UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 06, 2020

Mark Gilbertson
Associate Principal Deputy Assistant Secretary
US Department of Energy
Office of Environmental Management
Office of Regulatory and Policy Affairs (EM-4)
1000 Independence Ave., SW
Washington, DC 20585-1290

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON THE DRAFT WASTE INCIDENTAL TO REPROCESSING EVALUATION FOR VITRIFIED LOW-ACTIVITY WASTE DISPOSED ONSITE AT THE HANFORD SITE (DOCKET NUMBER PROJ0736)

Dear Mr. Gilbertson:

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the “Draft Waste Incidental to Reprocessing Evaluation for Vitrified Low-Activity Waste Disposed Onsite at the Hanford Site, Washington” dated April 2020, the “Performance Assessment for the Integrated Disposal Facility, Hanford Site, Washington” dated August 2018 and associated documentation provided by the U.S. Department of Energy (DOE). This independent review was conducted in accordance with an Interagency Agreement between the DOE and the NRC. This agreement requests the NRC’s consultative technical review to determine if the draft Waste Incidental to Reprocessing (WIR) evaluation meets DOE Manual 435.1-1 criteria for Vitrified Low Activity Waste Disposed Onsite at the Hanford Site may be managed as low-level radioactive waste. In the agreement, DOE requests NRC consultative emphasis on DOE M 435.1-1 criterion 2 regarding meeting safety standards comparable to the performance objectives set out in 10 CFR Part 61 Subpart C, over DOE M 435.1-1 criterion 1 regarding the removal of key radionuclides. Additionally, the DOE requests consultation pertaining to reasonable expectation of compliance with the performance objectives for a compliance period of 1,000 years.

NRC has attached a Request for Additional Information (RAI), which is a list of comments for which the NRC staff needs responses from the DOE before the NRC can complete its review. This RAI is based on our risk-informed review of the draft WIR evaluation and supporting documentation. NRC evaluated DOE’s intruder and performance assessment results, including sensitivity cases in developing the risk insights used to inform the review. The NRC staff reviewed DOE’s performance assessment model developed with the software package GoldSim. The NRC staff used DOE’s model to evaluate alternate assumptions and cases to develop additional risk insights.

As NRC continues its review of DOE documents and RAI responses, NRC may develop additional comments for which NRC will need a response from DOE.

To meet the current schedule and complete the review by June 16, 2021, NRC requests responses to the RAI on or before February 12, 2021. If it would be useful to DOE, NRC would be willing to meet with DOE to discuss the RAI or DOE's responses.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

If you have any questions related to this letter, please contact Maurice Heath, Project Manager in the Division of Decommissioning, Uranium Recovery, and Waste Programs at 301-415-3137 or by e-mail at Maurice.Heath@nrc.gov.

Sincerely,

Chris McKenney, Branch Chief
Risk and Technical Analysis Branch
Division of Decommissioning, Uranium Recovery
and Waste Programs
Office of Nuclear Material Safety
and Safeguards

Enclosure: Request for Additional Information

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON THE DRAFT WASTE INCIDENTAL TO REPROCESSING EVALUATION FOR VITRIFIED LOW-ACTIVITY WASTE DISPOSED ONSITE THE HANFORD SITE

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Request for Additional Information
Draft Waste Incidental to Reprocessing Evaluation for
Vitrified Low-Activity Waste Disposed Onsite at the Hanford Site, Washington

Structure of Comments
The U.S. Nuclear Regulatory Commission (NRC) staff's review comments are separated into categories based on the Department of Energy (DOE) Manual 435.1-1 evaluation process criteria to determine if Waste Incidental to Reprocessing (WIR) can be managed as low-level waste. DOE Manual 435.1-1 states that WIR can be managed as low-level waste if an evaluation shows that the following criteria are met:

1. Have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical; and

2. Will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C, *Performance Objectives*; and

3. Are to be managed, pursuant to DOE's authority under the *Atomic Energy Act of 1954*, as amended, and in accordance with the provisions of Chapter IV of this Manual, provided the waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C low-level waste as set out in 10 CFR 61.55, *Waste Classification*; or will meet alternative requirements for waste classification and characterization as DOE may authorize.

The path forward provided for each comment is one recommended approach to resolution; the NRC staff understands that there may be more than one method to adequately address the technical issues raised in the comments. Appropriate responses to some comments may depend on the nature of the resolution of other comments. A sub-topic is presented parenthetically after each Request for Additional Information (RAI) number.

NRC staff performed a review of the draft waste evaluation for Waste Management Area-C (WMA-C) and issued a technical evaluation report (TER) (ML20128J832). The analyses DOE completed for WMA-C used common information and modeling techniques for numerous aspects of the performance assessment (PA) for WMA-C and Vitrified Low Activity Waste (VLAW) (e.g., infiltration rates, biosphere). While NRC staff had recommendations on technical topics associated with the WMA-C review, most of the technical topics were not risk-significant given the current understanding and residual waste inventory expected to remain in the tanks. However, those technical recommendations could be significant with respect to VLAW decision making. DOE was not able to assess NRC's recommendations for WMA-C in time to determine the impact on VLAW because of the differences in timing of the two projects. There are specific examples used in the following RAIs from those recommendations. As DOE reviews NRC's comments, they should consider whether other recommendations are relevant to the VLAW PA.

DOE indicated that the scope for the VLAW draft waste incidental to reprocessing (WIR) evaluation was limited to vitrified waste generated as part of the Direct Feed Low Activity Waste (DFLAW) process (DOE-ORP-2020-01, 2020)[1]. Secondary solid wastes (SSW) generated by the process were not within the scope of the draft waste evaluation though those secondary wastes were evaluated in the PA as they would be disposed in the same facility as the VLAW. NRC evaluated the scope of the evaluation in the acceptance review and determined that the

[1] This report will be referred to as the draft WIR evaluation throughout this document.

DOE approach was not consistent with the intent of the incidental waste process. DOE's election of vitrification as the primary waste production process results in some key radionuclides that are volatized and effectively separated from the waste (e.g., ^{129}I), or are removed in other processing steps. If the majority of that activity that is separated or removed will be disposed in near-surface disposal (i.e. as other than high-level waste), then the resulting wasteforms and waste streams are within the scope of the draft waste evaluation, especially for DOE Manual 435.1-1 Criterion 2 as the key radionuclides drive the long-term risk for the disposal. As a result, NRC has included secondary solids wastes within the scope of the review.

<u>RAI 1-1</u> (Removal of ^{90}Sr to the Maximum Extent Practicable)

Comment:
Additional information is needed on the amount of soluble ^{90}Sr expected to be in the waste processed for DFLAW and the technologies that may be used to remove it to the maximum extent practical.

Basis:
In the draft WIR evaluation, DOE states that most of the ^{90}Sr is insoluble but strontium can be soluble in some tanks with higher organic concentrations. Tanks with soluble ^{90}Sr are not currently planned to be part of the DFLAW campaigns and therefore are not discussed in the draft WIR evaluation. DOE indicated there would likely be a separate WIR evaluation for the tanks beyond DFLAW. However, in the draft WIR evaluation, DOE stated that the Integrated Disposal Facility (IDF) PA includes VLAW from tanks with soluble ^{90}Sr (see footnote 51). The NRC is reviewing the total risk from the disposal of waste in the IDF, including the potential for soluble ^{90}Sr to be part of the inventory for IDF, as within the scope of this evaluation, and therefore is requesting additional information on what amount of soluble ^{90}Sr is expected in the tanks and the technologies that may be used to remove soluble ^{90}Sr to the maximum extent practical.

Table 3-29 of the PA document[2] shows that nearly 100% of the ^{90}Sr is assumed to remain in the high-level waste (HLW) (i.e., not listed as waste going to IDF) under Case 7, 8B, 9, 10A, and 10B. It is unclear if any of these cases include the processing of the tanks with the soluble ^{90}Sr waste.

Waste Form	Case 7	Case 8B	Case 9	Case 10A	Case 10B
Non-IDF (HLW)	99.46%	99.61%	99.71%	99.54%	99.63%
ILAW Glass	0.54%	0.39%	0.29%	0.45%	0.36%
LAW Melters	0.00%	0.00%	0.00%	0.00%	0.00%
ETF-Generated Secondary Solid Waste	0.00%	0.00%	0.00%	0.00%	0.00%
Secondary Solid Waste	0.00%	0.00%	0.00%	0.00%	0.00%
IDF Total	**0.54%**	**0.39%**	**0.29%**	**0.46%**	**0.37%**

Path Forward
Please provide additional information on what percentage of the ^{90}Sr is estimated to be soluble versus insoluble in the tanks. Please provide additional basis for what percentage of the soluble ^{90}Sr DOE estimates can be extracted using the ion exchange columns, or other technologies planned to be used.

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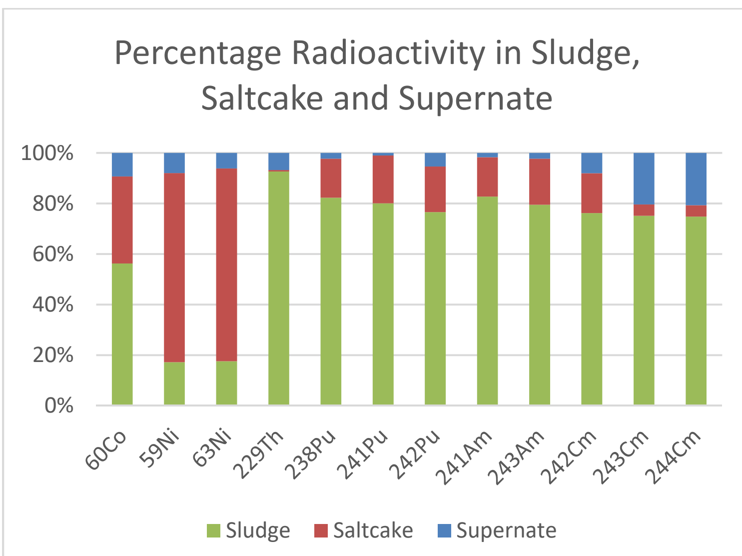
[2] The PA document is RPP-RPT-59958 Rev. 1, Performance Assessment for the Integrated Disposal Facility, Hanford Site, Washington, Department of Energy, Richland, WA, August 2018. The PA consists of computer files (models) as well as a supporting document.

RAI 1-2 (Percentage of Key Radionuclide Removal)

Comment
Additional information is needed on the percentage of key radionuclides removed from the waste that will be disposed in the integrated disposal facility.

Basis
Table 3-29 in the PA document lists the wasteform distributions resulting from the five cases considered to potentially be disposed of in IDF for 11 of the 25 key radionuclides identified by DOE. The 14 key radionuclides that are not listed in Table 3-29 are ^{60}Co, ^{59}Ni, ^{63}Ni, ^{228}Rn, ^{229}Th, ^{232}Th, ^{238}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm. (Note that ^{233}U and ^{235}U are listed in Table 3-29 but they are not listed as a key radionuclide in the draft WIR evaluation.)

In Table 2-3 of the draft WIR evaluation DOE lists the estimated total radioactivity in the 177 underground waste tanks, broken down by how much radioactivity is in the supernate, saltcake, and sludge for certain radionuclides (Note that key radionuclides ^{228}Rn, ^{232}Th are not listed in Table 2-3 of the draft WIR evaluation). For 12 of the 14 key radionuclides that are not listed in Table 3-29, Table 2-3 of the draft WIR evaluation shows that there is some percentage of these radionuclides that remains in the supernate or saltcake. For example, more than 80% of the ^{59}Ni and ^{63}Ni is in the saltcake or supernate, and about 45% of the ^{60}Co is in the saltcake or supernate. About 20-25% of the ^{238}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, and ^{244}Cm are not in the sludge. In the draft WIR evaluation, DOE states that ^{59}Ni is present in very low concentrations in the vitrified LAW and is an insignificant contributor to dose after IDF closure. However, it is unclear what percentage of these radionuclides may be present in the various waste forms that will be disposed of at IDF and what was removed by processing.



Data from Table 2-3 in the draft WIR evaluation

Path Forward
Please provide the percentages of key radionuclides removed for those key radionuclides (see Table 4-3 of the draft WIR evaluation) that are not already included in Table 3-29.

<u>RAI 1-3</u> (Percentage of ^{99}Tc and ^{129}I Recycled versus Removed)

Comment
Additional information is needed on the percent of the ^{99}Tc and ^{129}I that could potentially be removed from the waste versus remaining in either the VLAW or the SSW. (See also RAI 2-10).

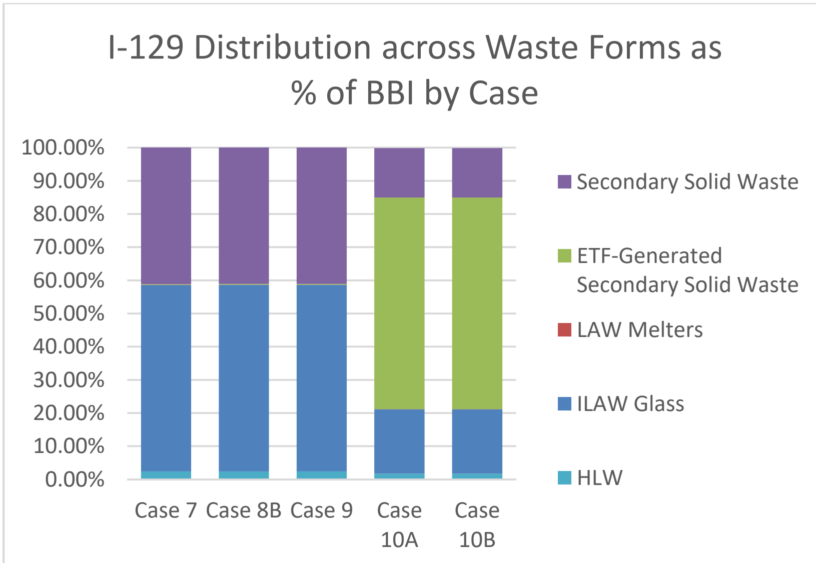
Basis
In the draft WIR evaluation, DOE stated on page 4-12 that, "with respect to ^{99}Tc and ^{129}I, the LAW Vitrification Facility is designed to maximize the capture of these radionuclides in the vitrified waste form. The LAW Vitrification Facility off gas system is designed to recycle and/or capture that portion of volatile radionuclides (including ^{99}Tc and ^{129}I) which are not vitrified (see Section 2.5.3)." During the LAW vitrification process, the volatile components will be drawn off through the melter offgas treatment system, will go through a submerged bed scrubber (SBS) and Wet Electrostatic Precipitator (WESP), two stages of high-efficiency particulate air (HEPA) filters, as well as two carbon adsorber beds, which remove the ^{129}I.

The draft WIR evaluation states that the ^{99}Tc and ^{129}I in the liquid condensate resulting from the offgas system (from the SBS and WESP) can potentially be routed in three ways:

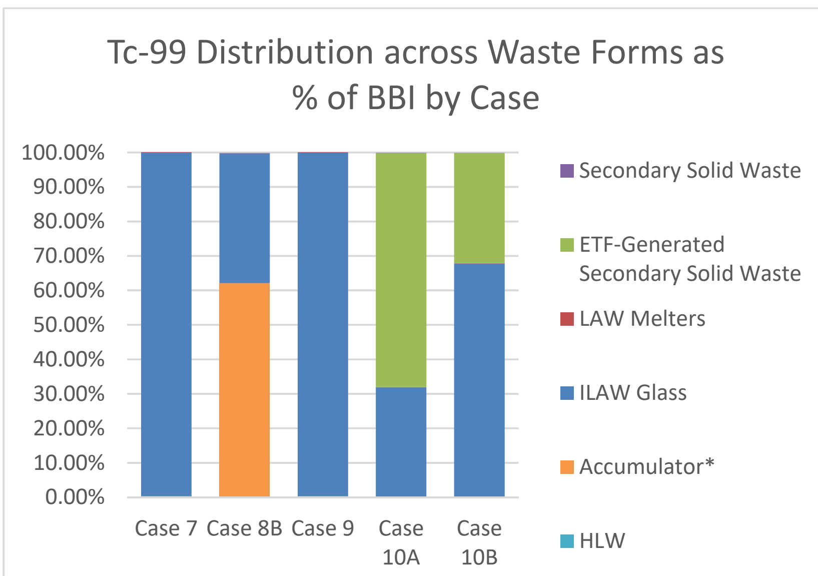
1. recycle back to the LAW Vitrification Facility for blending with incoming DFLAW feed;
2. return back to the Hanford tank farms; and
3. purge via a tanker truck load-out station ((RPP-RPT-58971, Effluent Management Facility Evaporator Concentrate – Purge Alternatives Evaluation).

Table 3-29 in the PA document provides the summary of radionuclide inventories and wasteform distributions for five cases. As shown below, the majority of the ^{99}Tc and ^{129}I does not remain in the HLW tanks. In Case 7, Case 8B, and Case 9, about 40% of the Best Basis Inventory (BBI) of ^{129}I ends up in the Secondary Solid Waste (SSW). In Case 10A and Case 10B, nearly 80% of the ^{129}I ends up in either the Effluent Treatment Facility-Generated (ETF-Generated) SSW or the SSW. A negligible amount of the ^{129}I remains in the HLW in any of cases.

In Case 7, over 99% of the ^{99}Tc is assumed to end up in the Immobilized Low-Activity Waste (ILAW) glass. In Case 8B, as a result of removing ^{99}Tc from the LAW off-gas stream, 16,500 Ci of ^{99}Tc accumulates over the course of the waste treatment and immobilization plant mission. The footnote on Table 3-29 states that "these curies would likely be processed in High-Level Waste and would not end up at the IDF". In the PA document, DOE further describes Case 8B as follows: "a ^{99}Tc removal unit operation has been added after the submerged bed scrubber (SBS) wet electrostatic precipitator (WESP) to remove ^{99}Tc at an efficiency of 99% from the LAW liquid off-gas steam prior to being recycled back to pretreatment. This inventory would not be disposed of at IDF and represents the lower range of ^{99}Tc inventory in glass and grout waste forms."



I-129 Distribution across Waste Forms as % of BBI by Case (*Adapted from data in Table 3-29 of the PA)*



Tc-99 Distribution across Waste Forms as % of BBI by Case (*Adapted from data in Table 3-29 of the PA)*

Path Forward
Please provide additional information on the percentage of ^{99}Tc and ^{129}I that is expected to be recycled back to the DFLAW feed, percent returned to the tank farm to be disposed of as HLW, and percent purged via the Effluent Management Facility (EMF) evaporator concentrate.

Please provide additional information about the accumulator mentioned in Table 3-29 of the PA document. Please specify if the ^{99}Tc is assumed to be accumulating in one component or in multiple parts of the waste treatment plant (WTP). Please provide the hypothetical plan for disposal of the accumulator waste under this scenario.

<u>RAI 1-4</u> (Alternative Technology Evaluation Impacting ^{99}Tc and ^{129}I)

Comment
Additional information is needed on the alternative technologies considered for removal of ^{129}I and ^{99}Tc.

Basis
Depending on the processes used and separation that may occur, a moderate to significant amount of ^{129}I and ^{99}Tc may end up in the SSW (See the previous figures in RAI 1-3). If these key radionuclides in the SSW ultimately are disposed of in IDF, the NRC staff considers those secondary waste streams within the scope of the draft WIR evaluation and therefore considers the impact of the secondary waste and the alternative technologies considered for the wasteforms. The NRC staff note that in discussions with the DOE, the DOE has stated that the scope of the draft WIR evaluation is limited to the vitrified waste. DOE has stated that the SSW is a newly generated waste stream and will include a wide variety of waste (HEPA filters, or carbon filters) that will be generated after the low activity waste has been vitrified. Therefore, DOE considers SSW to be outside the scope of the WIR evaluation, but DOE has included the SSW as part of IDF PA analysis and the SSW will be classified to ensure it meets the Waste Acceptance Criteria (WAC) for IDF.

DOE has conducted previous studies to compare alternative technologies for removal of radionuclides from the Hanford tank wastes. These studies are summarized in the Technical Basis Summary Report (WHC-SD-WM-TI-699) completed in 1996. In the draft WIR evaluation DOE summarized the findings of the report, but DOE did not discuss alternative technologies considered for the SSW waste forms or technologies that would selectively drive the ^{99}Tc and ^{129}I back into the HLW.

Path Forward
As a result of high operating temperatures, the vitrification process appears to selectively partition the ^{99}Tc and ^{129}I to the SSW waste stream during processing of the waste. Given that ^{99}Tc and ^{129}I are key risk drivers, please provide information regarding potential technologies that may have been considered to connect the offgas system to other waste treatments that would result in those key radionuclides being incorporated into HLW compared to the VLAW or SSW.

<u>RAI 1-5</u> (Removal and Disposal of Separated ^{129}I)

Comment

In the draft WIR evaluation, DOE indicated that they did not identify a technology that could practically remove ^{129}I from tank wastes. It isn't clear why the ^{129}I that is separated very efficiently by the vitrification process could not be disposed as HLW.

Basis

DOE identified ^{129}I as a key radionuclide with respect to protection of an offsite member of the public. Iodine-129 is present in tank waste in very low concentrations and therefore it is difficult to remove. Iodine-129 is very long-lived and mobile in the environment. The vitrification process operates at very high temperatures and volatizes nearly all of the ^{129}I present in the waste streams. This ^{129}I then goes into the offgas system and is captured or recycled back to the glass melters. Even with recycling the capture of ^{129}I in the glass is less than 50%.

Section 4.2.2.6 of the draft WIR evaluation states that "DOE has also explored whether there is an available technology to remove ^{129}I. However, the ^{129}I concentration in the tank wastes is typically 1,000 to 10,000 times lower than would exist in commercial fuel dissolver solutions for which an available iodine removal technology was developed. Iodine-129 removal is not considered to be technically practical because no technology has been demonstrated for the relatively low concentrations in the Hanford Site tank waste (WHC-SD-WM-TI-699, Technical Basis for Classification of Low-Activity Waste Fraction from Hanford Site Tanks)."

In section 2.5.3 of the draft WIR evaluation DOE stated that the "The volatile components will be drawn off through the melter offgas treatment system, go through a submerged bed scrubber (SBS) and Wet Electrostatic Precipitator (WESP). The LAW melter offgas system consists of two stages of high-efficiency particulate air (HEPA) filters for the purpose of removing radioactive particulates from the offgas, in order to achieve compliance with both environmental and occupational dose limits. Downstream of the HEPA filters are two carbon adsorber beds filled with granular activated carbon media. By design, the purpose of these beds is to remove mercury, halides, and acid gases as well as ^{129}I." It was not clear whether DOE evaluated the disposal of the ^{129}I separated by the vitrification process as HLW.

Path Forward

Please provide information as to whether DOE evaluated the disposal of the ^{129}I that would be adsorbed by the two carbon adsorber beds filled with granular activated carbon media as HLW. Please also describe what percentage of the ^{129}I can technically and practically be removed using the carbon adsorber beds.

RAI 2-1 (Scope of PA Compared to Scope of Draft WIR Evaluation)

Comment
The results from the PA that are directly applicable to the scope of the draft WIR evaluation are not clear. One factor could have been the timing of the completion of the PA and draft WIR evaluation.

Basis
The performance assessment was completed before the draft WIR evaluation. The performance assessment was completed in 2018 while the draft WIR evaluation was completed in 2020. Although DOE evaluated a number of different scenarios associated with waste volumes and the fraction of key radionuclides that would end up in different waste streams, the translation of the PA results to the scope of the draft waste evaluation is not clear. The inventory in the PA was larger and encompasses the smaller inventory associated with DFLAW. For example, in the PA, DOE evaluated disposal of 130,000 canisters of vitrified waste whereas the DFLAW approach is estimated to generate 13,500 canisters. The impacts associated with some wastes do not scale linearly with volume or radioactivity. Intruder impacts generally scale linearly with activity whereas impacts to an offsite intruder through all-pathways generally scale linearly with volume. In order to properly risk-inform the review process, the baseline results are needed consistent with assumptions in the draft waste evaluation.

DOE did not include secondary wastes generated as part of waste processing within the scope of the draft waste evaluation. However, as discussed previously, if significant fractions of key radionuclides are separated or partitioned as a result of waste processing (e.g. volatilization) and those radionuclides are ultimately disposed as non-HLW then those waste streams would be within the scope of the evaluation according to NRC practice and regulation. The cumulative impact of all radioactive material disposed of at IDF needs to be considered when evaluating the performance objectives of 10 CFR Part 61.

Path Forward
Please ensure that the cumulative impact of all radioactive material disposed of at IDF is considered when evaluating the performance objectives of 10 CFR Part 61, including the doses resulting from the DFLAW inventory and the associated secondary wastes generated from processing the DFLAW inventory. If significant portions of key radionuclides end up in processing components, such as the off-gas system and those components are disposed as non-HLW, then they should be included in the results. Please also provide the waste classification results for all relevant waste streams and a demonstration that those streams will be incorporated into a solid physical form.

RAI 2-2 (Model Support for the Performance Assessment)

Comment
Additional information is needed related to demonstrate whether the conceptual and numerical models used in the performance assessment (PA) were adequately supported over the range of projected future conditions.

Basis
DOE used performance assessment modeling to integrate the results of process models and other numerical representations. The resultant system-level model was developed with the GoldSim software package. The system model was used to transfer information between models, to propagate uncertainties, and to integrate the results. The performance assessment modeling represented the present-day IDF and was used to estimate the releases of radioactivity to the environment for thousands of years into the future. Though performance assessment models cannot be validated in the traditional manner of other numerical models, performance assessment models must have adequate support of the results for the models intended purpose.

DOE presented the information used to develop the modeling in the PA report (RPP-RPT-59958, Rev. 1, 2018). The PA report was very extensive. Technical studies have been completed for decades at the Hanford site on a wide range of topics (e.g., infiltration, waste release, hydrology). Though studies have been completed to evaluate the performance of systems (e.g., engineered cover performance, unsaturated zone hydrology) and other studies are planned (e.g., glass lysimeter studies), most of the technical work has been used to develop parameter values for input to the various models, rather than to develop confirmatory information supporting the results of the PA models or the underlying conceptual models. DOE provided limited support that the conceptual models were implemented appropriately with the numerical models in the PA or that the model projections would likely bound anticipated impacts.

The IDF in its current configuration has been in existence for almost 15 years, with initial construction completed much earlier. DOE has completed numerous iterations of performance assessment calculations over the years, with the current calculations being performed prior (2018) to the recent WMA-C performance assessment that NRC reviewed (2019). In the IDF PA, DOE could not incorporate lessons learned or address recommendations made by NRC on WMA-C, one of which was for increased model support. Model support is an essential element of numerical modeling, especially of complex systems. Key intermediate results of the PA modeling include, but are not limited to, the secondary minerals that form during glass corrosion, the amount of water that contacts the wasteforms (capillary effects included), the transport time of radionuclides to the underlying aquifer, and the amount of dilution in the aquifer. Support should be provided for the key intermediate results of the numerical modeling. For example, on page 3-170 of the PA document a discussion of groundwater contaminants on the Central Plateau is provided. One of those contaminants listed is ^{90}Sr. The PA model, with appropriate changes to inputs to represent operating conditions, should be able to produce a travel time of ^{90}Sr to the water table in the approximately 40 years that was observed. It isn't clear that the existing model could generate the observed result, suggesting some aspect of infiltration, geology and properties of the unsaturated zone, unsaturated zone hydrology, or unsaturated zone geochemistry may not be appropriate. The potential non-uniqueness of calibrated inputs places greater importance on confidence-building activities.

Because of the importance of model support to the decision-making process, a dedicated plan, strategy, and document summarizing model support for the VLAW PA could enhance confidence that the numerical models adequately project or bound future impacts.

Path Forward
Please provide a summary of model confidence-building activities for the VLAW PA model or provide DOE's strategy and plan for future verification activities that are anticipated to be completed. Describe activities that have been included in the PA maintenance plan. Complete modeling with the PA model used for the VLAW PA that shows that transport of ^{90}Sr to the aquifer can be generated in the modeling results under reasonable historical operating conditions.

RAI 2-3 (PA Modeling Discretization)

Comment
From the information provided, it isn't clear that the numerical model utilized had a discretization sufficient to ensure acceptable accuracy.

Basis
STOMP modeling was used to estimate the flow of water through the different materials in the system (e.g., engineered cover, wasteforms, unsaturated zone) and to calculate the flux rates of radionuclides from the wasteforms to the unsaturated zone. Numerical modeling of materials with very different properties, especially moisture characteristic curves which describe the unsaturated flow processes, must use a discretization of the model that is fine enough such that the numerical modeling results have converged and are sufficiently precise.

Figure 4-22 from the PA document (shown below), provided the numerical grid and material properties assigned for near-field flow. It appears that some layers are not continuous in the model. It is not clear how the discretization was fine enough to ensure proper precision. The text of the PA document describes the stair-stepped discretization but does not explain how it was determined that the model discretization was sufficient. The potential impact is shown in Figure 5-11 from the PA document (partial figure provided below) where the saturation values and liquid flux rates appear to have large variations as a result of the numerical grid selection.

DOE described the discretization of the model for glass degradation on page 5-49 of the PA document. An attempt was made to strike a balance between computational time and accuracy. Modeling with a 2 cm by 2 cm grid spacing resulted in execution times of over a month per simulation. The resultant fractional release rate (FRR) from the glass was 26% higher than the coarser grid, which DOE did not believe was significant in the context of performance assessment uncertainties. A 2 cm grid spacing is extremely coarse to represent the processes occurring at the glass/engineered system interface. Layers that are tens of microns thick can significantly influence the ingress and egress of species at the interface. With only two data points with respect to the influence of grid discretization, it is unknown how much the FRR's will increase as the grid spacing is further refined and at what point further refinement will have a minimal impact.

Path Forward
Please provide additional basis for the discretization of the numerical models used in the PA to simulate near-field flow, release, and transport. Demonstrate that the simulated releases were not artificially biased by the stair-stepped grid representation of the slopes of the engineered

Figure 4-22. Two-Dimensional Vertical Cross-Section Model of Integrated Disposal Facility Showing Numerical Grid and Surface Barrier Materials.

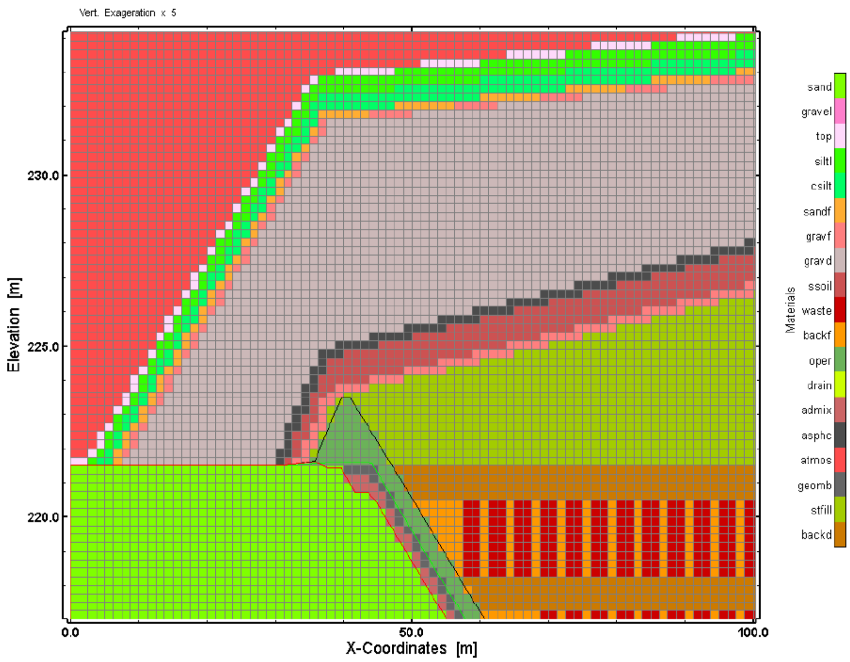
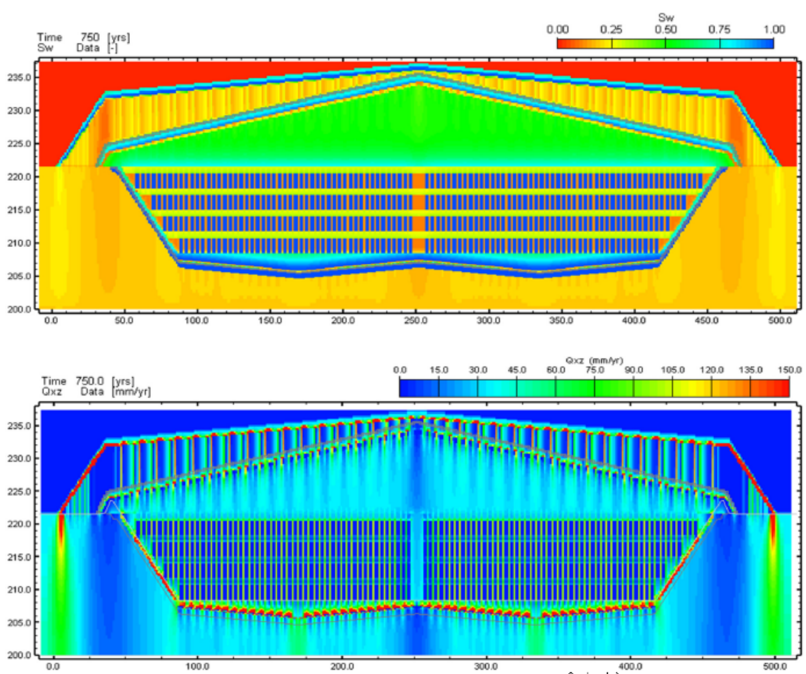


Figure 5-11. CLD3-Infiltration: 33.0 mm/yr. Spatial Distribution of Saturation (Top) and Magnitude of Water Fluxes (Middle), and Vertical Fluxes (Bottom) at 750 Years [After Cover/Liner Degradation].



cover and liner system. Provide a basis for the amount of increase, at the limit, on the FRR's from the glass as a function of refined numerical grids for release, and how the model should appropriately account for the uncertainty of a coarser numerical grid.

RAI 2-4 (Near-field and UZ Modeling Approach)

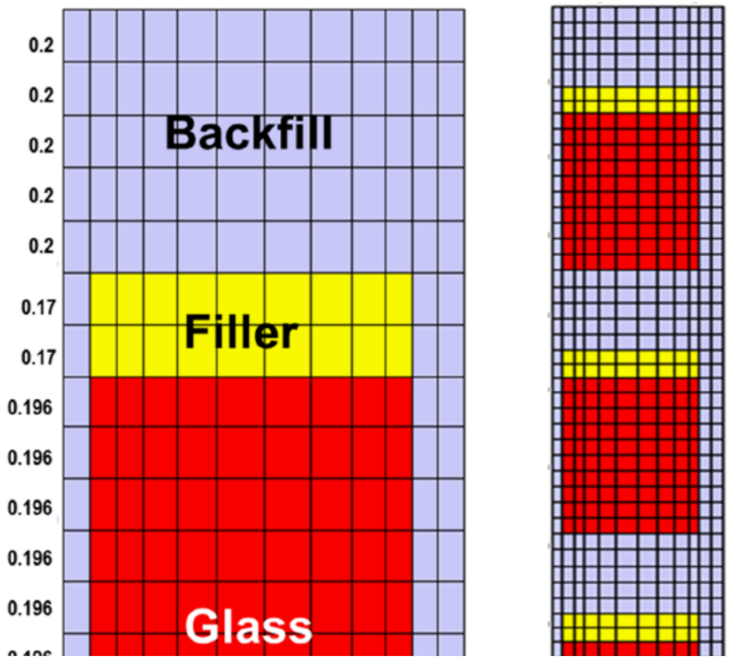
Comment
Use of uniform properties and discrete layers may not yield accurate contaminant flux rates primarily for near-field flow.

Basis
DOE assigned uniform hydrologic properties for different material layers in the simulation domain. Moisture characteristic curves (MCC's) are used to describe the relationships between saturations, pore pressures, and hydraulic conductivity. MCC's can take different forms, with some of the most common being Van Genuchten and Brooks/Corey. These relationships are derived by the fitting of empirical data to mathematical expressions. The data are generally uncertain and therefore the derived parameters are also uncertain. DOE recognized this uncertainty. For the unsaturated zone, a number of measurements were available to derive the parameters. Data on particle-size distribution, moisture retention, and saturated hydraulic conductivity (Ks) were cataloged for over 284 samples from throughout the Hanford Site, including locations in the 200 East and West Areas (PNNL-13672, "*A Catalog of Vadose Zone Hydraulic Properties for the Hanford Site*"). By comparison, very limited samples were available to define MCC's for the hydrologic layers within the boundaries of the engineered disposal facility.

Throughout the PA document (for example page 5-19), DOE indicated that highly permeable layers, such as the drainage layers, act as capillary barriers under most conditions. In addition, Table 5-14 showed that the simulated flux of water through the wasteform is approximately ten times less than the infiltration rate. Adequate model support was not provided for the simulated capillary barrier effects provided by the modeling. The figure below (Figure 5-1 from RPP-CALC-61031) show the coarse discretization and discrete layers used in the near-field modeling. It isn't clear how much of the observed reduction in flow through the wasteform is a result of the modeling approach selected.

In the real system, a fractured glass surrounded by soil will have some of the fine-grained soil that infills the fractures at the boundary of the glass (NRC notes that the glass is contained in a stainless steel canister and DOE has elected to ignore the effects of the canister). When finer-scale modeling is used, water that flows along the surface of the unfractured glass will enter the fractures and saturations can build up locally resulting in dynamic rivulets of flow which otherwise would not be observed in the coarse-scale modeling with large timesteps. In addition, the MCC parameters have natural variability which is not represented in the DOE modeling approach where uniform properties are prescribed for every cell of a given material type. Modeling of the capillary barrier effect using heterogeneous properties showed much earlier breakthrough of moisture than would be estimated using homogeneous properties (Ho and Webb, 1998).

Figure 5-1. Grids for Alternative Disposal Configurations Involving One Waste Package or Four Vertically Stacked Waste Packages.



Path Forward
Please provide information that demonstrates that the numerical grids used for near-field flow were sufficient to evaluate performance. This could include performing numerical modeling of near-field flow and transport using refined spatial and temporal representations with natural heterogeneity. . Please provide additional support for the numerical model results.

<u>RAI 2-5</u> (Disposition of Nitrate)

Comment
Previous evaluations by DOE had a large amount of nitrate ($9x10^6$ kg) that would be disposed of in IDF. The current inventory cases have values ranging from $1.6x10^5$ kg to $2.2x10^6$ kg. It is not clear how the nitrate is being removed or where it will be disposed.

Basis
Previous studies identified high nitrate feed as a potential problem for the productions of high-performance glass (RPP-54130, 2012). The Environmental Impact Statement (EIS) had about nine million kilograms of nitrate in the waste feed whereas the present PA for IDF has about $1.6x10^5$ to $2.2x10^6$ kg depending on the inventory case (DOE/EIS-0391, 2012). It wasn't clear to the staff how the nitrate is being removed or where and how it will be disposed.

Path Forward
Please describe the removal and disposition of nitrate from the waste feed for VLAW.

<u>RAI 2-6</u> (Glass Wasteform and Volatile Species Distribution)

Comment
Additional information is necessary to demonstrate that volatile species would be uniformly distributed in the glass inside a canister after cooling.

Basis
Volatilization of certain radionuclides (^{99}Tc, ^{129}I, ^{137}Cs) during glass production is well-known but also presents a challenge. DOE's assessment of glass performance in the VLAW PA assumed that volatile species retained in the glass will be uniformly distributed throughout the glass matrix. Volatilization and subsequent deposition of certain radionuclides during the cooling process may result in a non-uniform distribution of activity (e.g., ^{99}Tc) within the waste canister. Deposited radionuclides at the surface of the glass or on the walls of the canister would be available for more rapid release than the $1x10^{-7}$ to $1x10^{-8}$ 1/yr fractional release rates calculated in the PA.

The volatilization temperature of ^{99}Tc is well below the glass melting point (Tc_2O_7 boils at 311 C) such that after production and prior to solidification, ^{99}Tc may be volatized during the cooling of the glass wasteform. In addition, the glass may have a significant temperature gradient from the centerline to the walls of the canister for production-scale canisters creating complex dynamics in terms of gas and liquid flows. Experiments on the laboratory-scale demonstrated that up to 20% of volatile species were deposited on the walls of crucibles and ampules during testing (Kim, 2017). Experiments completed to examine the impact of a fire on a solidified glass canister showed that not only could cesium that had been immobilized in a glass matrix be volatized but that it ended up as a gas in the head space of the canister (PNNL-11052, 1996).

In RPP-54130, the distribution of Tc was estimated throughout a glass production system with recycle of the off-gas. Sulfate salt phases were observed on the melt pool surfaces. The salt phases showed an approximate fifty-fold enrichment in concentrations of Tc compared to the concentration in the glass suggesting that the distribution of Tc throughout the solidified glass may not be uniform.

Path Forward
Please provide additional basis that ^{99}Tc will be uniformly distributed in the glass of production-scale canisters and that deposition on the canister surfaces will not occur. If information is not available, provide plans that describe how the assumed distribution of ^{99}Tc will be verified prior to full-scale production, or show that the results of the performance assessment, including uncertainties, are not sensitive to the distribution of ^{99}Tc within the glass wasteform.

RAI 2-7 (Glass Wasteform Fractional Release Rate)

Comment
The development of the fractional release rate expressions to represent glass degradation may not adequately reflect all significant sources of uncertainty.

Basis
DOE developed expressions to represent the fractional release rate from glass as a function of different variables for different glass compositions. The expressions were based on the results of reactive transport calculations using process-level models (RPP-CALC-61031 and RPP-CALC-61192). Uncertainties were then represented in the expressions by performing regression on glass-type-specific kinetic dissolution parameters.

It isn't clear that the derived expressions reflect all important sources of uncertainty. The process for developing the expressions derived parameters from experimental data to then parameterize glass degradation expressions. However, the experimental data was in some cases highly uncertain and the uncertainty associated with fitting curves to the experimental data was not carried forward or was more limited than suggested by the data. In addition, some parameters, arguably the most important parameters, were set constant when the data do not suggest they should be fixed.

The figure below from PNNL-24615 shows the empirical data for determining the sodium ion exchange rate for LAWABP1 glass. The value was estimated to be 5.3×10^{-11} mol/m^2 s by extrapolating the data to 15 °C and assuming the value was constant in the PA (draw a line horizontally from the best fit slope at 15 °C to the y-axis). The fractional release rates are sensitive to this parameter and based on the four data points, their uncertainty, and the extrapolation of the data the basis provided by DOE was not adequate for fixing this value as a constant.

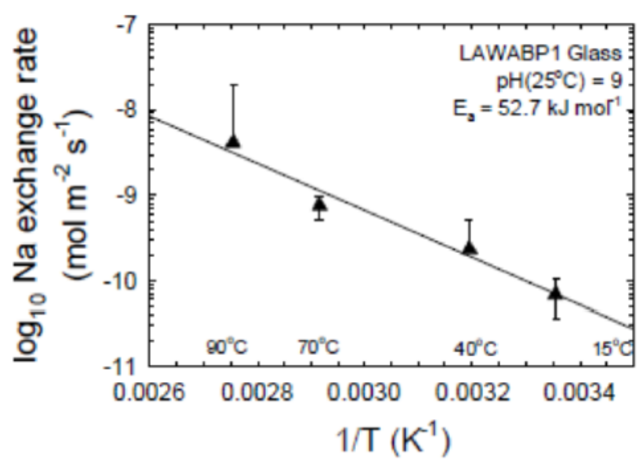


Figure 4.6. Sodium Ion Exchange Rate vs. Reciprocal Temperature for LAWABP1 Glass (McGrail et al. 2001a)

Figure 4-7 from PNNL-24615 has a relatively poor fit of the data at 40 °C, which impacts the uncertainty assigned to the coefficients. When NRC staff attempted to use a broader range for activation energy in the FRR regression equations in order to understand the significance of the uncertainty the expression yielded non-physical results. Table 4-1 from PNNL shows the derived parameters for the different glass compositions. As implemented in the PA, the three most important parameters (the forward rate constant (k_o), the glass apparent equilibrium constant (a), the Na ion-exchange rate (r_{IEX})) were all fixed as constants. Even if the previously discussed shortcomings are dismissed, some of the derived parameters have R^2 of only 0.78 on a logarithmic scale.

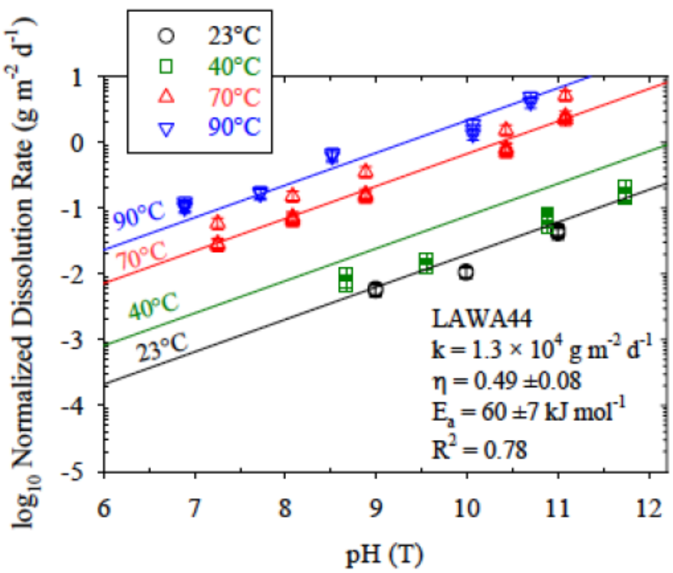


Figure 4.7. Normalized Glass Dissolution Rate, Based on Boron, as a Function of pH(T) for LAWA44 (from Pierce et al. 2004).

Table 4.1. Summary of Rate Law Parameters for LD6-5412, LAWABP1, LAWA44, LAWB45, and LAWC22 at 15 °C

	Parameters							
	$\bar{k}_0$		K_g [(a)]	η	E_a	σ	r_{IEX}	
Glass	Reported Forward Rate Constant (g/[m^2 d])	Converted[(b)] Forward Rate Constant (mol/[m^2 s])	Glass Apparent Equilibrium Constant Based on Activity Product a[SiO_2(aq)]	pH Power Law Coefficient	Glass Dissolution Activation Energy (kJ/mol)	Temkin Coefficient	Na Ion-Exchange Rate (mol/[m^2 s])	Refer
LD6-5412	9.7×10^6	1.8×10^0	1.14×10^{-4}	0.40 ±0.03	74.8 ±1.0	1	1.74×10^{-11}[(c)]	McGrail et
LAWABP1	3.4×10^6	5.7×10^{-1}	4.90×10^{-4}	0.35 ±0.03	68 ±3.0	1	3.4×10^{-11}	McGra (200
LAWA44	1.3×10^4 ($R^2 = 0.78$)	2.2×10^{-3}	1.87×10^{-3} ($R^2 = 0.95$)	0.49 ±0.08	60 ±7	1	5.3×10^{-11}	Pierce et a
LAWB45	1.6×10^4 ($R^2 = 0.96$)	3.0×10^{-3}	1.79×10^{-3} ($R^2 = 0.78$)	0.34 ±0.03	53 ±3	1	0.0×10^{0}[(d)]	Pierce et a
LAWC22	1.0×10^5 ($R^2 = 0.96$)	1.8×10^{-2}	1.80×10^{-3} ($R^2 = 0.94$)	0.42 ±0.02	64 ±2	1	1.2×10^{-10}	Pierce et a

Additional uncertainties not reflected in the empirical FRR expressions include the assumption that the properties of laboratory-scale produced glass will be identical to production-scale glass, the Temkin coefficient was assumed to be one without empirical data to justify the selection, and the uncertainties associated with assumed secondary mineral formation in the process modeling (i.e., the need to assume Chalcedony formation without it being empirically observed).

In the PA, DOE properly took the information from the underlying technical reports, however it is not clear that in the underlying technical reports DOE properly characterized and reflected the uncertainties in the empirical data and the assumed theoretical degradation expressions.

Path Forward
Please address the treatment of the uncertainties described in the basis portion of this comment with respect to development of glass fractional release rate expressions for the PA. If necessary, revise the expressions and generate new PA results that reflect the full range of uncertainty in the glass degradation rates. NRC staff understands that a variety of sensitivity cases were examined by DOE for various uncertainties associated with glass degradation. However, most of those examinations were one-at-a-time evaluations that can be mathematically compounded but may not yield a fair assessment of the importance of these uncertainties combined with other uncertainties raised by this RAI package. If appropriate, the expressions used in the system model uncertainty analyses should be revised.

The uncertainty associated with the performance of production-scale glass may be addressed by providing DOE's performance verification plan to assure the quality and performance of production-scale glass.

RAI 2-8 (Glass Cracking)

Comment
Additional information is needed on the basis for the assumed factor of 10 increase in specific surface area of the glass to account for cracking.

Basis
Glass release rates for many glass formulations, although not all, are directly proportional to the specific surface area (m^2/g). To account for potential cracking of the glass during cooling and handling, DOE increase the geometric surface area of the glass by a factor of 10. In sensitivity cases (termed RSA) examined in the PA document, DOE further increased the specific surface area by a factor of 10 and showed a proportional increase in the fractional release rate for two of three glass formulations. The results clearly show the influence of specific surface area but do not address what the value will be for production-scale glass canisters which will have different glass compositions.

Many of the reports dealing with cracking of glass are older (e.g., PNL-5947) but do have useful information to try to evaluate the DOE assumed value of 10. PNL-5947 had a maximum adjusted relative surface area of 65 with many of the reported values greater than 10. Table 1 show below (from ML040130177) provides some observed values from the literature (including values from PNL-5947).

Table 1. Summary of studies examining surface area increases due to thermal fracturing

Glass Composition	Glass block size (relative)	Surface Area Increase (relative to unfractured glass)	Reference
SRL211	large-scale	2 - 40	Smith and Baxter 1981
SRL211, SRL131	large-scale	7 - 18	Peters and Slate 1981
SRL211, SRL131	small-scale	0 - 18	Peters and Slate 1981
borosilicate	large-scale	9.0 - 16.3	Laude et al. 1982
SRL165	large-scale	25 - 35	Bickford and Pellarin 1987
borosilicate	small- to large scale	1.1 - 86	Faletti and Ethridge 1988
borosilicate	medium-scale	2.0 - 10	Lutze et al. 1986
R7T7	small-scale, 1:10	10 -12	Vernaz and Godon 1991
PNL76-375	large-scale	8 - 45	Martin 1985
PNL76-375	small-scale	1.1 - 12	Martin 1985
borosilicate	medium-scale	not measured	Keinzler 1989
BRET HLW borosilicate glass	medium-scale	not measured	Farnsworth et al. 1985

NRC staff were not able to determine the canister filling and cooling procedures that DOE would use to determine the appropriateness of the assumed factor of 10. Rapid cooling can contribute to more significant crack formation, however slow cooling can impact the quality and performance of glass that is produced. NRC staff were also not able to identify verification plans for the specific surface area in production-scale glass canisters.

Path Forward

Please provide additional technical basis for the assumed effect of cracking on glass specific surface area, utilize a bounding value, or provide plans to verify the assumed value in production-scale glass canisters. Please describe the cooling cycles anticipated to be used during glass production.

<u>RAI 2-9</u> (Glass Stage III)

Comment
Additional information is needed to support the basis that Stage III glass corrosion will not occur for disposal of vitrified waste at IDF.

Basis
Stage III behavior in glass corrosion or degradation is believed to occur late in the reaction sequence as a result of the formation of zeolites and other phases which deplete silicon and other species in solution faster than which those species are added to solution. The phenomenon is not well-understood but has been associated with higher temperatures and closed systems. DOE indicated that Stage III conditions are not likely at IDF because it will be low temperature (15°C) and an open system. The temperature of the system is fairly certain unless degrading organic matter or heat-generating waste were to be disposed in the facility. The openness of the system is more uncertain. The impermeable asphalt layer combined with the Geosynthetic Composite Layer (GCL) liner under the facility can result in very low flow rates and conditions that could approximate a closed system. In addition, the fluid that reacts with the glass in the simulations was not a fluid that had reacted with the overlying engineered barriers but was a Hanford groundwater composition.

The modeling of glass degradation used Chalcedony as a kinetic control because the results of chemical reaction progress modeling were found to agree reasonably well with experimental results involving several different glass types at 90°C if chalcedony was assumed to form (PNNL-20781, 2011); however, chalcedony has not been directly detected as an alteration product of glass corrosion (PNNL-24615, 2015). In other words, Chalcedony and the assumed kinetic controls is a calibration parameter used by DOE to fit the empirical data. That does not justify the assumption of the absence of Stage III behavior. In the assessment of different secondary mineral formation, DOE didn't consider minerals that could conceivably generate Stage III glass corrosion because such behavior was considered unlikely under IDF-relevant conditions (PNNL-24615, 2015).

Path Forward
Please demonstrate Stage III behavior is unlikely taking into consideration the potential temperature ranges and degrees of openness of the disposal system. This may be done by performing geochemical modeling or generating experimental data using relevant fluid compositions and appropriate minerals. PA calculations could be used to address the significance of the formation of Stage III behavior.

<u>RAI 2-10</u> (Volatile Species and Glass)

Comment

One of the most important aspects of uncertainty associated with the VLAW PA appears to be the assumed partitioning of various species, especially volatile species, between different waste types. Additional information is needed to support the amount of volatile species that will be retained in glass (Case 7 – the base case).

Basis

Initial testing of glass production determined a low retention rate of volatile species (Tc, I, Cs). The glass process was modified to recycle the off gas back into the glass feed to increase the loading of volatile species in the glass. The modifications were effective in increasing Tc retention percentages from generally less than 50% to approximately 75% (Pegg, 2015). The Tc retention in the glass assumption in the base case PA is approximately 99.9% - essentially all of the Tc ultimately is disposed in the glass waste form.

Detailed evaluation of Tc recycle was completed by Catholic University's Vitreous State Laboratory (RPP-54130, 2012). The glass production system uses a wet electrostatic precipitator (WESP) among other components. By measuring the amount of Tc in the WESP effluents, DOE estimated that the Tc retention was 99.8%. The Case 7 inventory partitioning of Tc is consistent with the WESP emissions. The amount of Tc observed in the glass was 68%. In addition, the mass balance of Tc across all tests averaged approximately 90%. DOE indicated that a significant amount of technetium was held up in the system during testing, particularly in the WESP internals, the film cooler, and the transition line. This material was therefore not available for recycle and incorporation into the glass. All of the Tc that gets deposited and retained in the various processing equipment eventually has to be disposed and those components are not going to be disposed as glass. Sulfate salt phases were observed on the melt pool surface after two of the tests. The salt phase showed an approximate fifty-fold enrichment in technetium over the glass; rhenium and halides also showed significant enrichments. These phases may be considered as being "retained" by the glass but would not likely have the same release properties as the glass.

DOE's assumed retention of Tc in glass for Case 7 would appear to be unrealistically optimistic. Case 10A had approximately 32% of the Tc in the glass reflecting no recycling of the off gas, which is likely to be unrealistically pessimistic if recycling is used. As discussed above, the mass balance was limited to approximately 90% and the amount of Tc observed to be in the glass was approximately 70%. Appropriate base case values for Tc retention in glass would appear to be in the 70 to 95% range.

The design, operation, and especially the reliability of the off-gas system with recycle would appear to be extremely important to justify the assumed Tc retention in the glass. Staff did not identify information to support the assumed 100% reliability of the off-gas system. System downtime would significantly contribute to not being able to achieve extremely high Tc retention in the glass.

Path Forward
Please provide additional basis for the base case inventory or revise the base case inventory to be consistent with the observed testing data. For the base case inventory, DOE should observe mass balance, glass concentrations of volatile species, concentration of species in salt phases, and reliability of the off-gas system. For the base case inventory DOE should also account for the disposal of volatile species that build up in the system components and in what form they will be disposed.

RAI 2-11 (Comparison of STOMP and GWB)

Comment
Additional information is needed to support why some comparison cases for glass release rates generated with STOMP and Geochemist's Workbench (GWB) have not applicable (NA) entries.

Basis
In the evaluation of sensitivity cases for release from the glass wasteform, DOE used both STOMP and GWB to estimate fractional release rates. In most cases the agreement between the two programs was reasonable (see Table 5-4 and Table 5-9 of the PA document). However, in a number of entries in the table results were only provided for one model. Because these types of calculations can have large uncertainties it is good practice to calculate results with two models. Additional information should be provided to provide the basis for only using one model for certain entries.

Path Forward
Please provide the results to complete Tables 5-4 and 5-9 of the PA or describe why the use of one model was appropriate for certain entries. Confirm that the missing entries were not a result of a lack of numerical convergence or similar issues.

RAI 2-12 (Sensitivity and Uncertainty Analyses)

Comment
The sensitivity and uncertainty analyses presented by DOE did not include some aspects that may be important to risk-inform the review process and to determine if the relevant criteria are likely to be met.

Basis
In section 5.2.3 of the PA document, DOE described the uncertainty and sensitivity analyses completed for the draft waste evaluation for VLAW (disposal in the near surface at IDF). Key uncertainties identified by DOE included release rates, recharge rates, vadose zone hydraulic properties, vadose zone transport properties, saturated zone hydraulic properties, and waste loading configuration in the disposal facility. Sensitivity and uncertainty analyses were completed with the deterministic process models as well as the probabilistic system model. The types of uncertainties examined were reasonable and consistent with NRC's understanding of the system. However, there were some uncertainties that were not included within the scope of the evaluation that may be important to understand in order to risk-inform the review and

determine if the criteria will be met with reasonable expectation (DOE) or reasonable assurance (NRC).

In section 5.1 of the PA document, DOE described a number of different analyses cases completed to evaluate near-field flow and source-term release. The computational results were presented in RPP-CALC-61029. DOE examined the timing of engineered layer "failures" by examining a case where the surface cover and liner had a step change in properties at 500 years post-closure. While this is an appropriate case to examine, given the uncertainties being addressed, the evaluation is incomplete. The performance of the engineered cover is reliant to a large extent on an asphalt layer whereas the performance of the liner is reliant on a GCL. Because of numerical difficulties, the hydraulic conductivity of the asphalt layer was only increased an order of magnitude in a step manner. The properties of fresh, intact asphalt compared to aged, cracked asphalt would be expected to differ by much more than an order of magnitude. In addition, because of the different materials involved, there is limited expectation that the different layers would "fail" at the same time or same rate. In general, engineered layers closer to the land surface experience a more diverse set of processes and events that lead to more rapid alteration by nature. It would be reasonable to examine a case with a degraded cover and an indefinitely performing liner system. NRC was not able to find information describing the drain and sump systems to evaluate their propensity for plugging or decrease in performance.

In section 5.1.2 of the PA document, DOE examined sensitivity of glass release rates to various parameters. One case, termed HYDRL, examined the effect of changes to hydraulic properties of the glass wasteform. Moisture characteristic curves (MCC's) can have a significant impact on release rates if differing materials are present and simulation of capillary barrier effects occurs. The HYDRL and combined cases should be expanded to include the impact of uncertainties in MCC's. There may be reduced sensitivity of the results to changes in some inputs, such as recharge, due to the masking effect of the asphalt layer performance and the MCC's assigned. The SRMN cases evaluated the impact of the secondary mineral formation network. The SRMN cases showed a large impact from uncertainty in what minerals form and therefore their thermodynamic properties. The SRMN cases should be expanded and, if possible, supported by information from experiments and the literature. The discussion in the PA indicates that selection of chalcedony (see RAI 2-7) was essentially a calibration as it was used to match empirical results, and actual phases observed in experiments were not used because acceptable glass degradation rates could not be achieved. This is a source of uncertainty within the modeling and the simulated degradation rates in the PA are essentially an extrapolation of short-term empirical observations. This type of uncertainty should be reflected in the base case results, or the potential impact on the base case results communicated to decisionmakers, otherwise a false sense of confidence may be assigned to the assurance as to whether the regulatory criteria will be achieved.

DOE indicated that when considered together, multiple sources of conceptual and parameter uncertainty with respect to glass release modeling may thus have a cancelling effect in predictive models of glass corrosion. The overall impact of these uncertainties on fractional releases may consequently be relatively small. It is unclear how this conclusion was arrived at unless inverse correlations between the relevant uncertainties were observed (and this is not common). Uncertainties will propagate and expand the potential range of outcomes. A probabilistic assessment of glass release rate uncertainties would help better define the range of uncertainty in glass release rates.

A large source of uncertainty in the performance of the IDF is from the inventory splits, or the fraction of key radionuclides that end up in different waste streams. DOE's base case (Case 7) has a very high percentage of the ^{99}Tc that ends up in the glass wasteform because of recycling. By comparison, cases 10A and 10B have a low percentage of ^{99}Tc that ends up in the glass wasteform as a result of volatilization during processing. This uncertainty is the only uncertainty discussed that by itself that can swing the results from compliance to non-compliance. This uncertainty was not included in the global uncertainty analyses completed with the system model. The significance of an individual uncertainty depends on all other uncertainties, how they impact the results, and how close the results are to the regulatory standards. If a key uncertainty is identified but left out of the comprehensive uncertainty analyses the importance of individual uncertainties may be misinterpreted.

DOE examined some types of inventory uncertainties through special cases. The magnitude of the inventory in the base case did not reflect the uncertainty in the inventory that would be generated and processed into the various waste types. NRC had made various comments on the development of inventory values for WMA-C, some of which are also relevant to this draft WIR evaluation (ML20128J832). While it is true that this type of uncertainty has a relatively linear impact on the dose results and can be easily projected, this type of uncertainty directly compounds with other types of uncertainties in the PA and increases the range of potential outcomes thereby decreasing the certainty with which demonstration of compliance with the criteria can be achieved.

Path Forward
Please expand the sensitivity and uncertainty analyses to include the items discussed above in the basis part of this comment (e.g., additional glass release uncertainties, inventory splits, inventory uncertainties).

<u>RAI 2-13</u> (Quality Assurance)

Comment
Some aspects of the quality assurance program were not clear from the documentation provided.

Basis
DOE provided detailed information on most aspects of the quality assurance program applied to the development of the analyses supporting the draft waste evaluation for VLAW. However, a few aspects of the quality assurance program were not clear. The quality assurance status and controlled use of the major software or computer programs was demonstrated (e.g., GoldSim, STOMP). The quality assurance status or verification activities for ancillary software was not provided in all instances. For example, the Hanford Defined Waste model (HDW) was used as part of the inventory development process. Some components of the HDW were previously assessed and found to contain significant errors, but other components of the HDW were not verified (ML20128J832). The thermodynamic database used in the geochemical modeling for

glass degradation (thermo.com.V8.R6+.tdat) came from Lawrence Livermore National Laboratory. The qualification status of that database was not clear.

Verification information was provided for STOMP verification and test cases that demonstrated select aspects of the software. However, it wasn't clear from the documentation provided how those verification activities demonstrate or verify the correct functioning of the software for the key aspects of the performance assessment, namely the glass degradation and release rate calculations and the unsaturated flow phenomena especially the capillary barrier effects. Verification of unsaturated flow phenomena in general is not the same as verifying that STOMP correctly produces results for very dissimilar materials using a coarse numerical grid.

Path Forward
Please provide the qualification status of software and databases that supply information to the performance assessment calculations, or the plans to determine the qualification status of the referenced software and databases. Please provide the verification results or plans for verification of the release rate and unsaturated flow phenomena simulated by STOMP for glass degradation as applicable to the performance assessment.

RAI 2-14 (Geologic Uncertainty)

Comment
The basis for the interpretation of the geology underlying the footprint of the IDF that removed the Ringold E formation is not clear.

Basis
DOE's previous interpretation of the geology underlying the IDF had a layer termed the Ringold E present in the northwest corner of the footprint of the facility (see the figures below). DOE explained that some geologic information was reinterpreted, and the geologic framework model was revised. The data shows quantitative information to suggest the boundary of the layer is somewhere between the boreholes (see first figure below). If the modeling was revised, it isn't clear how the change in the boundary of the unit was validated in the absence of additional data. The significance of the unit is that the Ringold E is much less permeable than the Hanford unit such that fluxes of contaminants into the unit experience lower dilution and therefore result in higher concentrations. If groundwater protection standards apply to all geologic units, then it may be more difficult to demonstrate that groundwater protection standards have been met.

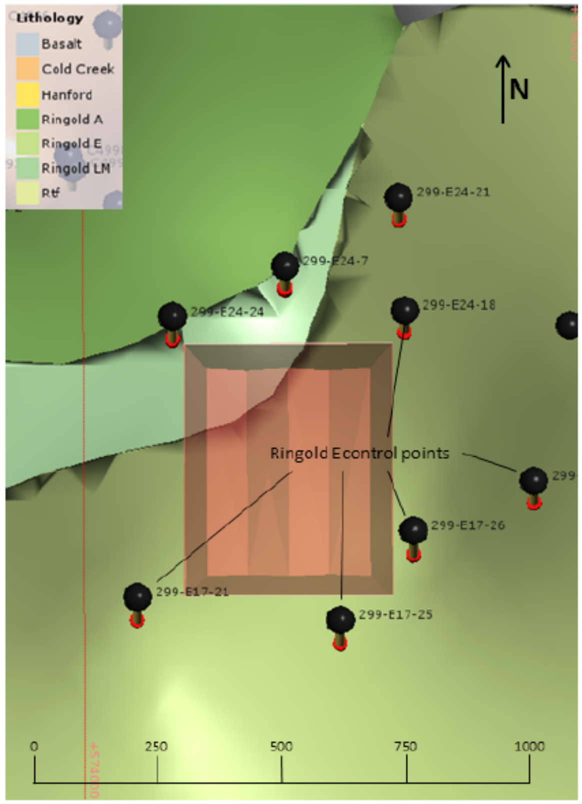
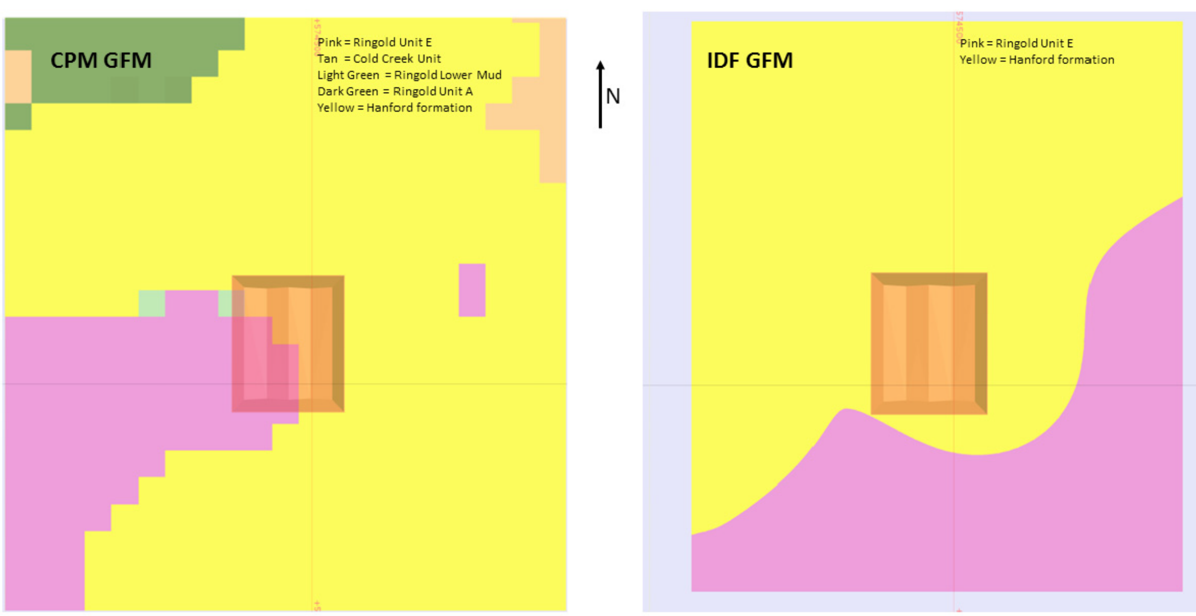


Figure 3-62. Plan View Comparison of Central Plateau and Integrated Disposal Facility Geologic Framework Models at Long-Term Steady-State Water Table (119.5 meters).



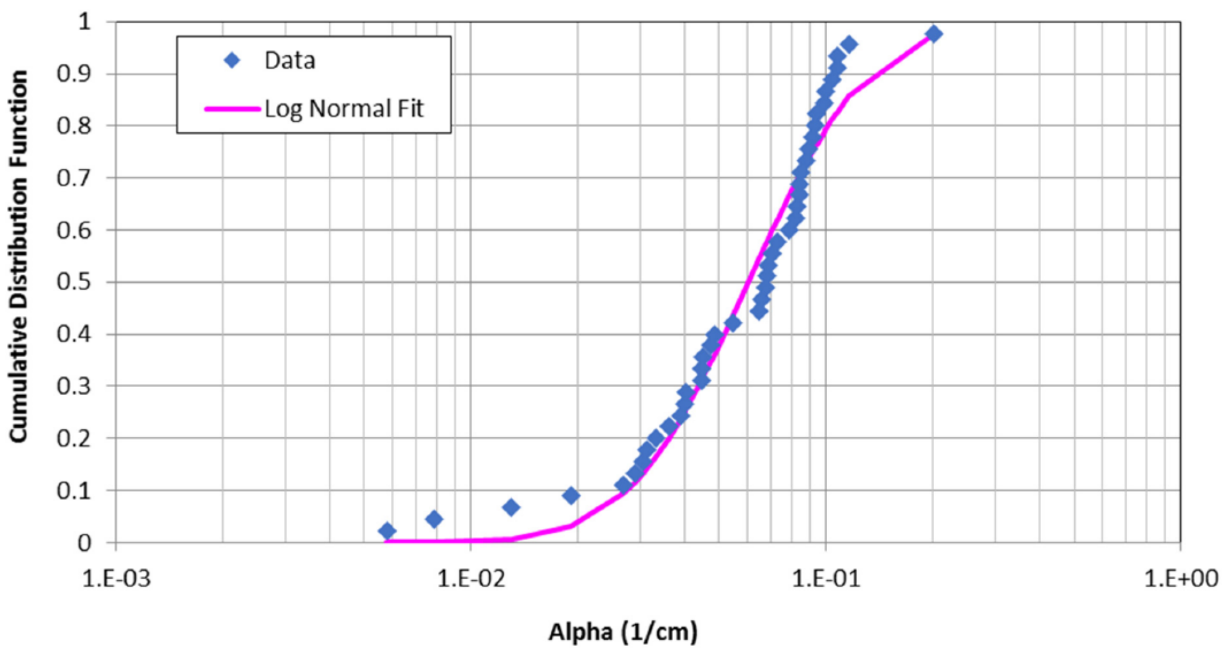
Path Forward
Please provide additional basis for the reinterpretation of the location of the Ringold E and indicate whether groundwater protection standards apply to this unit.

<u>RAI 2-15</u> (Vadose Zone Parameters)

Comment
The log-normal fit of the van Genuchten alpha parameter for the H2 unit does not appear to represent the data well at the tails of the distribution.

Basis
To develop uncertainty distributions of unsaturated flow parameters the empirical data were fit with statistical distributions. The van Genuchten alpha parameter for the H2 unit was assigned a log-normal distribution. The fit of the equation to the data showed deviations at the tails of the distributions (See Figure 4-56 from the PA document provided below).

Figure 4-56. Fitted Log-Normal Distribution to the van Genuchten "Alpha" Parameter Dataset Used for the H2 Unit.

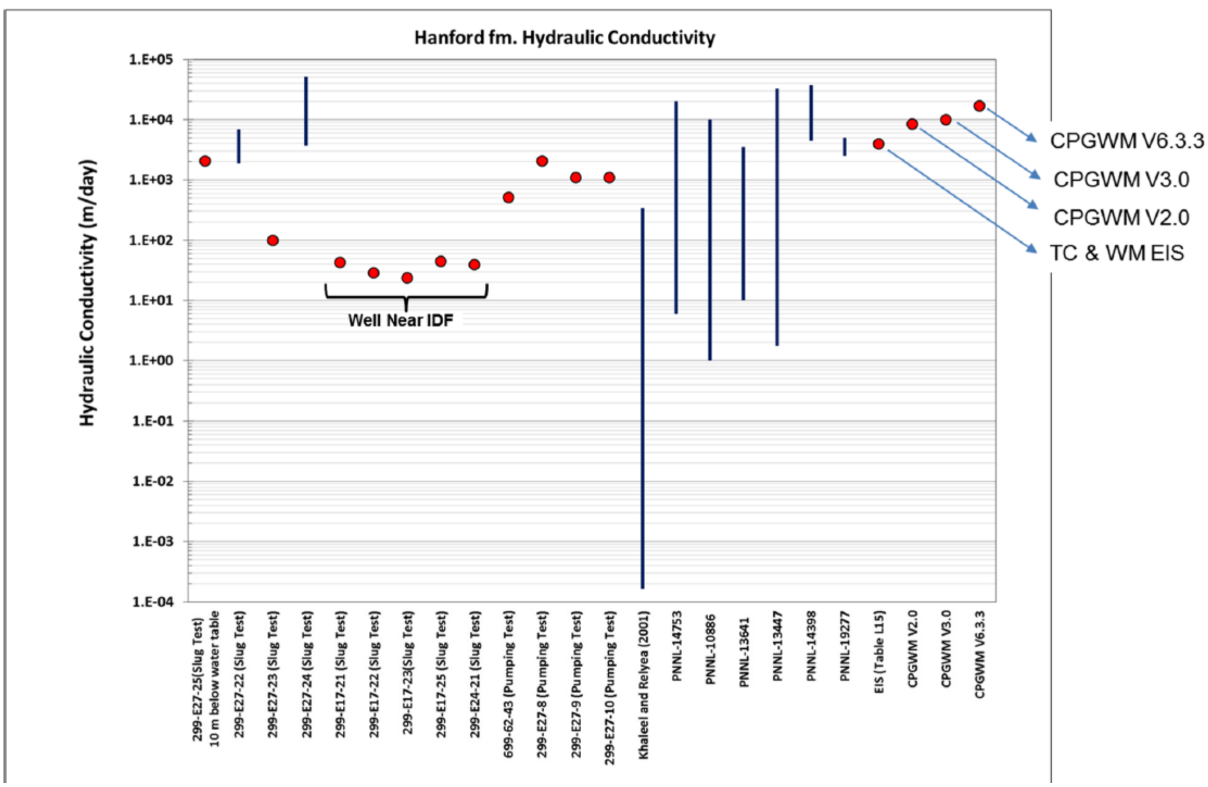


Path Forward
Please discuss the implications of the deviation in the fitted distributions from the underlying data.

RAI 2-16 (Saturated Zone Hydraulic Conductivity)

Comment
The changes to the estimated hydraulic conductivity values for the saturated zone over time suggest the base case value best estimate may not be reliable.

Basis
The saturated zone hydraulic conductivity, or rather the product of the saturated zone hydraulic conductivity and the gradient, is a key factor with respect to the reduction of risk at the Hanford site. Flux rates of water through the unsaturated zone are relatively low whereas the flow rate of water through the saturated zone are comparably high. This creates a large dilution effect on contaminant fluxes thereby reducing risk to an offsite receptor. DOE has investigated the saturated hydraulic conductivity over numerous decades and produced the very detailed figure (Figure 4-94 in the PA document) shown below. The figure highlights two important features. First, that the local values can deviate significantly from the global values and secondly, that the estimated values have been highly volatile over time. Since completion of the TC & WM EIS the values have increased almost an order of magnitude. While the specific discharge values may have increased by a lessor amount than estimates of the saturated zone hydraulic conductivity, it isn't apparent why the base case value is thought to be reliable and unlikely to change given the past changes as more data has been collected and additional modeling has been completed.



Path Forward
Please discuss confidence building activities to support the base case saturated zone hydraulic conductivity values assigned. Please discuss plans to verify the base case saturated zone hydraulic conductivity.

<u>**RAI 2-17**</u> (Intruder)

Comment
DOE provided the dose result to an inadvertent intruder resulting from the average waste but did not provide the range of potential intruder doses that could be anticipated. NRC provided a number of comments and recommendations associated with intruder analyses for WMA-C that DOE was not able to address in this draft WIR evaluation due to timing differences.

Basis
DOE communicated the dose to an inadvertent intruder resulting from unanticipated future intrusion into a particular waste type. However, DOE did not communicate the range of inadvertent intruder doses that could be anticipated for each waste stream.

The waste will be disposed in the IDF which is a near-surface disposal facility. However, the IDF is a "deep" near-surface disposal facility such that an excavation scenario is highly unlikely. The intruder scenario evaluated was a drilling scenario where the installation of a well to recover resources results in the extraction of geologic materials and some waste in the drill cuttings. The amount of waste potentially disturbed is much lower than what is expected in an excavation scenario. Whereas in the excavation scenario, the concentration of radionuclides in the disturbed waste may more closely approximate the average concentration, for a discrete event like drilling the concentration in the disturbed waste is likely to reflect the variability of radionuclide concentrations in the waste. The risk is a product of probability and consequence, and the probability goes down as drilling at locations of less likely concentrations occur (e.g., the most highly concentrated waste). NRC and most other regulators impose limits that reflect the average concentration of the waste but also impose requirements as to how much averaging can be used (e.g., the NRC's Branch Technical Position on Concentration Averaging and Encapsulation (ML12254B065)).

NRC issued the technical evaluation report (TER) for review of the draft waste evaluation for WMA-C in May 2020 (ML20128J832). In that report, NRC provided recommendations and comments on the inadvertent intruder analyses. The inadvertent intruder analyses for VLAW used essentially an identical approach, but the VLAW analyses were completed prior to the NRC TER, and therefore, DOE could not address NRC comments. In addition, those comments were not risk-significant in the context of WMA-C but could be risk-significant with respect to the VLAW analyses.

Path Forward
Please provide the range of dose impacts to an inadvertent intruder from each waste stream disposed in the IDF that is within the scope of the draft waste evaluation as discussed previously in this document. The assessment should consider risk-significant comments made on the WMA-C intruder assessment, if applicable to the intruder assessment for VLAW.

RAI 2-18 (^{90}Sr Inventory Uncertainty)

Comment
Additional information is needed regarding the uncertainty in the ^{90}Sr inventory estimate and how the inventory uncertainties are propagated into the GoldSim model.

Basis
Strontium-90 is a key radionuclide, especially for the intruder scenario. The projected chronic dose to the hypothetical human intruder 100 years after IDF closure under the rural pasture resident scenario is 43.3 mrem/yr (as compared to DOE's performance metric of 100 mrem/yr). The peak dose is driven by the milk ingestion pathway (40.5 mrem/yr). The total dose is principally due to ^{90}Sr and ^{99}Tc, which contribute 29.8 mrem/yr and 10.4 mrem/yr, respectively, to the total dose (page 7-38 of the PA document).

There is uncertainty in the inventory of ^{90}Sr within the Best Basis Inventory (BBI). The PA document (page 3-232) indicates that ^{90}Sr estimates in the BBI could be off by as much as 80 percent. However, it is unclear how uncertainty in ^{90}Sr inventory coming from the BBI model is included and propagated into the GoldSim model. In a public teleconference with DOE on September 28, 2020 (ML20311A202), DOE indicated that because of the direct correlation between inventory and dose they intentionally did not include inventory uncertainty in the GoldSim model to be able to see the effects of other aspects of uncertainty more readily. Instead, DOE indicated that they evaluate uncertainty in risk with inventory in RPP-CALC-63176. Specifically, the PA results are scaled to forecast the impact to groundwater in this tool instead of using GoldSim model. DOE also indicated that for the intruder analysis, in RPP-CALC-61254 Rev 3, DOE analyzed different ^{90}Sr levels and how it influences the need for intruder barriers.

Path Forward
Please provide a description of how ^{90}Sr inventory uncertainty impacts the dose to the inadvertent intruder. Please provide the reference RPP-CALC-61254, Rev 3.

RAI 2-19 (Releases from the ETF-LSW Waste)

Comment
Additional information is needed on the modeled release of ^{129}I and ^{99}Tc from the ETF-LSW waste.

Basis
Several aspects of the modeling used for the release from the ETF-LSW waste could result in the underestimation of the potential release. The ETF-LSW waste originates from the treatment of liquid waste from WTP operations, including the liquid effluent from the melter primary off-gas treatment system and the LAW vitrification secondary off-gas/vessel vent treatment system.

As described above in RAI 1-3, there is uncertainty in how much of key radionuclides, such as ^{129}I and ^{99}Tc, will end up in the ETF-LSW waste stream due to uncertainty in how well these radionuclides will be incorporated into the glass. For example, in Case 7, the inventories of ^{129}I and ^{99}Tc in the ETF-LSW waste are assumed to be a small percentage of the total, while in Cases 10A and 10B, the inventories of ^{129}I and ^{99}Tc are assumed to be much higher. In the GoldSim model, the calculated fractional release rates of ^{129}I and ^{99}Tc from the ETF waste on a unit inventory basis (i.e., “fractional_release_rate_all”) are approximately an order of magnitude less than the release rates from other waste streams, such as the ion exchange waste stream and the waste streams that are encapsulated in cementitious mortar or paste.

The basis for the best-estimate parameter values assumed for the release from the solidified ETF-LSW waste is provided in PNNL-25194 (2016) and is summarized in the PA document. One key modeling assumption for the release from the ETF-LSW stream, that appears to be non-conservative, is the assumed probability distribution of the effective diffusivity coefficient. A log-uniform distribution with a range of 1.8×10^{-13} cm^2/s to 5.5×10^{-8} cm^2/s was assumed for the ETF-LSW waste, while the probability distribution assumed for the diffusivity for the other cementitious materials was higher. The data cited in PNNL-25194 includes effective coefficient data for iodine and technetium, which are expected to sorb slightly, as well as data from sodium, nitrate, and nitrite. The measured diffusivity coefficients at the low end of the observed range were for iodine and technetium and likely include sorption. This item was also identified by the Low-Level Waste Disposal Facility Federal Review Group (LFRG) as Issue Number IDF-S19-PA12-05. The corrective action stated for this item was to add a clarifying discussion to Section 6.3.2.3 of the PA document. This discussion acknowledges that the effective diffusivities include sorption and states that the minimum value of the distribution was based on the assumption that the Kd of the species in question (iodine and technetium) is equal to 0. The assumption that the Kd values for these elements is equal to 0 is inconsistent with the assumed Kd values in the model. Additionally, the response to the LFRG comment stated that “the ETF-LSW waste form is a negligible contributor to dose as shown in Figure 6-63”. This figure reference appears to be incorrect. However, Figure 6-58 does show the dose contribution from the ETF-LSW waste being less when the Case 7 inventory is assumed. This figure does not provide any information as to the relative dose contribution of the ETF-LSW waste if a higher inventory of ^{129}I and ^{99}Tc end up in this waste stream.

Additionally, the diffusive length assumed in the GoldSim model for the ETF-LSW waste (0.2 m) is approximately an order of magnitude longer than the diffusive lengths for the other waste forms, but it is not clear what the basis is for this waste length being longer. It also is not clear what effect this assumption has on the modeled diffusive fluxes of the radionuclides out of the wasteform and the potential dose from this release.

Path Forward

Please provide additional information on the expected fractional release rate for ^{129}I and ^{99}Tc for the ETF-LSW wasteform and describe whether the modeled performance is consistent with this wasteform’s expected performance as compared to the other cementitious wasteforms included in the PA. Consider providing an evaluation of the potential fractional release and the dose from the ETF waste using an effective diffusion coefficient value that does not include sorption. Also consider providing an evaluation of the potential dose if the inventory in this waste stream is higher than assumed in Case 7 (i.e., a Case 10A/10B inventory) using a revised effective diffusion coefficient value. Please provide additional information on the basis for the diffusive length assumed for the ETF-LSW waste. If this diffusive length was an error, provide an updated evaluation of the fractional release rate from the ETF-LSW waste that incorporates a corrected value for the diffusive length. Consider providing an updated value for the diffusive length, if appropriate, in the evaluation requested above.

RAI 2-20 (I Sorption on the SSW-GAC and SSW-AGM Wasteforms)

Comment
Additional information is needed for the assumed sorption of ^{129}I on the SSW-GAC (Granular Activated Carbon) and SSW-AgM (Silver Mordenite) wasteforms.

Basis
The modeled release of ^{129}I from the SSW-GAC and SSW-AgM wasteforms is significantly reduced by the assumed sorption of the ^{129}I onto these wasteforms. The assumed distributions of the distribution coefficient (Kd) values for ^{129}I on the SSW-GAC and SSW-AgM wasteforms are much larger than the distributions assumed for the other cementitious wasteforms in this model and are much larger than typically observed Kd values for ^{129}I on cementitious materials. The Kd values assumed for ^{129}I for the SSW-GAC and the SSW-AgM wasteforms are based on an average of a Kd value for sorption of ^{129}I onto a typical cementitious material and the sorption of ^{129}I onto GAC or AgM. Although the sorption of the ^{129}I onto the GAC and AgM contained in these wasteforms could improve the overall sorption of ^{129}I on these wasteforms compared to typical cementitious wasteforms, the sorption on the combined material might not be equal to the average sorption of its components due to changes in the chemical environment in the wasteform. It is not clear what support exists for the Kd values assumed in the GoldSim Model, and it is not clear if analytical measurements of the ^{129}I sorption on simulated wasteforms comparable to the SSW-GAC and SSW-AgM wasteforms have been performed.

The sorptive capacity of ^{129}I on the waste forms for the SSW-GAC and the SSW-AgM waste streams was identified during the LFRG review as issue number ISF-S09-PA03-02. As a corrective action/resolution for this issue, a sensitivity case was performed for the air pathway that used Kd values that were lower than were previously assumed but were still much higher than for typical cementitious materials. However, a sensitivity analysis was not performed to evaluate the potential effect of less sorption of ^{129}I on these wasteforms on the dose. The corrective action/resolution states that the "full range of uncertainty in the Kd values was considered in the probabilistic analysis" for the groundwater pathway. However, the entire range of Kd values included in the GoldSim model is high and this range does not appear to capture the potential for the sorption of ^{129}I to be much lower than assumed. The corrective action/resolution to this issue also stated that R&D activities were expected to be performed in the future to characterize the Kd on the carbon media and silver mordenite waste forms.

Path Forward
Please provide additional information to support the assumed Kd values for ^{129}I on the SSW-GAC and the SSW-AgM wasteforms, and if any R&D activities have taken place on this topic to date, provide the results of those activities. Alternatively, provide a sensitivity analysis showing the effect on the release rates and potential dose from ^{129}I from these two wasteforms if the sorption on these wasteforms is lower than assumed in the model.

RAI 2-21 (Releases from Cementitious Wasteforms)

Comment
More information is needed on the process for determining and evaluating the final cementitious grout specifications for waste streams stabilized with cementitious grout.

Basis
In the PA document, DOE indicated that secondary waste streams generated as the result of WTP operations will be solidified or encapsulated using cementitious materials. The PA document states that the "final specification of the solidification and encapsulation matrices, however, are currently uncertain and will eventually be selected on the bases of several performance factors: adequate mechanical strength for handling, transportation and emplacement; compatibility with other engineered barriers and waste forms in the IDF; [and] limited rates of release of COPCs into the IDF." The PA document further states "[b]ecause the details of the SSW cementitious grout mix specification(s) and final disposal configuration for SSW have not been defined, the SSW data package relied on available information from existing studies of cementitious materials considered representative of mixes that may be used for SSW encapsulation and/or solidification". It is not clear if any further information on the proposed specifications for the cementitious matrices has been developed since the PA document was written.

Additionally, in the LFRG review, one of the key issues identified was that the Waste Acceptance Criteria (WAC) for the current PA has not been developed to protect key assumptions and limits. The corrective action/resolution for this item is that a WAC document will be developed based on the analysis documented in this revision of the PA. It is not clear if this document has been developed yet and/or if any other documents have been prepared that describe the methodology that will be used to ensure that the performance of the selected grout mixtures is consistent with the performance assumed in the PA for key parameters (e.g., parameters related to the chemical and hydraulic performance of the wasteform).

Path Forward
Please provide additional information, if any, that has been developed on the planned specifications for the cementitious grout mixes to be used to stabilize waste generated as part of WTP operations. Provide the WAC for the current PA, if available. Provide a description of the process that will be used to design and the cementitious grout mixes to ensure that the performance of the grout mixtures will be consistent with the performance assumed in the PA for all of the expected compositions of the waste streams.

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