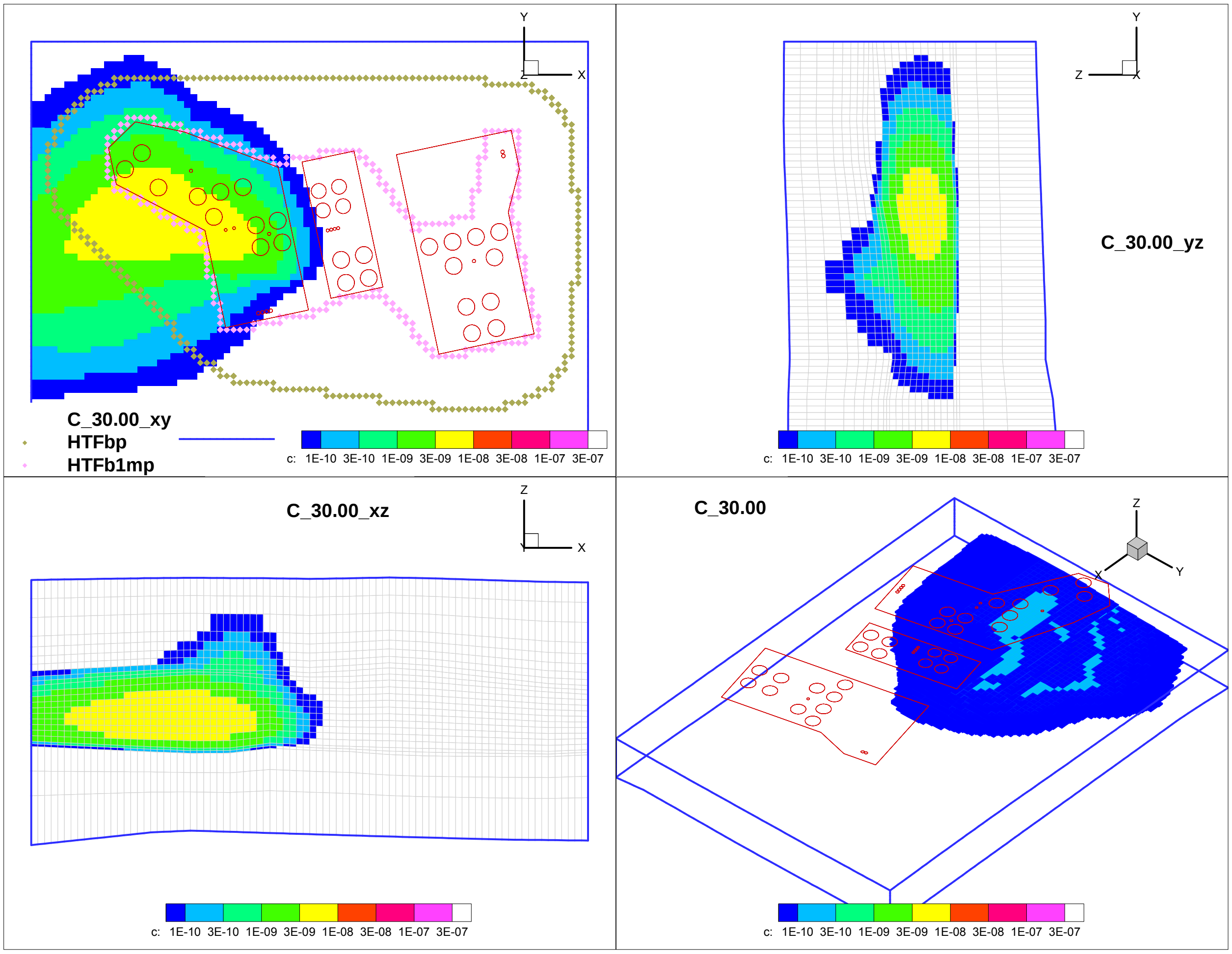


Figure CC-FF-2.11: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 22

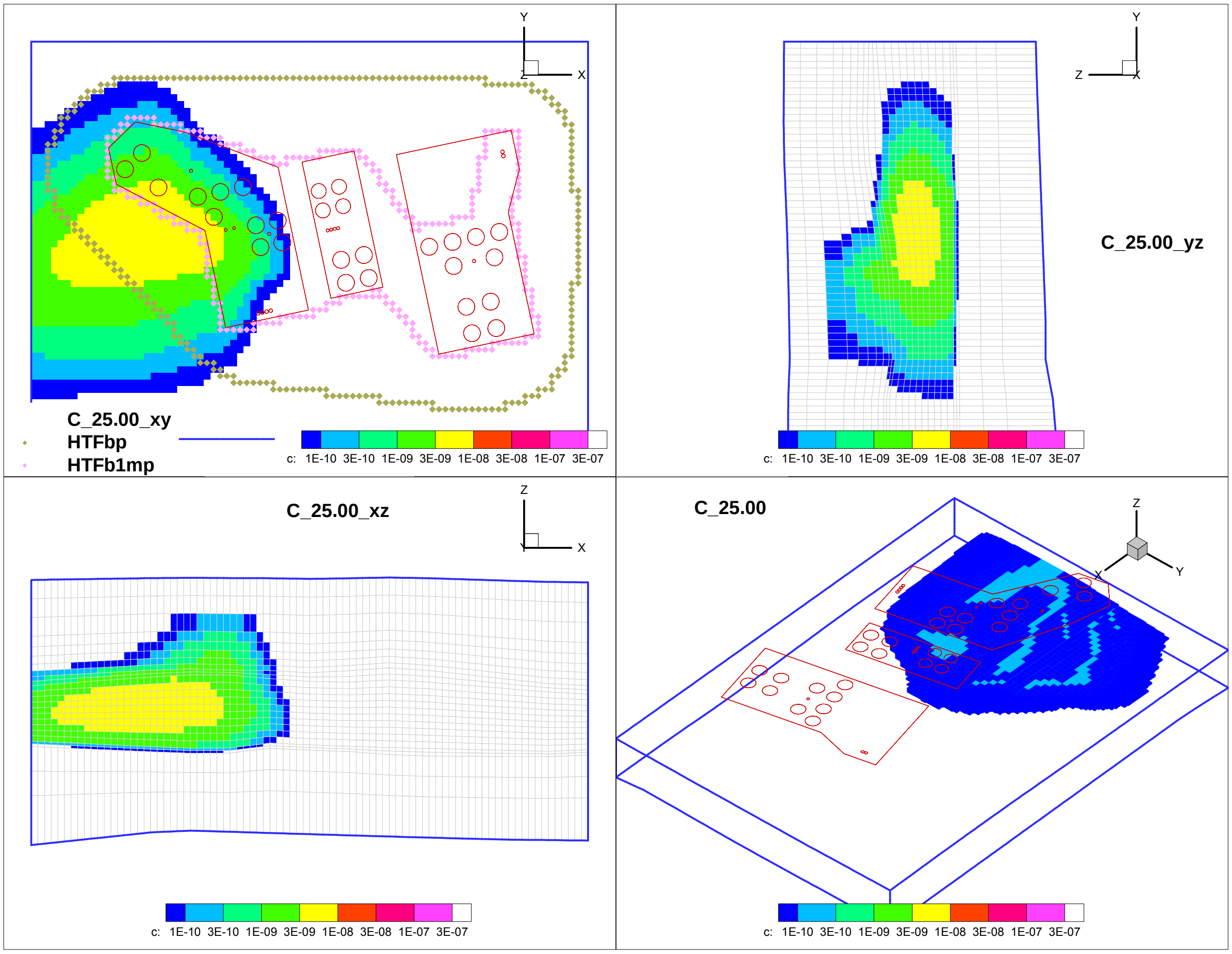


Figure CC-FF-2.12: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 23

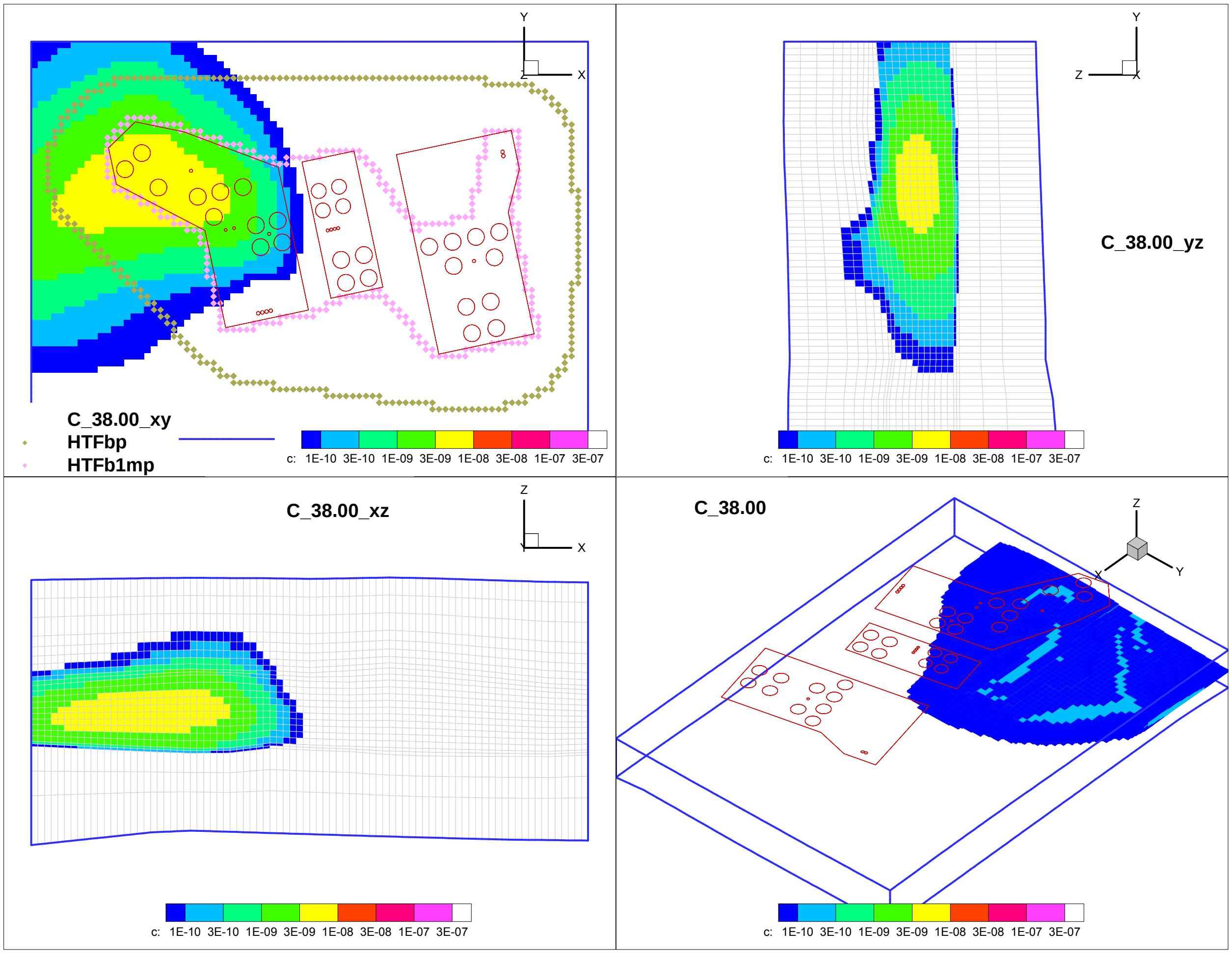


Figure CC-FF-2.13: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 24

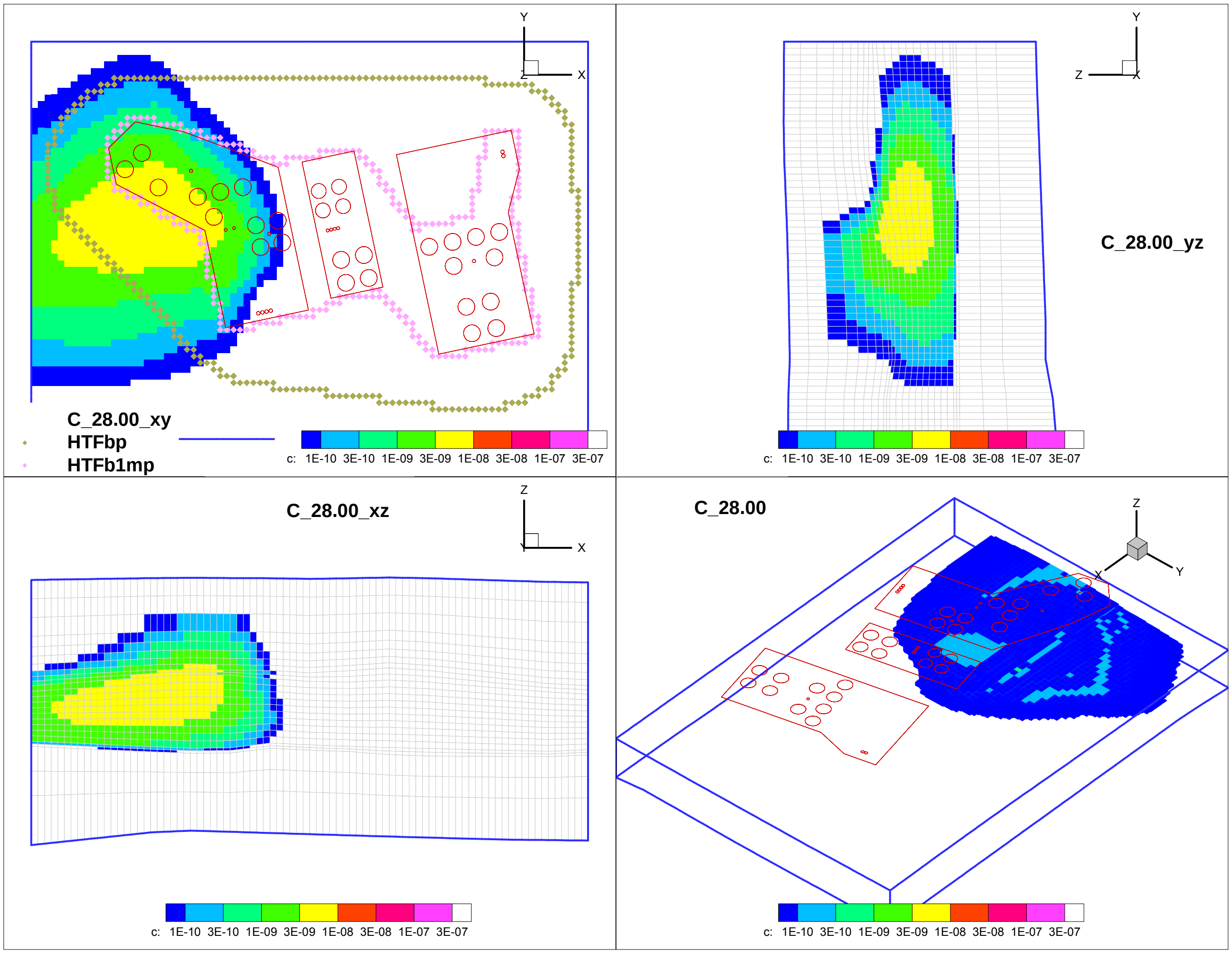


Figure CC-FF-2.14: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 29

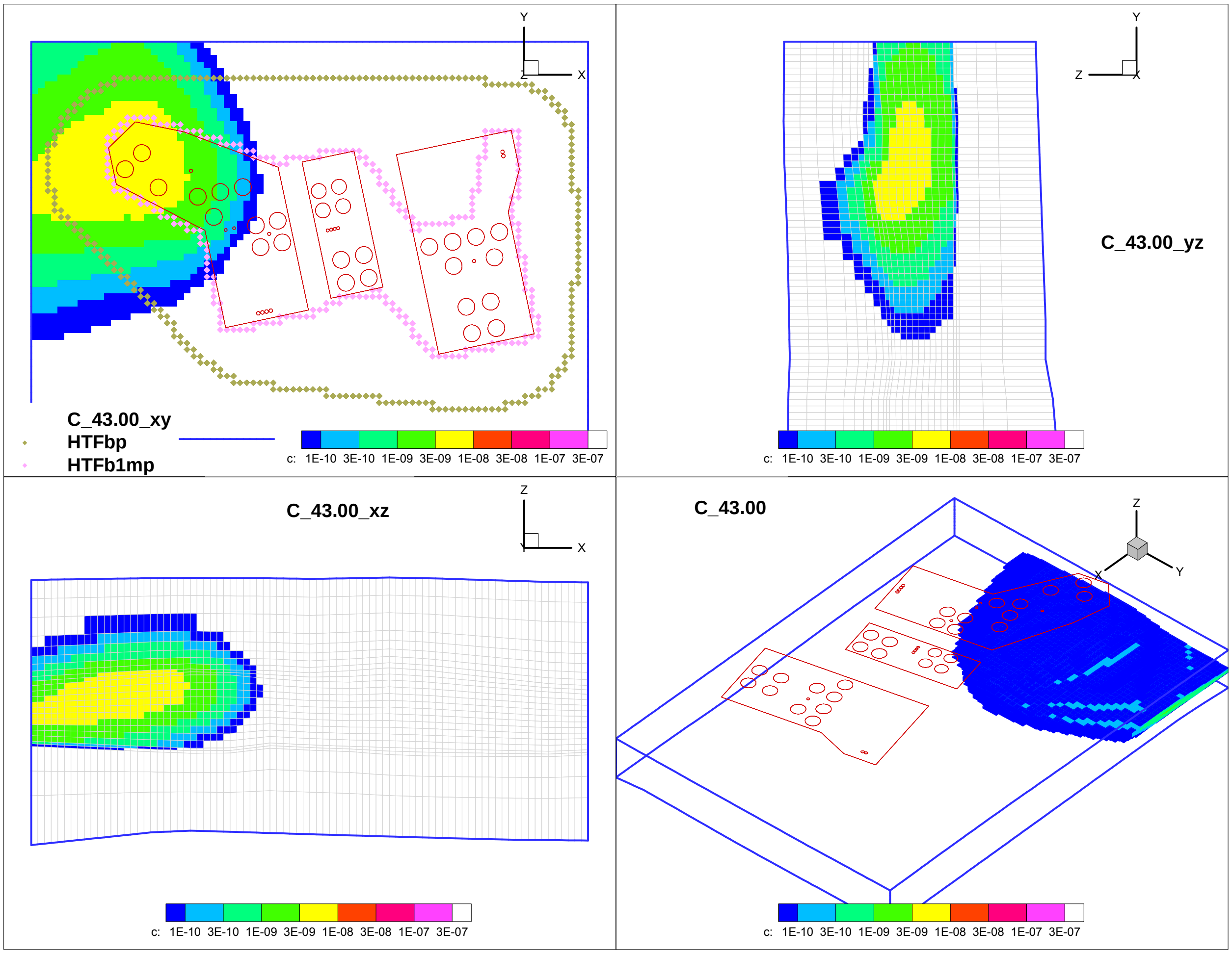


Figure CC-FF-2.15: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 30

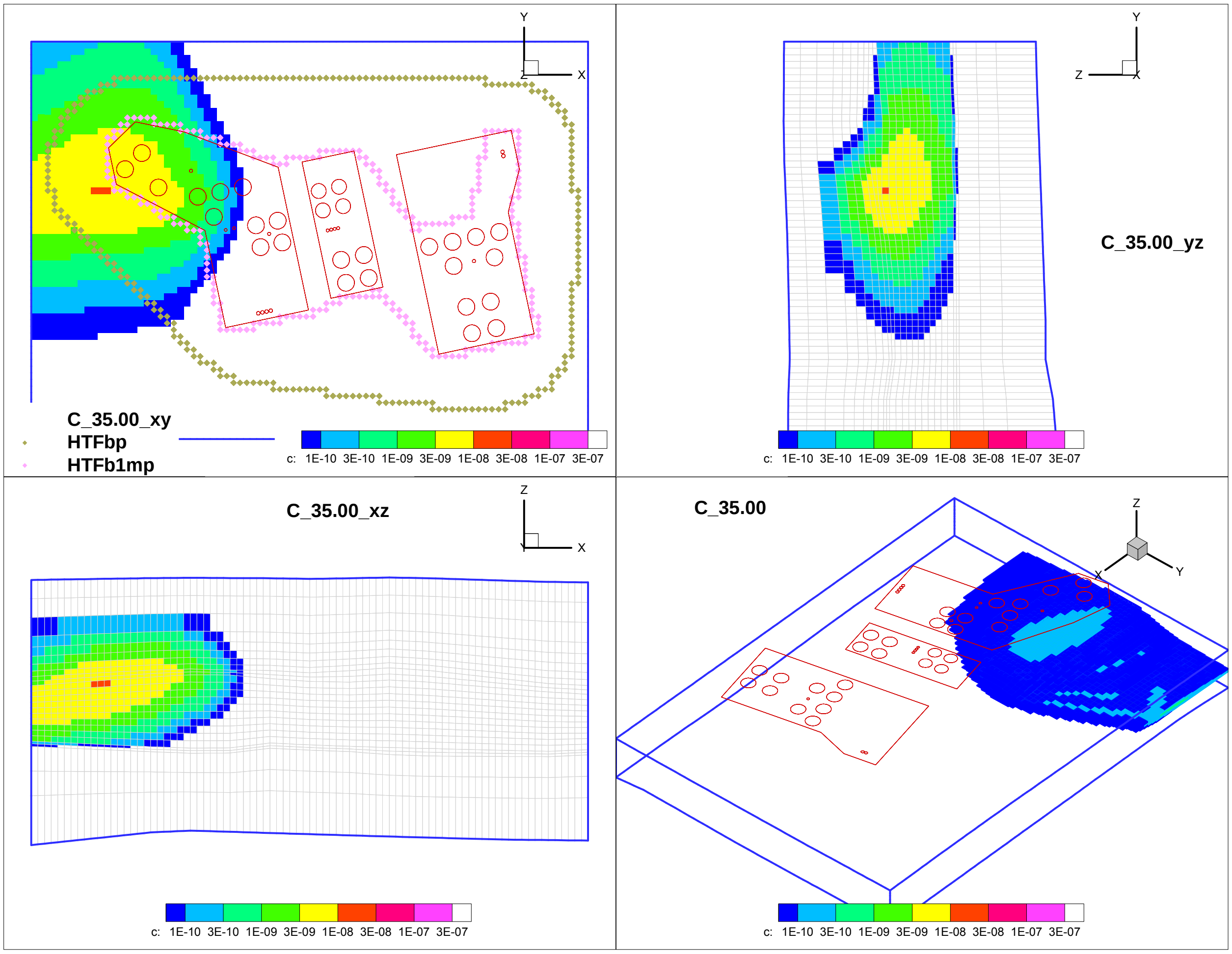


Figure CC-FF-2.16: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 31

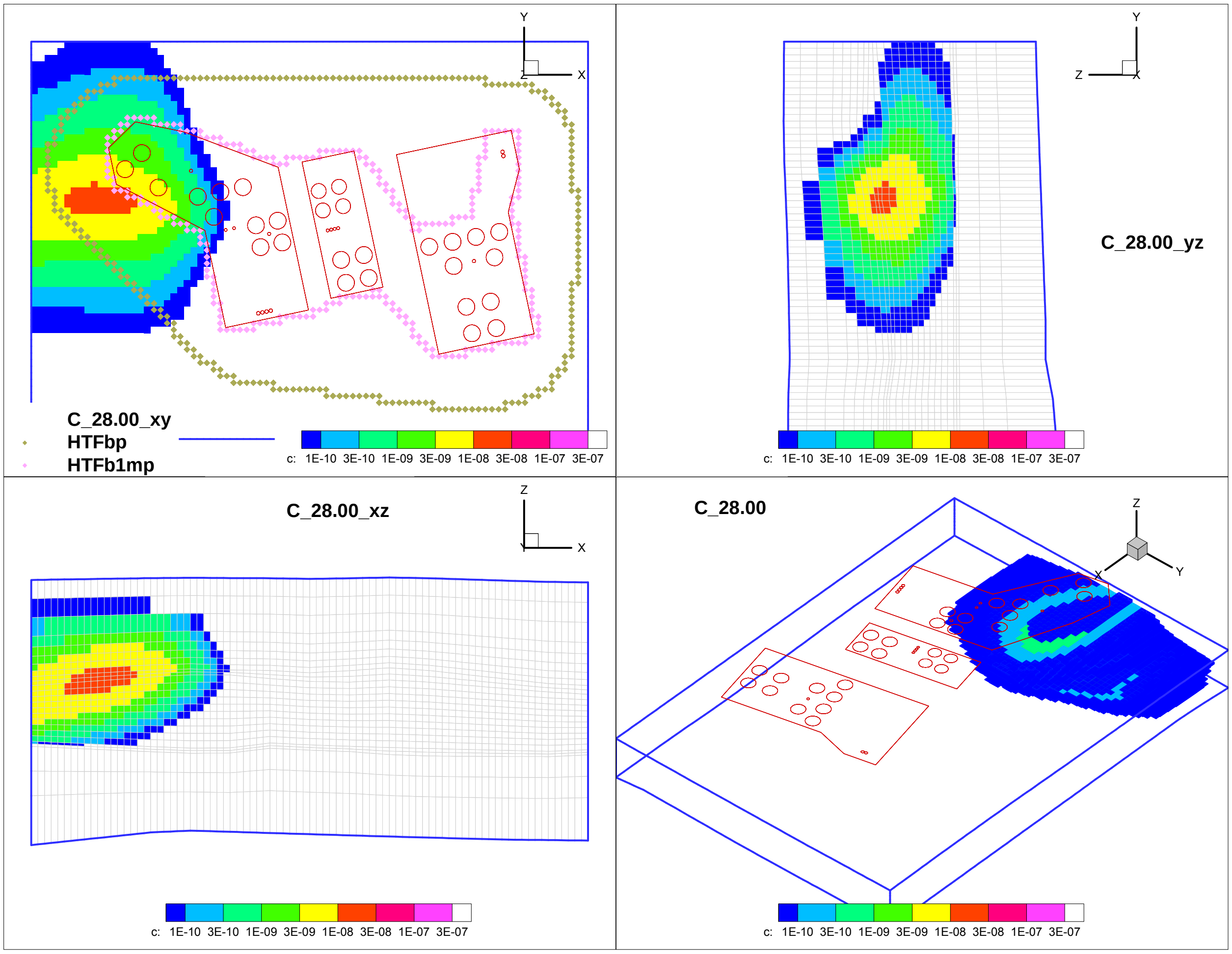


Figure CC-FF-2.17: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 32

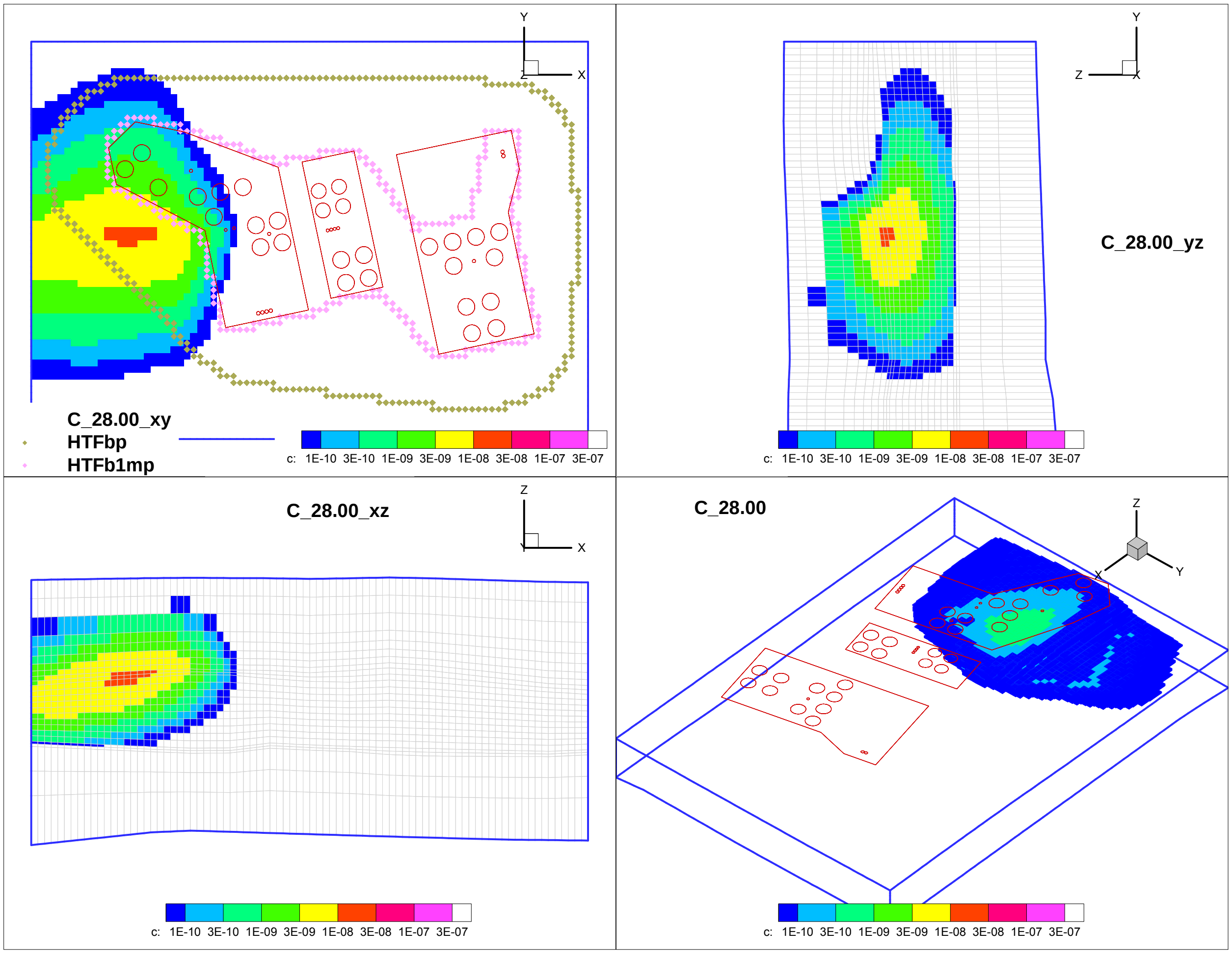


Figure CC-FF-2.18: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 35

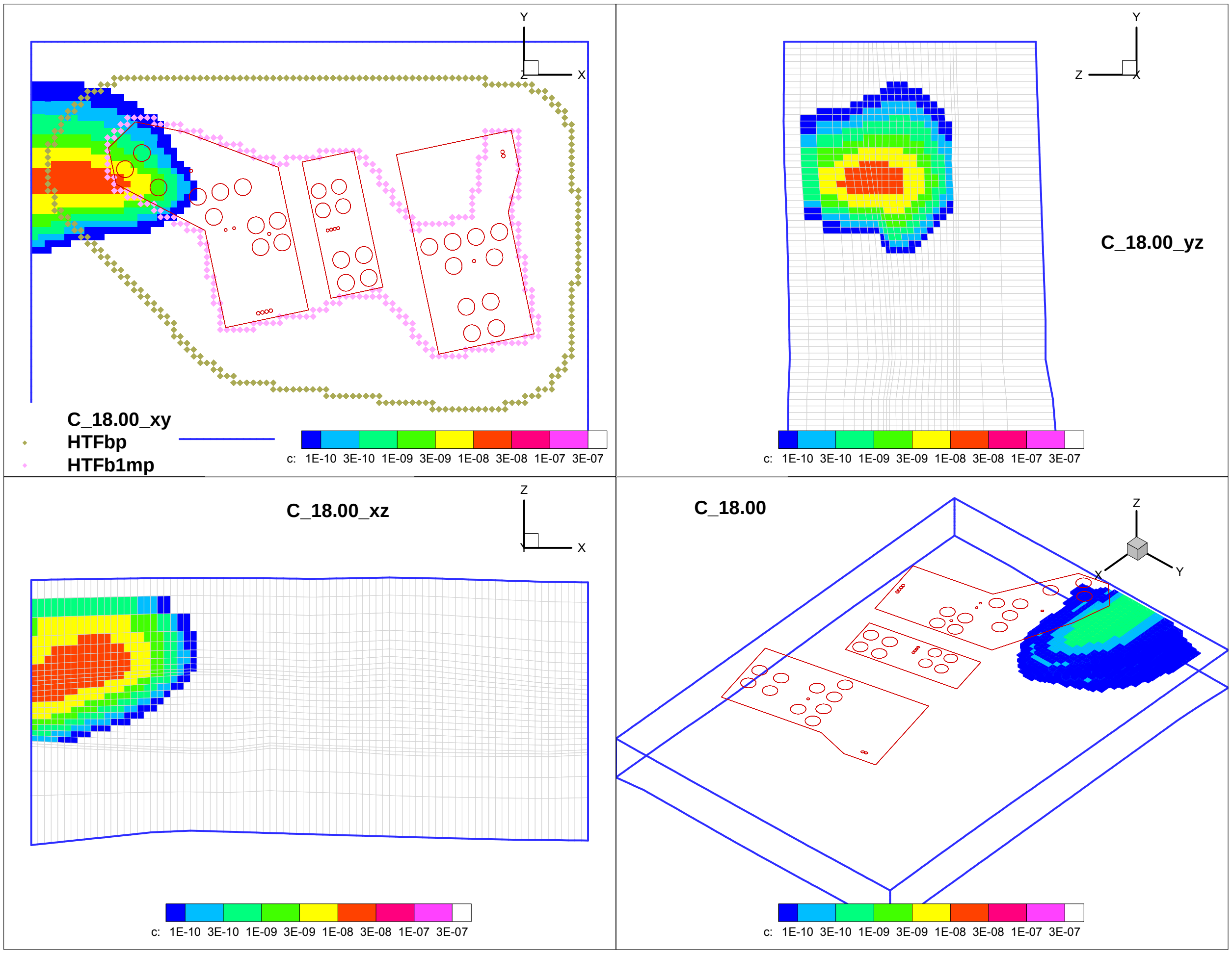


Figure CC-FF-2.19: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 36

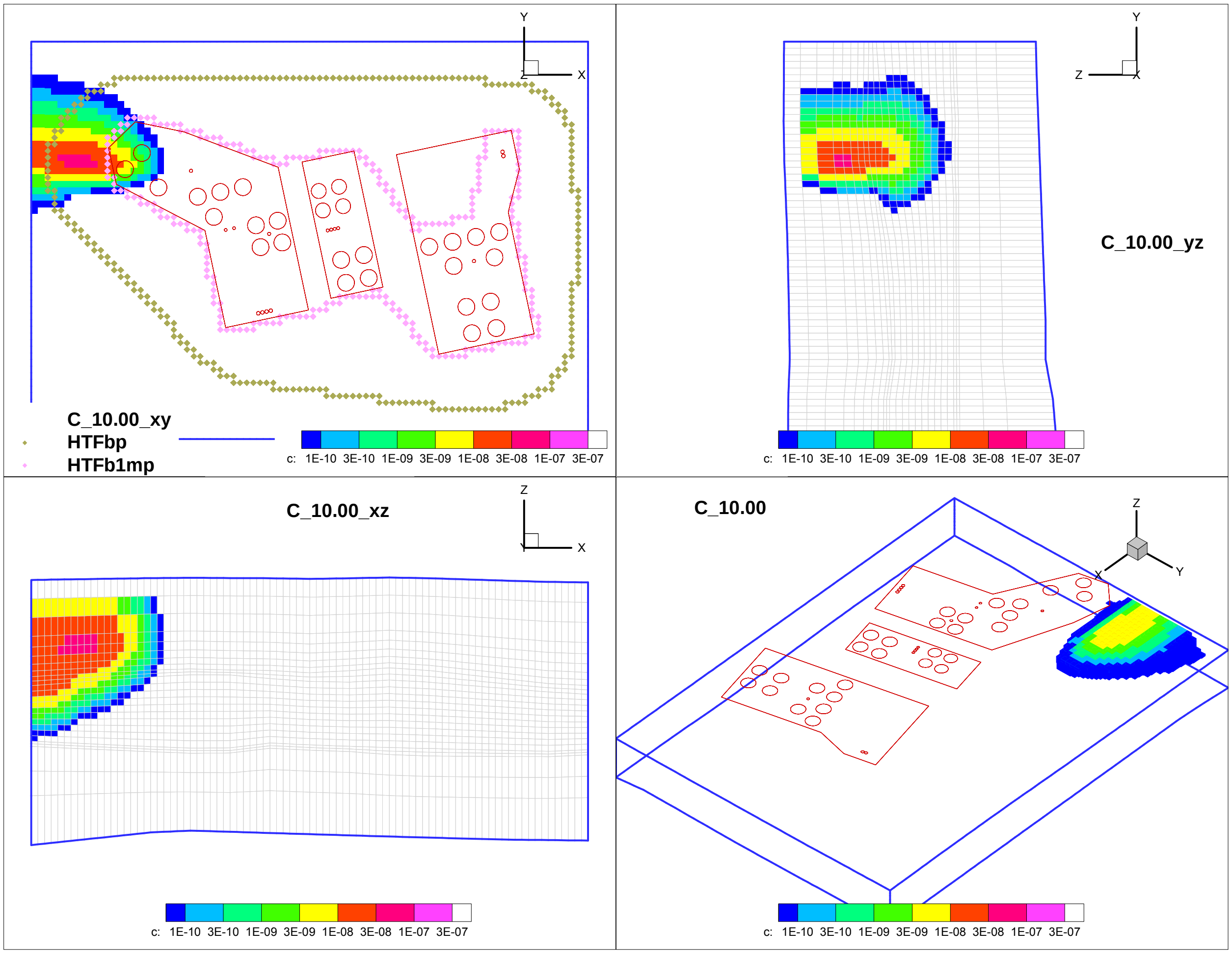
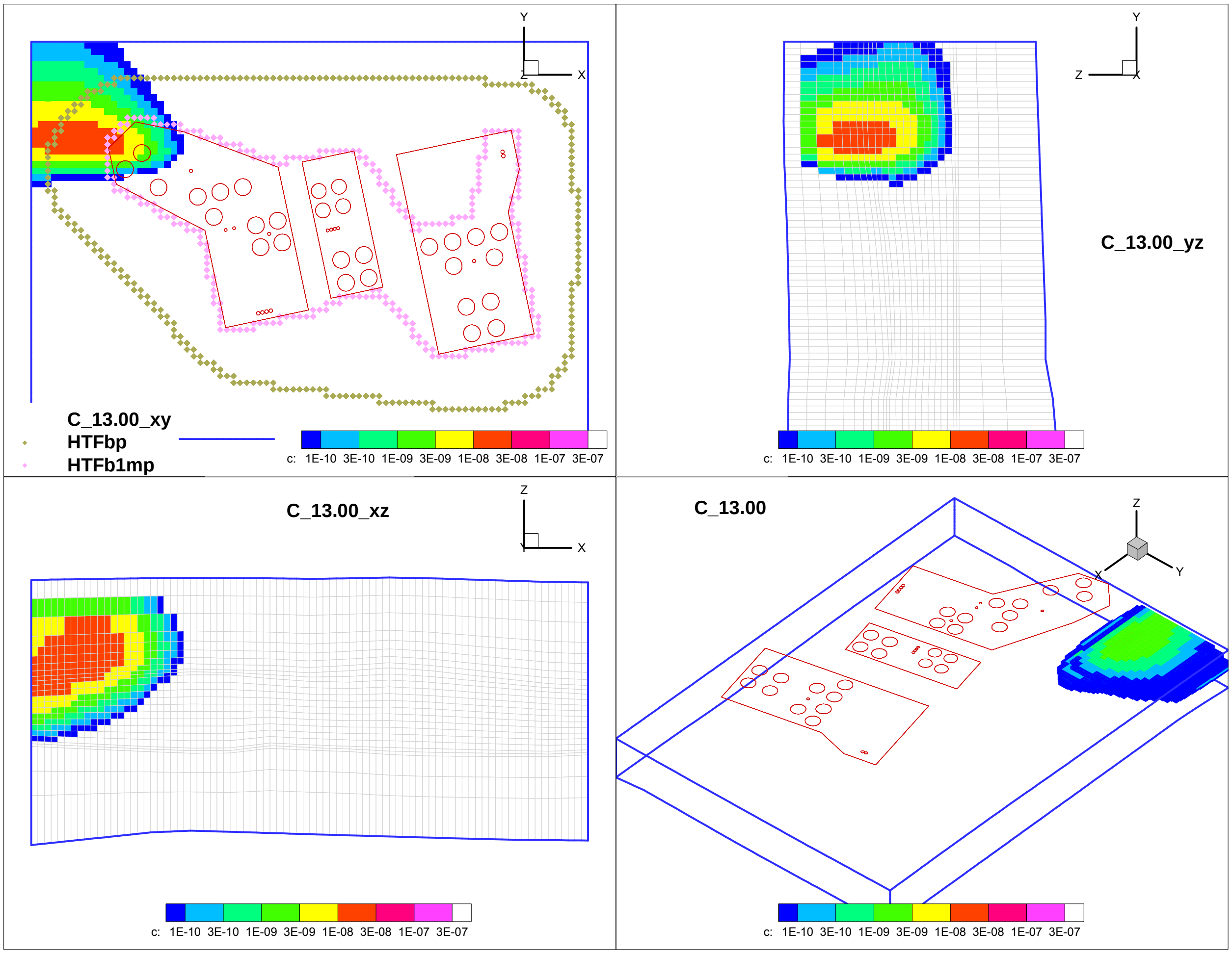
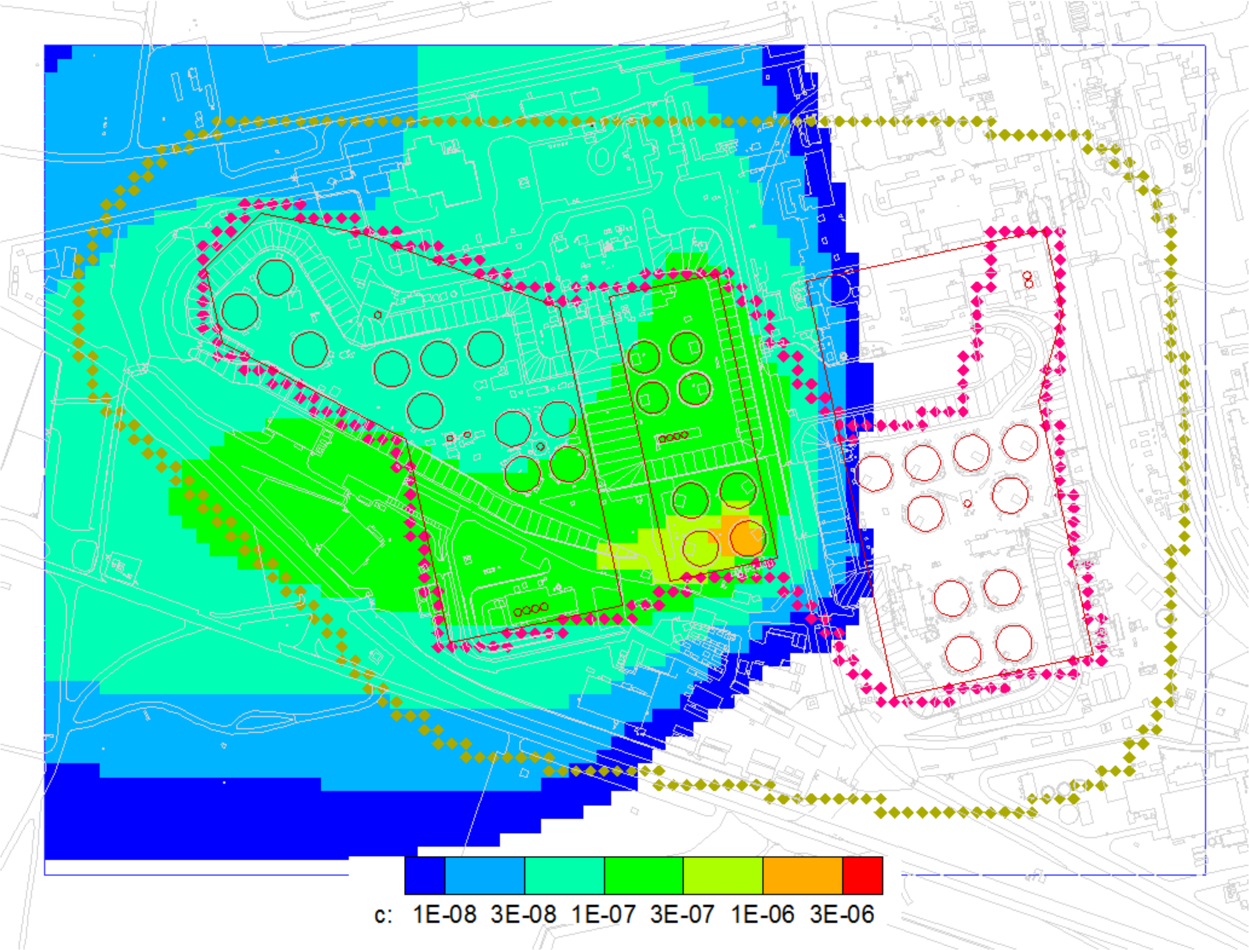


Figure CC-FF-2.20: Concentration Plume (mol/L) Formed by the Pulse Release of a Conservative Constituent from Tank 37



In addition, some source release plumes (Figures CC-FF-2.2 through CC-FF-2.20) show two lobes, indicating that part of the mass released does not follow the pathline defined by a particle track initiated at the center of the source (see the response to RAI-FF-3 for the pathlines used for each waste tank). This pattern is seen in instantaneous (pulse) releases from Tanks 13, 15, and 16 as shown in Figures CC-FF-2.6, CC-FF-2.8, and CC-FF-2.9, respectively. The implications of this pattern of release can be seen in the plot of a steady release from Tank 16 as shown Figure CC-FF-2.21. In GoldSim model simulations, this bimodal release of mass will lead to an overestimation of the waste tank-specific dose contribution in the sector transected by the flow pathline and an underestimation of dose contribution to any secondary sectors. Since the plots of the pulse releases with bimodal pathways showed a dominant path of migration, the addition of a secondary source and pathline was not considered in the GoldSim model.

Figure CC-FF-2.21: Concentration Plume (mol/L) Formed by the Steady Release of a Conservative Constituent from Tank 16



To compensate for the increased spreading of the plumes in the western zone of the HTF, the transverse dispersivity was increased by a factor of three from the PORFLOW value from 1.04 feet (0.316 meter) to 3.12 feet (0.948 meter). This increase improved the match between the PORFLOW and GoldSim models in Sectors A through C. Because of the similar scale of the plumes, complexity of the flow fields and the distortion of plume shapes, a single factor was chosen for the western side.

Note that the plumes from Tanks 35, 36, and 37, presented in Figures CC-FF-2.18, CC-FF-2.19, and CC-FF-2.20, respectively, seem like they might not be as influenced by the flow field. Because the plumes from these three waste tanks are close to the model boundary and are truncated it is hard to discern how much they are affected by the flow field. The decision to use the larger transverse dispersivity for these waste tank releases is based on the benchmarking exercise and the very close match (between the PORFLOW and GoldSim models) for the I-129 dose peaks in Sector B as shown in Figures A-2-3 and A-2.4 of the report, *H-Area Tank Farm Stochastic Fate and Transport Model*, SRR-CWDA-2010-00093.

CC-FF-3

DOE indicated during the June 6, 2013 (ML13183A410) site visit that additional mixing is performed at the end of the flow path in GoldSim® probabilistic modeling to account for increased velocities. Clarify effective dilution factors applied at the end of the flow path near the compliance boundary in GoldSim® modeling.

Response CC-FF-3

In the HTF GoldSim Model, radionuclide transport in the saturated zone is evaluated using GoldSim analytical-solution based pipe elements. The conceptual model uses a 1-D solution for the advective-dispersive transport of dissolved species in a porous medium subject to sorption, simple decay, and/or ingrowth. The solution utilized by GoldSim is a Laplace transformed solution which is evaluated in the Laplace domain, and the results are subsequently inverted numerically back to the time domain. The 1-D analytical solution is based upon a spatially invariant steady flow field and therefore, a representative flow rate must be approximated by the user. In the HTF GoldSim Model, individual Darcy velocities for each waste tank are developed along PORFLOW generated pathlines going from the waste tank centroid to a point a 100-meters from the HTF boundary. The Darcy velocities (V_{Darcy}) are based upon the time it takes for the peak of a breakthrough curve generated from a pulse release at the waste tank of a conservative species to reach the 100-meter point (T_{Pulse}), the length of the surface trace of the pathline from HTF to the 100-meter point ($S_{pathline}$), and, the effective porosity (φ). The derivation is as follows:

(Eq. 1)

$$V_{Darcy} = \frac{S_{pathline}\varphi}{T_{Pulse}}$$

The derived velocity represents a harmonically averaged Darcy velocity along the pathline. This Darcy velocity will accurately account for the time it will take the center of mass of a pulse release to reach the 100-meter point but may not accurately represent the degree of dilution in the migrating species by the time it reaches the 100-meter point. The degree of dilution at the 100-meter point is also a function of changes in flow rates along the pathline. For example, consider a simple 1-D conceptual model for dissolved species transport in a constant thickness aquifer where the velocity increases in the downgradient direction as a function of a spatially invariant infiltration rate. The governing equation for 1-D advective-dispersive transport of a dissolved species in a 1-D flow field can be written as:

(Eq. 2)

$$\frac{\partial(\varphi RC)}{\partial t} = D\frac{\partial^2 C}{\partial l^2} - \frac{\partial(vC)}{\partial l} - \varphi R\lambda C + \sum_{i=1}^{Np}\varphi R\lambda_{pi}C_{pi}$$

which can be rewritten in the form:

(Eq. 3)

$$\frac{\partial(\varphi RC)}{\partial t} = D\frac{\partial^2 C}{\partial l^2} - v\frac{\partial C}{\partial l} - C\frac{\partial v}{\partial l} - \varphi R\lambda C + \sum_{i=1}^{Np}\varphi R\lambda_{pi}C_{pi}$$

where,

R is the retardation coefficient, *C* is the concentration, *D* is the dispersion coefficient, *λ* is the decay coefficient, *v* represents the Darcy velocity, *l* is distance along the flowpath, and N_p is the number of parent species. The third term on the right hand side of Equation 3 is a first-order term that represents the effects of the spatial change in velocities on derived concentrations. For the above HTF GoldSim conceptual model, if *v* is the Darcy velocity at an upgradient point, by replacing $\partial v/\partial l$, with *i/L* where *i* is the infiltration rate and *L* is the aquifer thickness, the third term of the right hand side of Equation 3 represents the dilution due to the infiltration of clean water along the pathline. Note that the third and fourth terms on the right hand side of Equation 3 can be combined when solving the equation. The solution of the Equation 3 for the 1-D conceptual model then derives concentrations at downgradient points which result from dilution due to recharge along the pathline while using a constant velocity required by the analytical solution. Note that for this choice of velocity, the mass arrival time will not be accurate. Similarly, results at downgradient points can also be obtained by omitting the third right hand side term in Equation 3 and multiplying the derived concentrations at downgradient points by the following dilution factor (*DF*):

(Eq. 4)

$$DF = \frac{Q_I}{Q_I + Q_A}$$

where,

Q_I is the infiltration flux along the pathline (m^3/yr) and Q_A is the aquifer flux (m^3/yr). If the Darcy velocity is known at the downgradient point, the *DF* in Equation 4 could be redefined as:

(Eq. 5)

$$DF = \frac{v}{V_{DarcyDowngradient}}$$

In this form, any increase in velocity along a pathline does not have to change in a linear manner. The upgradient velocity, *v*, could be replaced by a harmonically averaged velocity.

The HTF GoldSim Model takes advantage of the use of the *DF* defined in Equation 5 by multiplying the concentrations derived at the 100-meter point by a *DF* equal to the ratio of the harmonically averaged Darcy velocity along the pathline to the Darcy velocity at the 100-meter point. By using this *DF* in conjunction with results based upon the harmonically averaged flow rate, the HTF model maximizes the accuracy of the breakthrough time and degree of dilution obtained at the 100-meter point for this simplified transport model abstraction.

CC-FF-4

Provide approximate (effective) dilution factors for various HTF sources in GoldSim® probabilistic model considering vertical and horizontal dispersion, as well as additional mixing due to increased dilution at the end of the flow path. Evaluate dilution for various source release profiles such as pulse or continuous releases with respect to peak dose for various source locations and radionuclides.

Response CC-FF-4

In the HTF GoldSim Model, mass released from individual HTF waste tanks is transported by advection, through the unsaturated zone to a sink cell. The mass release is integrated over time, and the integrated mass is applied as a cumulative input source term to a 1-D advective-dispersive analytical solution used in GoldSim pipe pathways. In the HTF GoldSim Model, the cross section over which the 1-D solution is evaluated is equivalent to the product of the waste tank-width and a vertical source thickness assumed to be 3 meters. The solution used in the pipe-pathway element considers the effects of longitudinal dispersion, but a separate function, the GoldSim plume function is used to evaluate the influence of horizontal- and vertical-transverse dispersion on the dilution of dissolved radionuclides transported in the aquifer. The GoldSim plume function uses a series of Green's function and spatially integrates Green's function solutions to evaluate the influence of transverse dispersion on a source release from rectangular planar or rectangular prismatic sources. In the HTF GoldSim Model, the 3-D aspects of dispersion are implemented by taking the product of the plume function values and the 100-meter concentrations (i.e., the concentrations at a point 100 meters from the HTF) generated by the pipe pathway analytical solution. The total degree of dilution of mass migrating from the waste tank to the plume becomes a function of:

- The mixing of the mass released to the saturated zone in the analytical model;
- The influence of longitudinal dispersion in the derivation of concentrations in the pipe-pathway solution; and
- The influence of horizontal- and vertical-transverse dispersion as determined by the plume function.

In addition, a dilution factor, which represents downgradient changes in flow rates (as described in the response to CC-FF-3), is applied in the HTF GoldSim Model to evaluate the total attenuation due to dilution.

Because the degree of attenuation (due to longitudinal dispersion) depends upon the temporal nature of the source term, the influence of longitudinal diffusion will be evaluated separately in this report. By disregarding attenuation (due to longitudinal dispersion) approximate (effective) dilution factors were defined for the HTF waste tank sources by evaluating the influence of the transverse spreading due to dispersion and the flow rates in the aquifer. The effective dilution factor associated with the processes discussed above can be described by Equation 1:

(Eq. 1)

$$DF_{Effective} = \frac{V_{SZ} W_{Source} T_{Source} DF_{Boundary}}{\pi r_{tank}^2 V_{UZ} PF_{Plume}}$$

where,

V_{SZ} is the Darcy velocity in the saturated zone, V_{UZ} is the Darcy velocity in the unsaturated zone, W_{Source} is the diameter of the waste tank, r_{tank} is the waste tank radius, and T_{Source} is the source thickness. Additionally, the plume function (PF_{plume}) is the product of the Green's function solutions describing horizontal and vertical transverse dispersion at the end of the pipe and $DF_{Boundary}$ is the dilution factor, associated with changes in flow rate along the pathway.

Table CC-FF-4.1 presents the approximate (effective) dilution factors evaluated using Equation 1 in conjunction with the data used in the Base Case (Case A) of the HTF GoldSim Model. Table CC-FF-4.2 presents the approximate (effective) dilution factors, excluding the boundary dilution factor ($DF_{Boundary}$).

Table CC-FF-4.1: Effective Dilution Factors for HTF (Including $DF_{Boundary}$)

Waste Tank Number	**Dilution Factor (DF_{Total})**
T9	33
T10	28
T11	38
T12	35
T13	92
T14	47
T15	69
T16	118
T21	77
T22	68
T23	82
T24	69
T29	60
T30	50
T31	42
T32	37
T35	33
T36	24
T37	27
T38	53
T39	52
T40	50
T41	45
T42	59
T43	68
T48	69
T49	72
T50	100
T51	67

Table CC-FF-4.2: Effective Dilution Factors for HTF (Excluding $DF_{Boundary}$)

Waste Tank Number	Dilution Factor (DF_{Total})
T9	6
T10	4
T11	7
T12	6
T13	27
T14	9
T15	16
T16	37
T21	20
T22	13
T23	17
T24	15
T29	9
T30	7
T31	7
T32	5
T35	7
T36	5
T37	6
T38	7
T39	8
T40	9
T41	9
T42	8
T43	7
T48	9
T49	12
T50	8
T51	5

These partial dilution factors are applicable for continuous or pulse releases and for all radionuclides. The dilution factors were calculated for each HTF waste tank as the source. Note that the dilution factors presented below are based on the final values of Darcy velocity in the unsaturated zone and are evaluated at the points where flow pathlines cross the 100-meter point.

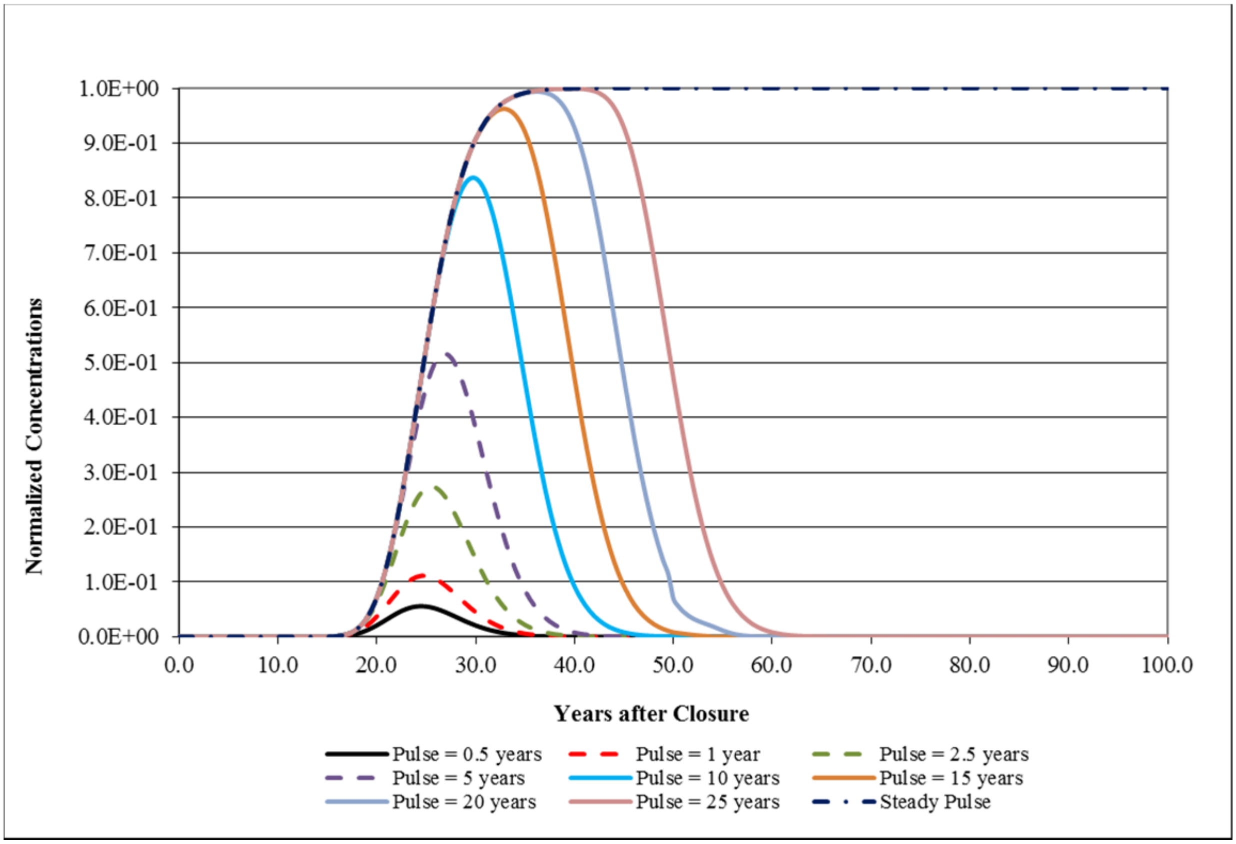
The total degree of dilution to which radionuclides are subject to while migrating through the aquifer is also a function of longitudinal dispersion or spreading. The degree of attenuation determined by the GoldSim pipe pathway analytical model is a function of the longitudinal dispersivity, the pipe length, and the temporal nature of the source term. Because this portion of the dilution is dependent on the release profile of the source term, the process was evaluated for a typical path length, Darcy velocity, and other parameters consistent with the HTF GoldSim Model aquifer parameters. The pipe pathway model was used to evaluate the attenuation of a conservative species migrating along a 1,000-foot pathway from the waste tank center to the 100-meter point using a Darcy velocity of 10 ft/yr, a porosity of 25 %, and the 10.4-foot HTF model longitudinal dispersivity. Note that the degree of attenuation is independent of the Darcy velocity, which controls only the timing of the breakthrough curve.

The evaluation used source pulse lengths of 0.5, 1, 2.5, 5, 10, 15, 20, and 25 years as well as a steady source. The results are presented as normalized doses that are normalized with respect to the steady-state results for a constant flux. The attenuation for each source pulse length is defined by a separate attenuation factor, and presented in Table CC-FF-4.3. The 100-meter concentration breakthrough curves for each simulation are presented in Figure CC-FF-4.1. Note that the attenuation factors are the inverses of the peak values of the normalized breakthrough curves presented in Figure CC-FF-4.1. Due to the model parameters used, the dilution factor for a 25-year pulse essentially equals the steady pulse after 25 years. A complete dilution factor can then be defined as the product of $DF_{Effective}$ and the attenuation factor.

Table CC-FF-4.3: Attenuation Factors for HTF

Pulse Length (yr)	Attenuation Factor
0.5	17.89
1	8.99
2.5	3.65
5	1.94
10	1.20
15	1.04
20	1.01
25	1.00
Steady Pulse	1.00

Figure CC-FF-4.1: Normalized 100-Meter Concentration Breakthrough Curves for Modeled Source Pulses



Inadvertent Intrusion (INT)

CC-INT-1

Clarify whether intruder doses presented in Section 6 of the HTF Performance Assessment (SRR-CWDA-2010-00128, Rev. 1) consider alternative cases.

Response CC-INT-1

Section 6.5.1 of the HTF PA includes the following IHI alternative scenarios:

1. The impact of drilling into a 4-inch line rather than a 3-inch line (Section 6.5.1.1)
2. The impact of drilling into a waste tank rather than a 3-inch line (Section 6.5.1.2)
3. The impact of drilling into locations closer to specific waste tanks rather than at the 1-meter boundary (Section 6.5.1.3)

However, Section 6 of the HTF PA did not include intruder dose results for the alternative waste tank configurations (i.e., Cases B, C, D, and E). Provided in Table CC-INT-1.1 and Figures CC-INT-1.1 through CC-INT-1.8 are the chronic IHI dose results for each of these alternative configurations (by sector and by key radionuclides). The 1-meter boundaries for each of the six sectors (Sectors A through F) are shown on Figure 5.2-5 of the HTF PA.

These Alternative Case modeling runs were developed using the HTF PORFLOW Model and the same GoldSim dose calculator as used for the Base Case (Case A of the HTF PA modeling configurations). These models only include the sensitivity run radionuclides, as identified in Table 5.2-9 of the HTF PA. This is why the Base Case value is lower than that reported in Section 6.4 of the HTF PA. Case E is discussed further in the response to CC-INT-2.

The acute IHI dose is based on exposure to drill-cuttings and does not rely upon contaminant transport via groundwater. In addition, applying an alternative waste tank configuration does not alter the acute IHI dose. Therefore, the acute IHI dose results are not discussed in this response.

Table CC-INT-1.1: Chronic IHI Peak Doses for the Base Case and Alternative Cases

Case	1,000-Year Peak Dose	10,000-Year Peak Dose	100,000-Year Peak Dose
A	40 mrem/yr at year 100	50 mrem/yr at year 10,000	348 mrem/yr at year 50,520
B	40 mrem/yr at year 100	41 mrem/yr at year 10,000	214 mrem/yr at year 19,070
C	40 mrem/yr at year 100	83 mrem/yr at year 9,070	286 mrem/yr at year 42,400
D	40 mrem/yr at year 100	63 mrem/yr at year 6,190	312 mrem/yr at year 45,370
E	367 mrem/yr at year 1,000	1,127 mrem/yr at year 2,160	1,127 mrem/yr at year 2,160

Case A = Base Case

Figure CC-INT-1.1: Case B Chronic IHI Dose

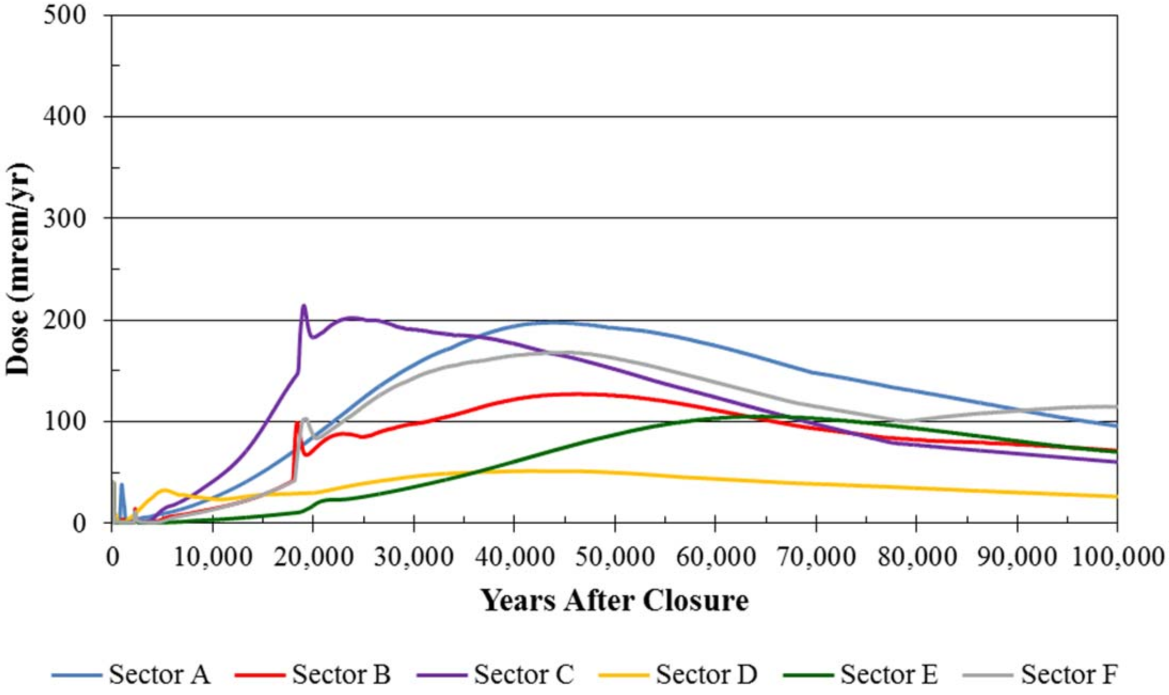


Figure CC-INT-1.2: Case B Individual Radionuclide Contributors to Chronic IHI Dose

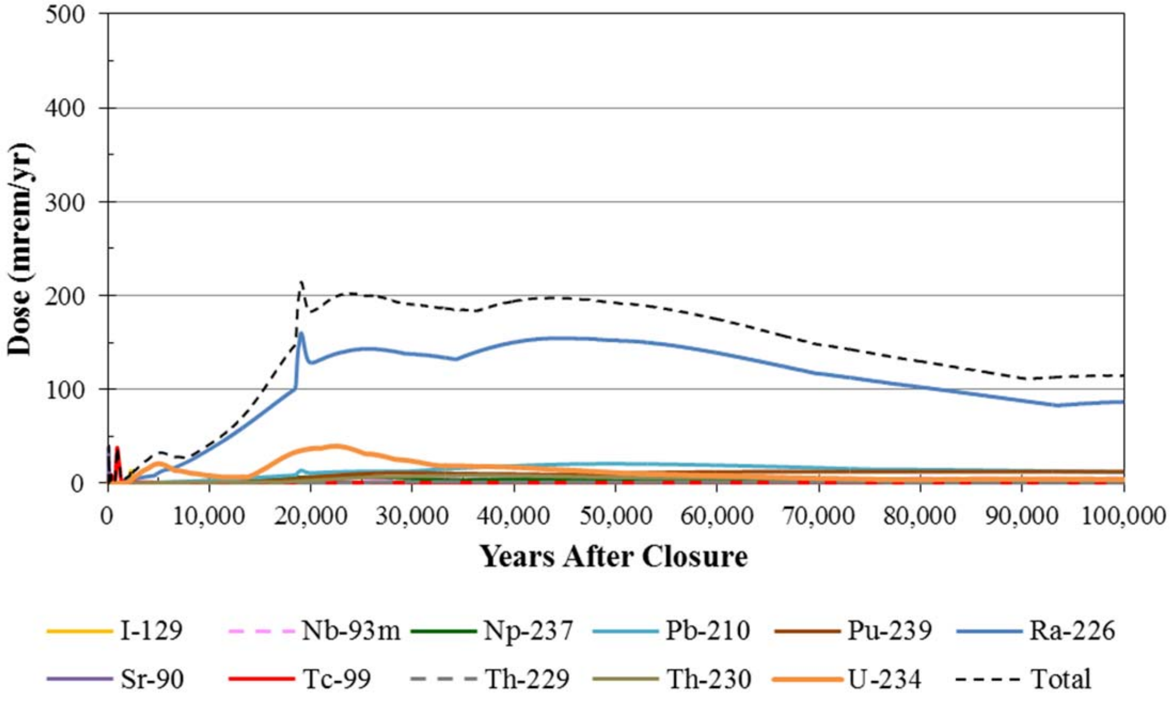


Figure CC-INT-1.3: Case C Chronic IHI Dose

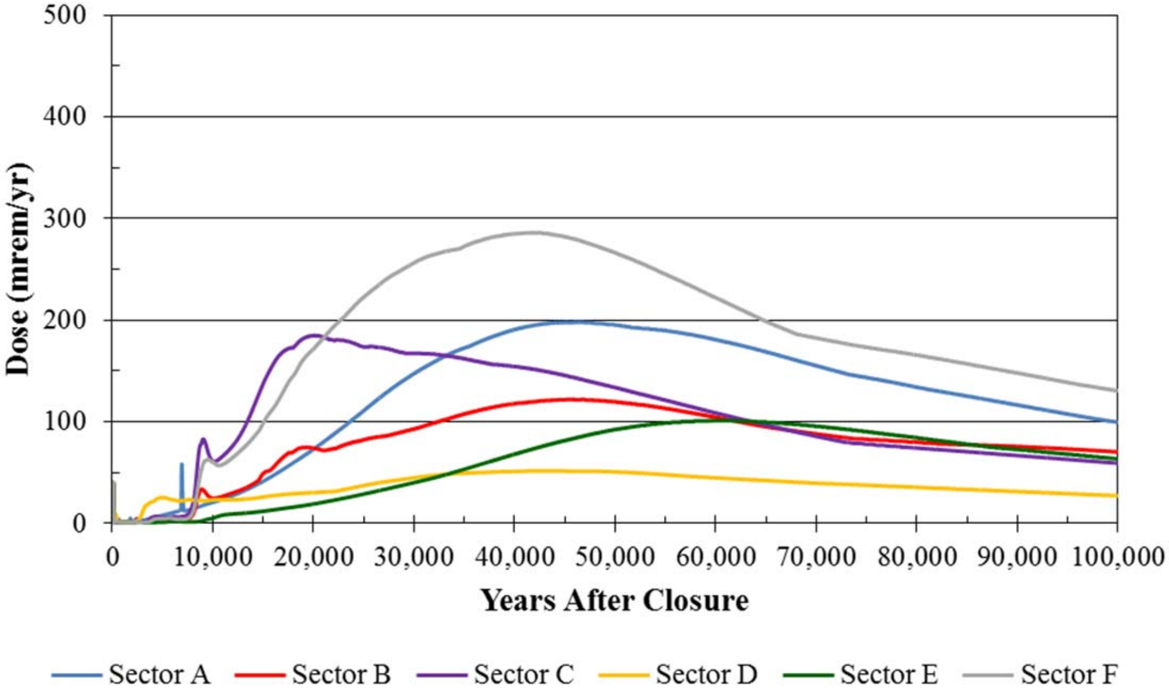


Figure CC-INT-1.4: Case C Individual Radionuclide Contributors to Chronic IHI Dose

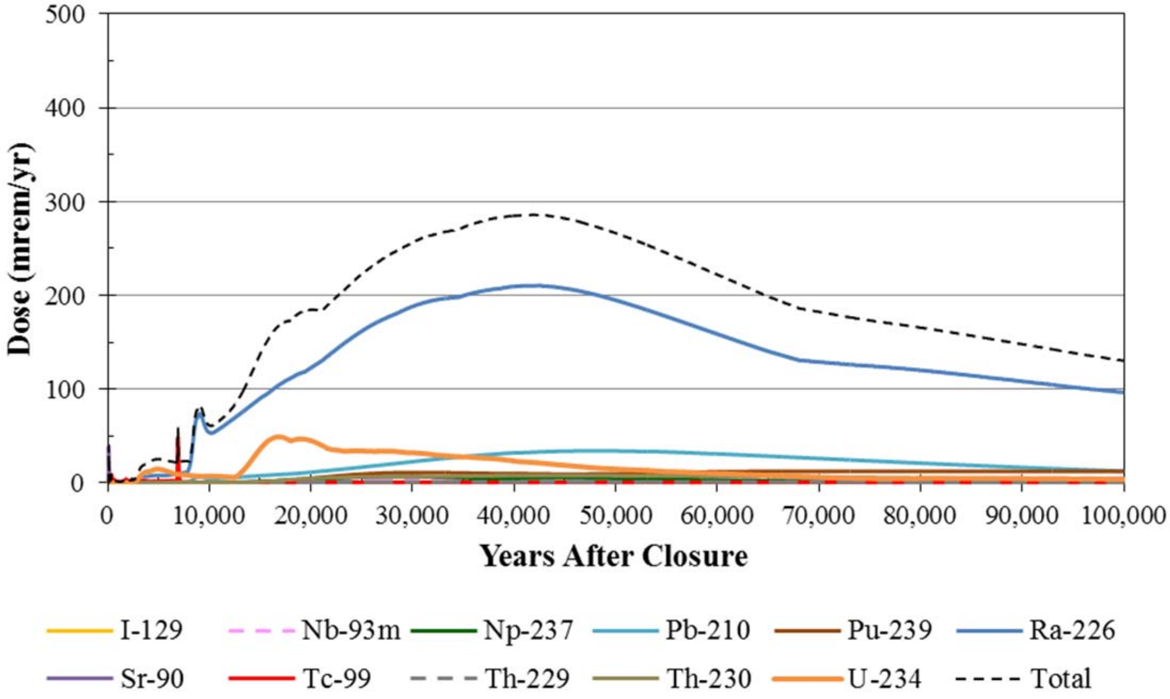


Figure CC-INT-1.5: Case D Chronic IHI Dose

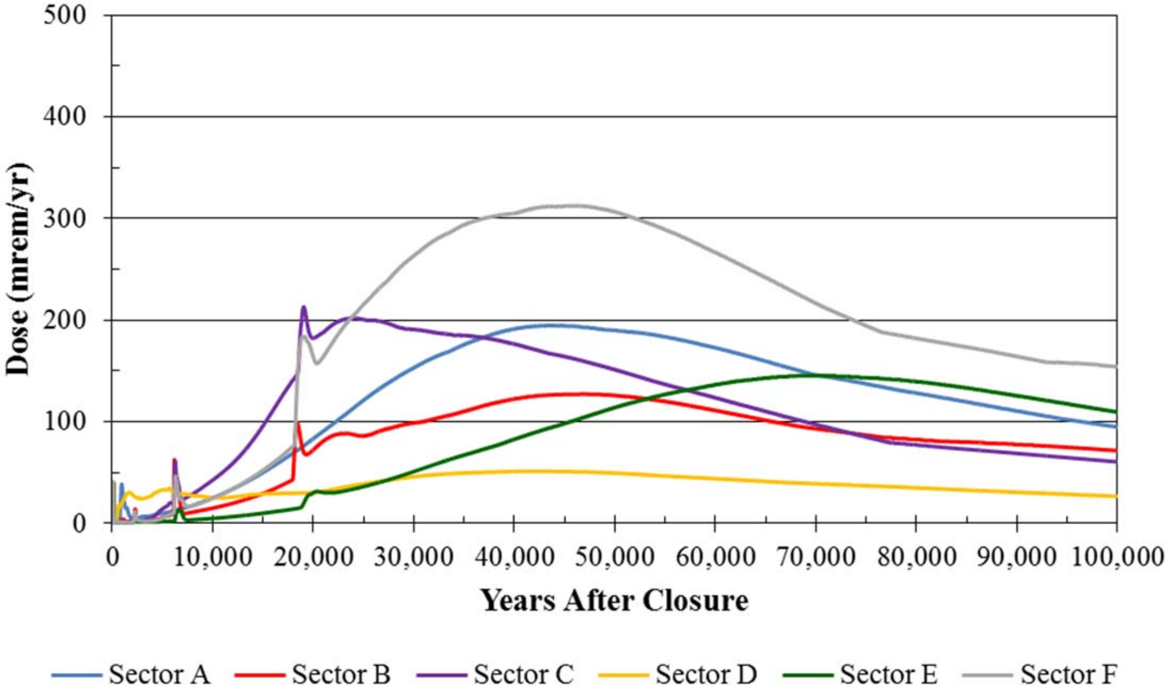


Figure CC-INT-1.6: Case D Individual Radionuclide Contributors to Chronic IHI Dose

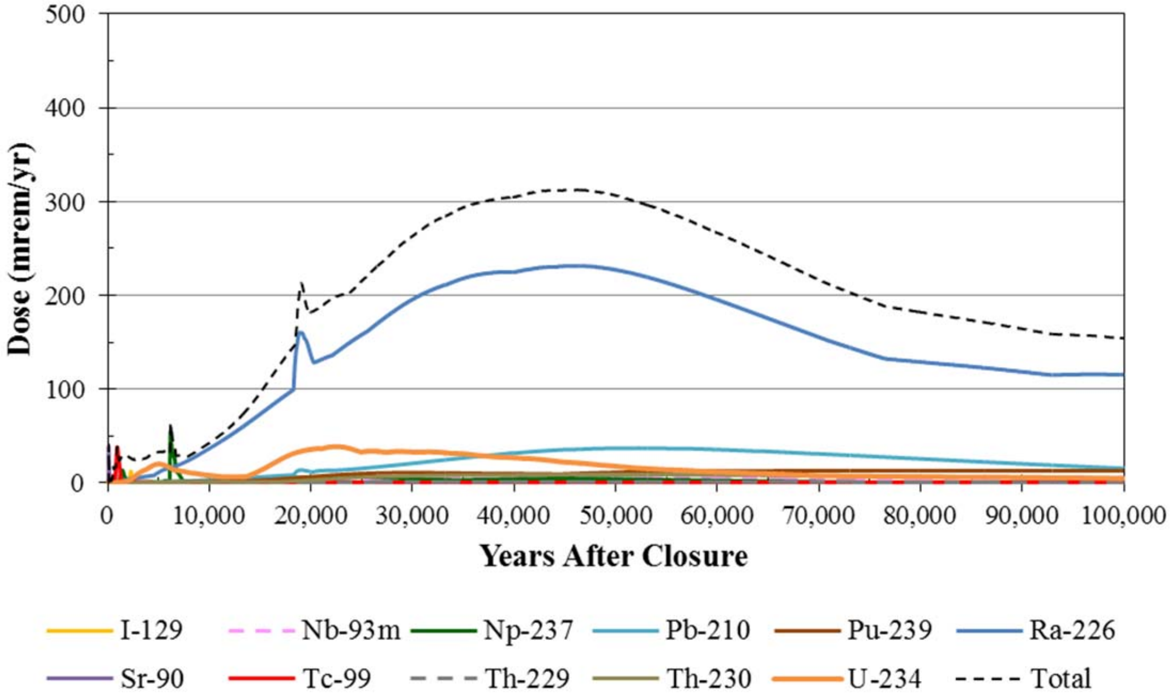


Figure CC-INT-1.7: Case E Chronic IHI Dose

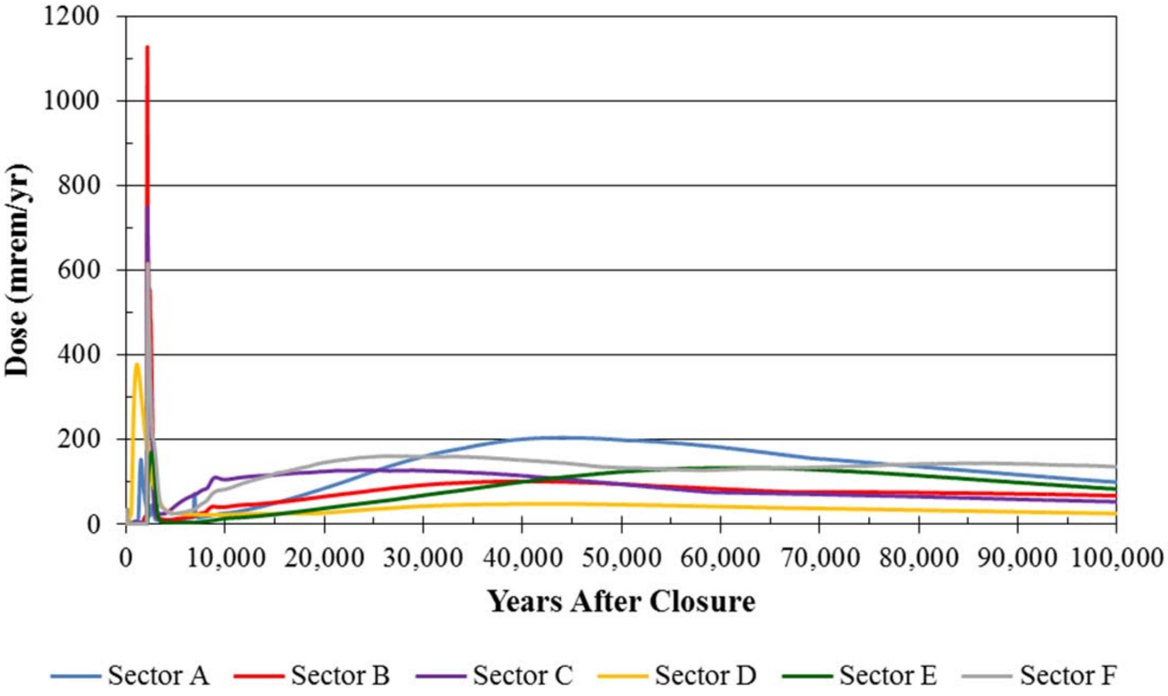
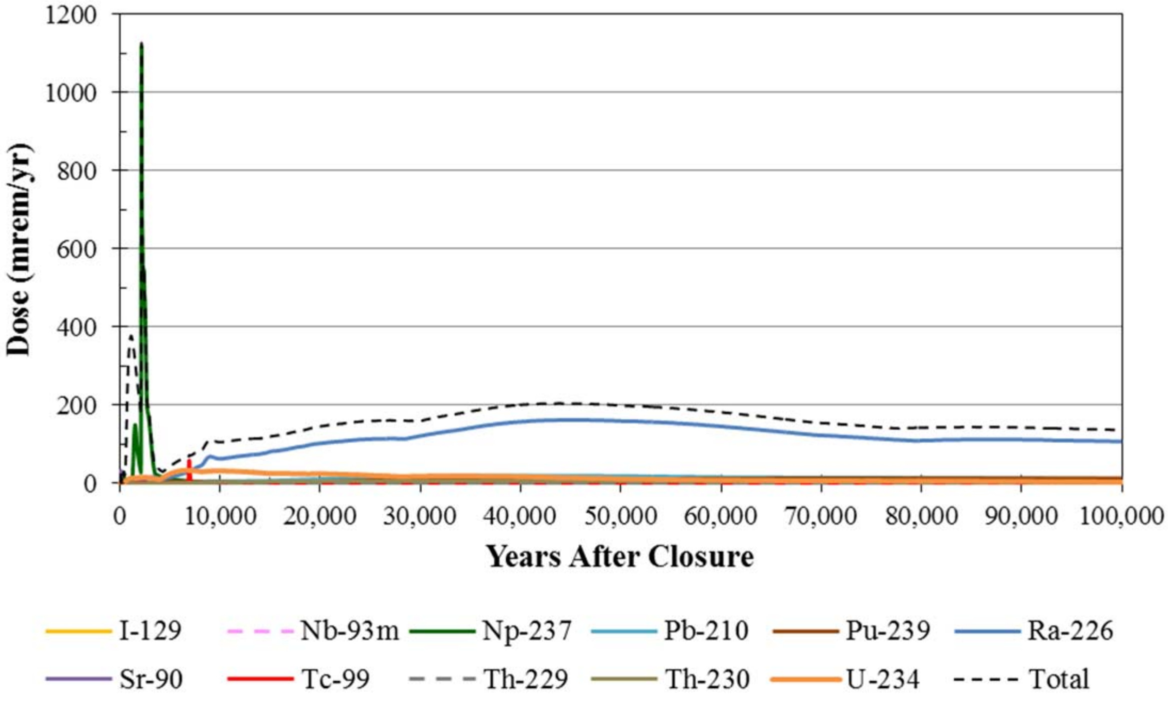


Figure CC-INT-1.8: Case E Individual Radionuclide Contributors to Chronic IHI Dose



CC-INT-2

Provide detailed results for alternative case E.

Response CC-INT-2

The chronic IHI dose for Case E was estimated to peak at 1,127 mrem/yr (at 2,160 years after HTF final facility closure), as shown in Table CC-INT-1.1 and Figures CC-INT-1.7 and CC-INT-1.8 of the response to CC-INT-1. Case E is considered the least probable configuration out of the scenarios identified in Table CC-INT-1.1 (see Section 5.6.3.2 of the HTF PA and the response to RAI-NF-10).

Note that Case E modeling is a configuration used to represent an unanticipated, bounding scenario, and is presented as a thought exercise to understand the system better. As described in Section 4.4.2.5 of the HTF PA, the fast flow path in Case E is non-mechanistically assumed to bypass a number of barriers in this scenario. This fast flow path would be:

1. Liquid passing through the roof of the waste tank;
2. Down the full height of the grouted waste tank through an assumed gap between the grout and the primary liner;
3. Along the entirety of floor of the waste tank contacting all of the residual waste (i.e., a gap would exist between the residuals at the waste tank bottom and the 100’s of thousands of gallons of grout above the residuals;
4. Out through the center of the primary liner floor;
5. Through the grout layer between the primary tank and the secondary liner;
6. Through the annular pan; and
7. Completely through the basemat.

The fast flow occurs in Case E without the grout’s reducing properties being imparted onto the associated liquid.

More than 99 % of the peak dose from Case E is attributed to a single radionuclide, Np-237, and the timing of this peak strongly indicates that the peak dose is driven by the simultaneous failure (at year 2,077) of all 17 of the Type III and Type IIIA tank liners. In other words, the fast flow path scenario outlined in items 1 through 7 above would occur for all 17 of the Type III and IIIA tanks simultaneously.

Case E is considered an unlikely scenario for the following reasons:

1. Fast flow paths are not expected to simultaneously bypass every barrier within every waste tank;
2. The concrete in the basemat is expected to retard the transport of Np-237;
3. The primary liner for each of the Type III and Type IIIA tanks are expected to fail at varying times;
4. Except for the type IV tanks, all waste tanks have cooling coils such that it is unlikely that fast flow paths, if they did occur, would develop only along primary tank walls and not along cooling coils. If fast flows occurred along the cooling coils the reducing capacity of the grout would be expected to be imparted onto tank pore water; and
5. Fast flows through the waste tank would not be expected to mobilize 100% of the contamination zone.

As such, it is expected that the peak doses associated with Np-237 in Case E would be significantly mitigated if these conservatisms were tempered in the modeling.

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