Technical Evaluation Report

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

Final Report

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ABBREVIATIONS/ACRONYMS

ADAMS	Agencywide Document Access and Management System
AEA	Atomic Energy Act of 1954, as amended
ALARA	As Low As Is Reasonably Achievable
AgM	Silver mordenite
Am	americium
Ba	barium
BBI	Best Basis Inventory
Bq	becquerel
C	carbon
°C	degrees Celsius
CCU	Cold Creek unit
CD	Consent Decree
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	*Code of Federal Regulations*
cfs	cubic feet per second
Ci	curie
Cm	curium
cm	centimeter
cm^3	cubic centimeters
CNWRA	Center for Nuclear Waste Regulatory Analyses
Co	cobalt
COPC	constituents of potential concern
Cs	cesium
CST	crystalline silicotitanate
CPGWM	Central Plateau Groundwater Model
DCF	dose conversion factor
DFLAW	direct-feed low-activity waste
DOE	U.S. Department of Energy
DOE-ORP	U.S. Department of Energy, Office of River Protection
DST	double-shell tank
EHM	equivalent homogeneous medium
EM	Environmental Management
EMCF	Environmental Model Calculation Files
EMF	Effluent Management Facility
EMMA	Environmental Model Management Archive
EPA	U.S. Environmental Protection Agency
ERDF	Environmental Restoration Disposal Facility
ET	evapotranspiration
ETF	Effluent Treatment Facility
Eu	europium
°F	degrees Fahrenheit
FEP	features, events, and processes
FFTF	Fast Flux Test Facility
FR	Federal Register
FRR	fractional release rate
ft	foot

ABBREVIATIONS/ACRONYMS (continued)

ft^3	cubic feet
FY	fiscal year
g	grams
GAC	Granular Activated Carbon
gal	gallon
GTCC	Greater-Than-Class-C
GWB	Geochemist's Workbench
^3H	tritium
H3	Hanford H3 gravel formation
HFEP	Hanford Features Events and Processes
HDPE	High-Density Polyethylene
HDW	Hanford Defined Waste or Hanford Defined Waste Model
HEPA	High-Efficiency Particulate Air
HFFACO	Hanford Federal Facility Agreement and Consent Order (Tri-Party Agreement)
HLW	high-level radioactive waste
HPB	Hanford Prototype Barrier
hr	hour
HWMA	Hazardous Waste Management Act
I	iodine
IA	Interagency Agreement
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiation Protection
ID	Idaho
IDF	Integrated Disposal Facility
ILAW	Immobilized Low-Activity Waste
in	inches
ISAM	Improvement of Safety Assessment Methodologies
ISO	International Standards Organization
IX	ion exchange
K_d	distribution coefficient
K_h	hydraulic conductivity
km	kilometers
L	liters
LAW	low-activity waste
LAWPS	Low-activity Waste Pretreatment System
LFRG	DOE Low-Level Waste Federal Review Group
LHS	Latin Hypercube Sampling
LLW	low-level radioactive waste
LSW	liquid secondary waste
m	meters
m^3	cubic meters
MBq	mega becquerel
MCC	Moisture Characteristic Curves
MCi	mega curies
MCL	Maximum Contaminant Level
mi	miles

ABBREVIATIONS/ACRONYMS (continued)

mL/g	milliliters/gram
MLLW	mixed low-level waste
mm	millimeters
MOU	Memorandum of Understanding
mph	miles per hour
mrem	millirem
mSv	millisievert
Nb	niobium
nCi	nanocurie
NCRP	National Council on Radiation Protection
NDAA	Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005
NEPA	National Environmental Policy Act
Ni	nickel
Np	neptunium
NRC	U.S. Nuclear Regulatory Commission
ORIGEN2	Oak Ridge Isotope Generation and Depletion Code 2
ORP	Office of River Protection
oz	ounce
PA	performance assessment
pCi/L	picocuries/liter
PCT	product consistency test
pH	measure of acidity (minus the log of the hydrogen ion concentration)
PMF	probable maximum flood
PMP	probable maximum precipitation
PNNL	Pacific Northwest National Laboratory
Pu	plutonium
PUREX	Plutonium Uranium Extraction Plant
QA	quality assurance
RAI	request for additional information
RCRA	Resource Conservation and Recovery Act of 1976
RCW	Revised Code of Washington
rem	unit of dose equivalent
RF	Ringold formation
ROD	Record of Decision
RSA	reactive surface area
s	second
Sb	antimony
SBS	Submerged Bed Scrubber
SC	South Carolina
SMRN	secondary mineral reaction network
SOF	sum-of-fractions
Sr	strontium
SRM	Staff Requirements Memorandum
SST	single-shell tank
SSW	secondary solid waste
STOMP©	Subsurface Transport Over Multiple Phases

ABBREVIATIONS/ACRONYMS (continued)

Sv	sievert
TBI	Test Bed Initiative
TBP	tributylphosphate
Tc	technetium
TC&WM EIS	Tank Closure & Waste Management Environmental Impact Statement
TEDE	total effective dose equivalent
TER	Technical Evaluation Report
Tri-Party Agencies	U.S. Department of Energy, Environmental Protection Agency, and the Washington State Department of Ecology
Tri-Party Agreement	Hanford Federal Facility Agreement and Consent Order (HFFACO)
TRU	transuranic
TRUEX	Transuranium Extraction
TSCR	Tank Side Cesium Removal
TST	transition-state theory
U	uranium
UDQE	Unresolved Disposal Question Evaluation
UZ	unsaturated zone (also the vadose zone)
VSL	Vitreous State Laboratory
VLAW	Vitrified Low-Activity Waste
VZ	vadose zone (also the unsaturated zone)
WA	Washington
WAC	Waste Acceptance Criteria
WESP	Wet Electrostatic Precipitator
WIPP	Waste-Isolation-Pilot-Plant
WIR	waste incidental to reprocessing
WRPS	Washington River Protection Solutions, LLC
WTP	Waste Treatment and Immobilization Plant
XRD	X-ray diffraction
Y	yttrium
yr	year

EXECUTIVE SUMMARY

The U.S. Department of Energy (DOE) Order 435.1, *Radioactive Waste Management* and DOE Manual 435.1-1, *Radioactive Waste Management Manual*, require all radioactive waste subject to the Order to be managed as either low-level waste (LLW), transuranic (TRU) waste, or high-level waste (HLW). DOE Manual 435.1-1 also states that waste resulting from reprocessing spent nuclear fuel determined to be Waste Incidental to Reprocessing (WIR) is not HLW and shall be managed under DOE's regulatory authority. The criteria for determining if the waste is not HLW, and can be managed as LLW, include:

(A) It [the waste] has been processed or will be processed to remove key radionuclides to the maximum extent that is technically and economically practical.

(B) It will be managed to meet safety requirements comparable to the performance objectives set out in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 61, Subpart C; and

(C) It is 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 the DOE Radioactive Waste Management 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 LLW as set out in 10 CFR 61.55 or will meet alternative requirements for waste classification and characterization as DOE may authorize.

The DOE has an Interagency Agreement (IA) with the U.S. Nuclear Regulatory Commission (NRC) in which it requested that the NRC provide independent technical advice and consultation regarding DOE WIR determinations for disposal of waste onsite at the Hanford Site, as well as other tasks such as closure of the HLW storage tanks in Waste Management Area C (WMA C).[1] In accordance with this IA, DOE provided the *Draft WIR Evaluation for Vitrified Low-Activity Waste (VLAW) Disposed Onsite at the Hanford Site, Washington* (DOE, 2020a) and the *Performance Assessment for the Integrated Disposal Facility (IDF), Hanford Site, Washington* (RPP-RPT-59958, 2018) to the NRC. DOE requested the NRC's consultative technical review of this information, including NRC's review of whether VLAW meets the DOE Manual 435.1-1 criteria for WIR to be managed as LLW.

VLAW is generated by the treatment of waste stored in underground storage tanks at the Hanford Site. The waste in each tank is separated into three major categories: (1) HLW that is transferred to the Waste Treatment Plant (WTP) for vitrification into glass that will be disposed in a geologic repository or other acceptable facility; (2) low-activity waste (LAW) that is vitrified into glass (i.e., VLAW) and will be disposed in the near surface in the IDF; and (3) supernatant that will be treated using a process demonstrated by the Test Bed Initiative (TBI). In the Draft WIR Evaluation for VLAW, DOE evaluated a portion of the entire VLAW waste stream referred

[1] As specified in the Interagency Agreement between the DOE and the NRC.

to as the Direct-Feed Low-Activity Waste (DFLAW). The DFLAW portion comprises approximately 10 percent of the entire LAW that will be vitrified. [2]

The NRC staff conducted an independent, risk-informed, technical review of the Draft WIR Evaluation for VLAW, using the risk insights and models developed by DOE as well as independent analysis. The NRC staff documented the results of its review in this Technical Evaluation Report (TER). The NRC staff review in this TER includes all the waste streams associated with the production of VLAW that will be disposed at the IDF, to assess the cumulative impacts from disposal. These waste streams include DFLAW (i.e., glass wasteform), the remainder of vitrified LAW, as well as other secondary wastes intended for disposal at the IDF. DOE evaluated the cumulative impacts of all these VLAW wastes in the IDF Performance Assessment (PA). The NRC staff provides conclusions for the DFLAW portion of the waste evaluated against DOE Manual 435.1-1 criteria, as well as observations about the remainder of the VLAW and secondary wastes that will be produced.

The results of the NRC staff's review of the Draft WIR Evaluation for VLAW and supporting documents are provided for Criteria A, B, and C in Sections 3, 4, and 5, respectively, of this TER. The information the NRC staff reviewed is divided into subsections covering different technical topics. These subsections are structured to summarize DOE's approach to the technical area in the Draft WIR Evaluation for VLAW (and supporting documents) followed by the NRC staff's evaluation of DOE's approach. Each subsection concludes with a summary of the NRC staff's review of that technical topic that identifies whether the NRC staff found DOE's approach to be reasonable, identifies sources of uncertainty and/or risk drivers in that area, and provides the NRC staff's recommendations for each specific technical area.

The recommendations from Sections 3, 4, and 5 are collected in Table 6-1. Table 6-1 identifies three categories of recommendations: (1) "DFLAW" means applicable to the Direct-Feed LAW that is evaluated in the Draft WIR Evaluation for VLAW; (2) "VLAW" means applicable to the remainder of the vitrified LAW or secondary wastes; and (3) "General" means if completed can improve the technical basis for DFLAW, VLAW, and/or future waste evaluations and is considered a best practice for performing waste evaluations. The recommendations from the WMA C TER that are applicable to VLAW are included in Appendix A. The recommendations do not change the conclusions of this TER with respect to meeting the DOE Manual 435.1-1 criteria, as summarized below.

The compliance period specified in DOE Order 435.1 for a performance assessment evaluation is 1,000 years after closure, however, DOE also provided results for a 10,000-year post-closure period to understand the potential impacts at longer timeframes. DOE estimated impacts to air and water pathways for all VLAW that were well below the performance standards during the compliance period. Air and water pathway impacts can be sensitive to the total quantity of waste whereas intruder impacts are more sensitive to radionuclide concentrations. The most limiting result from DOE's analysis for the compliance period was for the inadvertent intruder. DOE estimated a chronic intruder dose of 0.43 mSv/yr (43 mrem/yr), which is below the

[2] This document refers to the "Draft WIR Evaluation for VLAW", as DOE has entitled the waste evaluation, however, the waste evaluated in DOE's document is the DFLAW portion, which is a subset (10 percent) of the entire VLAW waste stream.

1 mSv/yr (100 mrem/yr) DOE performance objective. NRC applies a 5 mSv/yr (500 mrem/yr) dose limit for inadvertent intruders.

Overall Results and Conclusions

The NRC staff has the following general conclusions, followed by specific conclusions and recommendations for DFLAW and non-DFLAW wastes:

- DOE demonstrated numerous good practices and acceptable quality assurance in its clearly documented analyses.

- Limiting the scope of the evaluation to a particular wasteform (glass) resulting from waste processing adds uncertainty as to whether the cumulative impacts of the wastes that will be disposed at the IDF are acceptable. All waste streams resulting from processing that could produce a significant contribution to projected doses that were included within the IDF PA should be within the scope of the WIR evaluation. Alternatively, DOE should develop a separate evaluation that documents DOE's decision that all wastes disposed in the IDF meet the WIR criteria from DOE Order 435.1, with an indication of the point in the waste process flow that this determination is made.

- DOE did not adequately support the high retention values of ^{99}Tc and ^{129}I in glass provided in its response to the NRC staff's request for additional information (NRC, 2020a).

- The lack of verification plans for production-scale wasteforms is a large uncertainty given the novel wastes and recycling of off-gas used during vitrification. The NRC staff recommends that DOE collect operational data to verify that the actual wasteform performance is consistent with the wasteform performance in the Draft WIR Evaluation for VLAW.

DOE requested that the NRC staff provide conclusions for the DFLAW portion of the vitrified waste with respect to meeting DOE Manual 435.1-1 Criteria. <u>For DFLAW</u>, the NRC staff concludes:

- DOE demonstrated that the waste has been processed or will be processed to remove key radionuclides to the maximum extent that is technically and economically practical (Criterion A).

- DOE demonstrated that the waste will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C (Criterion B).

- DOE demonstrated that the waste will 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 the DOE Radioactive Waste Management Manual. The waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR 61.55 (Criterion C).

The following assumptions apply to the NRC staff's conclusions above and should be verified by DOE:

- DOE will produce wasteforms during operations that are of similar performance and characteristics to those currently estimated.

- Recycle of off-gases will not result in the buildup of deleterious species that significantly impacts glass performance.

- Cracking and the availability of cracked glass surface area for release will be comparable between production and surrogate data.

- DOE will achieve greater than 95 percent retention of ^{99}Tc and ^{129}I in glass using recycle.

- DOE will install a thick engineered cover with adequate erosion protection.

DOE included all low-activity wastes in the IDF PA analyses (RPP-RPT-59958, 2018) and included the combined doses from all low-activity wastes in the Draft WIR Evaluation for VLAW. DOE did not specifically identify the dose impacts resulting from the DFLAW portion of the waste, however, DOE provided sufficient information in the IDF PA to allow the NRC staff to understand the contributions to doses resulting from disposal of each of the low-activity wastes in the IDF, including DFLAW. DOE did not provide information associated with Criterion C for the non-DFLAW portions of the waste.

The NRC staff has the following observations associated with the <u>non-DFLAW</u> portion of the wastes:

- Select secondary wastes could be produced that exceed the Class C concentration limits (e.g., granular activated carbon (GAC)). These wastes would not meet Criterion C unless alternate criteria were applied.

- The remainder of the vitrified waste is likely to meet Criteria A, B, and C but the demonstration is not complete without DOE's demonstration that acceptable wasteforms can be made that account for the differences in the tank waste (e.g., soluble Sr, organics, minor species that buildup during recycle).

- The secondary wasteforms are under development, therefore, their performance is more uncertain. The risk-significance of the secondary wasteforms will be determined by the actual retention rates of volatile species in glass experienced after production begins.

- Waste variability may require the use of a longer institutional control period or the use of other mitigation actions to ensure the protection of inadvertent intruders.

- Impacts to water and air pathways during DOE’s 1,000-year compliance period are likely to be well below established limits for all wastes (vitrified and non-vitrified).

The results of NRC staff’s review of Criteria A, B, and C in this TER are being provided to DOE for consideration. NRC has no regulatory authority related to DOE’s waste determination activities. DOE has stated it will consider the information in the NRC staff’s TER and the comments from stakeholders before finalizing the WIR evaluation for VLAW (or DFLAW), which will contain the final waste determination of whether DFLAW can be managed as LLW.

1 INTRODUCTION

The Hanford Site in Washington is large and has a complex history. The U.S. Department of Energy (DOE) has generated a variety of different wastes over more than half a century. Much of the waste is currently stored in large underground storage tanks. The cleanup of the site involves more than 200 million liters (L) (53 million gallons) of radioactive and chemically hazardous waste in 177 underground storage tanks, about 750,000 m^3 (25 million ft^3) of buried or stored solid waste, spent nuclear fuel, and plutonium in various forms. The massive underground storage tanks were built throughout Hanford's 200 Areas in a series of groups (known as tank farms) to hold the wastes, ranging in capacity from 208,000 liters (55,000 gallons) to more than 3,785,000 liters (1,000,000 gallons). Though there are different tank designs, most tanks include a carbon steel shell surrounded by reinforced concrete. The materials inside waste tanks consist of liquids, gases, semi-solids, and solids.

Vitrified low-activity waste (VLAW) is generated in DOE's treatment process for waste stored in underground tanks at the Hanford Site. The waste in each tank is separated into three major categories: (1) High-level waste (HLW) that transferred to the Waste Treatment Plant (WTP) for vitrification and disposal in a geologic repository or other acceptable facility, (2) low-activity waste (LAW) that is vitrified into glass (i.e., VLAW) and will be disposed in the near surface in the IDF, and (3) supernatant that will be treated using a process demonstrated by the Test Bed Initiative (TBI).

DOE requested, by letter dated April 24, 2020 (DOE, 2020b), that the Nuclear Regulatory Commission (NRC) conduct a consultative review of its *Draft Waste Incidental to Reprocessing (WIR) Evaluation for Vitrified Low-Activity Waste Disposed Onsite at the Hanford Site, Washington* (Draft WIR Evaluation for VLAW) (DOE, 2020a). The DOE also provided the *Performance Assessment for the Integrated Disposal Facility, Hanford Site, Washington* (IDF PA) to the NRC (RPP-RPT-59958, 2018) for review along with other supporting documents.

DOE requested that the NRC perform a consultive review of the impacts from the disposal of waste resulting from the separation, pretreatment, and vitrification of approximately 89 million liters (23.5 million gallons) of VLAW from underground tanks at the Hanford Site. For the LAW considered in the Draft WIR Evaluation for VLAW, DOE plans to use a direct-feed low-activity waste (DFLAW) approach. The DFLAW approach is a two-phased approach that will separate and pretreat liquid wastes from some of the Hanford tanks to generate a LAW stream. In Phase 1 the liquid waste will be supernate (essentially the upper-most layer of tank waste that contains lower concentrations of long-lived radionuclides) whereas in Phase 2 the waste may also include dissolved saltcake and interstitial liquids. The DFLAW approach will result in approximately 13,500 canisters of vitrified glass that will be disposed onsite in the IDF. The DFLAW that will be disposed is about 29,000 m^3 (1,000,000 ft^3), which is approximately 10 percent by volume of the total VLAW glass that will be produced. The remaining wastes from the vitrification process consists of other forms of VLAW glass and secondary wastes that will mainly be stabilized with cement and disposed in the IDF.

In the Draft WIR Evaluation for VLAW, DOE evaluated the portion of the VLAW called DFLAW.[3] The NRC staff review in this Technical Evaluation Report (TER) included all the wastes associated with VLAW that will be disposed in the IDF at Hanford, to assess the cumulative impacts from disposal. These wastes include DFLAW (i.e., glass wasteform), non-glass wasteforms, as well as other secondary wastes intended for disposal at the IDF. DOE evaluated the cumulative impacts of the wastes generated from the VLAW processing in the IDF PA. The IDF will contain among the largest amounts of waste in the DOE complex and will have one of the largest inventories of long-lived radionuclides in a low-level waste (LLW) disposal facility.

1.1 Regulatory Framework

The overall concept of WIR is that some waste can be managed based on the risk to human health and the environment rather than based on the origin of the waste (e.g., reprocessing spent nuclear fuel). Much of the waste in the tank farms at the Hanford Site is highly radioactive and needs to be treated and disposed as HLW. However, other waste may be demonstrated not to require disposal in a geologic repository by means of a DOE analysis called a "waste determination". If it can be demonstrated that the waste poses a sufficiently small risk to human health and the environment and does not need to be disposed of as HLW, DOE may determine through the technical analyses documented in a waste determination that the waste is incidental (i.e., WIR) or non-HLW.

The NRC staff provide a complete history of the WIR regulatory framework in NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations" (NRC, 2007). The concept of incidental waste has been recognized since 1969 when the Atomic Energy Commission, NRC's predecessor agency, issued for comment a draft policy statement regarding the siting of reprocessing facilities in the form of a proposed Appendix D to 10 CFR Part 50, which addressed a definition of HLW (AEC, 1969). The term "incidental waste" was first used in NRC's 1987 advance notice of proposed rulemaking to redefine the definition of HLW (NRC, 1987). However, in the 1989 final rulemaking action on disposal of radioactive waste, the Commission did not redefine HLW (NRC, 1989).

In 1990, the States of Oregon and Washington petitioned the Commission to amend 10 CFR Part 60 to redefine HLW. The petition concerned whether Hanford tank waste was subject to NRC licensing jurisdiction. In response to the petition, the Commission approved specific criteria for determining whether waste was incidental and issued a Staff Requirements Memorandum (SRM) dated February 16, 1993, in response to SECY-92-391, "Denial of PRM 60-4: Petition for Rulemaking from the States of Washington and Oregon Regarding Classification of Radioactive Waste at Hanford." NRC published the criteria in the Federal Register as part of the petition denial, as follows (NRC, 1993):

> (1) The waste has been processed (or will be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical,

[3] This document refers to the "Draft WIR Evaluation for VLAW", as DOE has entitled the waste evaluation, however, the waste evaluated in DOE's document is the DFLAW portion, which is a subset (10 percent) of the entire VLAW waste stream.

(2) The waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR Part 61, and

(3) The waste is to be managed, pursuant to the Atomic Energy Act, so that safety requirements comparable to the performance objectives set out in 10 CFR Part 61, are satisfied.

In July 1999, DOE issued DOE Order 435.1, "Radioactive Waste Management" and the associated Manual, DOE Manual 435.1-1, "Radioactive Waste Management Manual," both of which were subsequently revised (DOE, 2021a; DOE, 2021b). DOE Manual 435.1-1 requires all radioactive waste subject to the Order to be managed as either LLW, transuranic (TRU) waste, or HLW. DOE Manual 435.1-1 also states that waste resulting from reprocessing spent nuclear fuel determined to be WIR is not HLW and shall be managed under DOE's regulatory authority as LLW. DOE Manual 435.1-1 discusses DOE's incidental waste evaluation process and the criteria for determining whether waste is incidental to reprocessing (see Section 1.2), which are based upon the criteria NRC issued above.

In 2004, Senator Lindsey Graham of South Carolina introduced legislation that would allow DOE to use a process similar to the incidental waste process in DOE Order 435.1 at the Savannah River Site (SRS). Congress passed the Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA) on October 9, 2004, and the President signed it on October 28, 2004. Section 3116 of the NDAA allows DOE to continue to use an incidental waste process to determine that waste is not HLW, and Section 3116(a) includes the requirement that DOE consult with the NRC on the DOE non-HLW determinations. However, the NDAA is applicable to only South Carolina and Idaho and does not apply to waste transported out of these states.

Although the NDAA only addresses consultation and monitoring activities within the NDAA-Covered States of Idaho and South Carolina,[4] at the DOE's request the NRC also conducts technical reviews for WIR management and disposal at sites in non-NDAA covered states (e.g., States of Washington and New York) following a similar process to that used at the covered states. For the Hanford Site, the criteria for DOE waste determinations are specified by DOE Order 435.1[5] and DOE Manual 435.1-1. Under DOE Manual 435.1, the DOE may consult with NRC on WIR determinations through an Interagency Agreement (IA).

DOE and NRC have had a series of IAs where DOE requested that the NRC provide technical advice and "consultation" regarding DOE WIR determinations for disposal of waste onsite at the Hanford Site, as well as other tasks such as closure of the HLW storage tanks in Waste Management Area C (WMA C).[6] The NRC staff performs this consultive role by conducting an independent technical review so that the NRC can reach its own conclusions as to whether DOE's proposed waste management approach satisfies the DOE Order 435.1 criteria. The NRC staff's guidance for the consultation activities are documented in NUREG-1854 (NRC, 2007). At the conclusion of the NRC staff's review of each WIR determination, the NRC provides DOE with a TER documenting its findings. NRC has no regulatory oversight over

[4] https://www.nrc.gov/waste/incidental-waste/wir-ndaa.html
[5] https://www.directives.doe.gov/directives-documents/400-series/0435.1-BOrder-chg2-AdminChg
https://www.directives.doe.gov/directives-documents/400-series/0435.1-DManual-1-chg3-ltdchg-1
[6] As specified in the Interagency Agreement between the DOE and the NRC.

DOE's WIR evaluations at the Hanford Site.

1.2 NRC Review Approach

DOE provided the *Draft WIR Evaluation for Vitrified Low-Activity Waste Disposed Onsite at the Hanford Site, Washington* (Draft WIR Evaluation for VLAW) (DOE, 2020a) and the *Performance Assessment for the Integrated Disposal Facility, Hanford Site, Washington* (IDF PA) (RPP-RPT-59958, 2018) to the NRC. DOE requested that the NRC staff review the Draft WIR Evaluation (and supporting information) for disposal of DFLAW waste onsite at the IDF.

The WIR determination process used by DOE is based on the criteria that is provided in DOE Order 435.1 and the related DOE Manual 435.1-1. DOE uses the process to determine if WIR is not HLW and can be managed as either LLW or TRU waste. The criteria for determining if the waste can be managed as LLW include:

(A) It [the waste] has been processed or will be processed to remove key radionuclides to the maximum extent that is technically and economically practical.

(B) It will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C; and

(C) It is 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 the DOE Radioactive Waste Management 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 LLW as set out in 10 CFR 61.55 or will meet alternative requirements for waste classification and characterization as DOE may authorize.

The NRC staff interprets the intent of Criterion A to limit the amount of inventory of key radionuclides that will remain after removal (e.g., for tank closure) or waste disposal. The goal is to minimize the impacts to the extent practical. Inventory can be reduced by decreasing the volume of waste at a constant concentration of radionuclides or by selectively removing radionuclides, thereby reducing the concentration of key radionuclides. In the case of disposal of waste onsite at the IDF, key radionuclide removal could occur by (1) disposal of some of the waste as HLW, (2) disposal of select wastes that are generated during waste processing as HLW, and/or (3) removal of key radionuclides from the waste prior to disposal of the remainder of the waste as LLW. For this case, Criterion A can involve the consideration of the risks versus the benefits of removing key radionuclides.

For Criterion B, the NRC staff evaluated DOE's documents to determine if the resultant actions or proposed actions will demonstrate compliance with the performance objectives in 10 CFR Part 61, Subpart C. The requirements in 10 CFR 61.40 include that land disposal facilities be sited, designed, operated, closed, and controlled after closure such that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in 10 CFR 61.41 through 10 CFR 61.44.

To evaluate compliance with the performance objective for the protection of the general population from releases of radioactivity (10 CFR 61.41), the NRC staff review must confirm that concentrations of radioactive material from DFLAW that may be released to the general environment in groundwater, surface water, air, soil, plants, or animals will not result in an annual dose to a member of the public that is greater than 0.25 millisieverts (mSv) [25 millirem (mrem)] and will be maintained as low as is reasonably achievable (ALARA). The regulation provides an older dosimetry method as the limit in the regulation, the Commission has a policy to use more modern dosimetry with a single annual dose criterion of 0.25 millisieverts (mSv) [25 millirem (mrem)] for WIR evaluations (NRC, 2007).

The performance objective for protection of individuals from inadvertent intrusion (10 CFR 61.42) requires that the design, operation, and closure of the land disposal facility will ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed. NRC typically applies a limit of 5 mSv/yr [500 mrem/yr] to assess compliance with §61.42, although the performance objective does not provide numerical dose criteria for protection for the inadvertent intruder, as discussed in NUREG-1854 (NRC, 2007).

The performance objective for the protection of individuals during operations (10 CFR 61.43) requires that land disposal facility operations will comply with the standards for radiation protection set out in 10 CFR Part 20, except for releases of radioactivity in effluents from the land disposal facility, which will be governed by 10 CFR 61.41. In addition, the performance objective requires that radiation exposures during operations are maintained ALARA.

The performance objective for stability of the disposal site after closure (10 CFR 61.44) requires that a disposal facility be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate, to the extent practicable, the need for ongoing active maintenance of the disposal site following closure, so that only surveillance, monitoring, or minor custodial care is required. Evaluation of compliance with 10 CFR 61.44 is limited to a review of site stability and those features, events, and processes (FEPs) that may impact site stability. The NRC staff reviewed the information associated with the determination of the frequency and magnitude of potential disruptive events and whether the effects of site instabilities were adequately modeled or bounded in the performance assessment (PA) and inadvertent intruder analysis.

For Criterion C, DOE Manual 435.1-1 prohibits waste that exceeds Class C concentration limits from being determined to be incidental waste, unless DOE authorizes alternate criteria. If DOE authorizes alternate criteria, the NRC staff would evaluate whether there is reasonable assurance that the alternate criteria can be met and whether the proposed alternate criteria are protective of public health and safety.

The NRC staff has carried out a risk-informed review of DOE's waste evaluation documents and information and documented the results in this TER. A risk-informed evaluation means that the review effort given to a technical topic during the evaluation should be commensurate with the risk-significance of that topic; therefore, more attention and review time was given to FEPs of waste processing and waste disposal at the IDF that have the potential to significantly impact public health and safety than to less significant FEPs. Although less significant FEPs are also evaluated during the review, not all FEPs or parameter range values are discussed in this TER.

The results of the NRC staff's review of the Draft WIR Evaluation for VLAW and supporting documents[7] are provided for Criteria A, B, and C in Sections 3, 4, and 5, respectively, of this TER. For each of the criterion, the information the NRC staff reviewed is divided into subsections covering different technical topics. These subsections are structured to summarize DOE's approach to the technical topic in the Draft WIR Evaluation for VLAW followed by the NRC staff's evaluation of DOE's information. Each subsection concludes with a summary of the NRC staff's review of that technical topic that identifies whether the NRC staff found DOE's approach to be reasonable, identifies sources of uncertainty and risk drivers, if applicable. NRC staff's recommendations for each technical topic are provided as well.

The recommendations provided in each subsection are collated in Table 6-1. Table 6-1 identifies three categories of recommendations: (1) "DFLAW" means applicable to the Direct-Feed LAW that is evaluated in the Draft WIR Evaluation for VLAW; (2) "VLAW" means applicable to the remainder of the vitrified LAW or secondary wastes; and (3) "General" means if completed can improve the technical basis for DFLAW, VLAW, and/or future waste evaluations and is considered a best practice for performing waste evaluations. The recommendations from the WMA C TER that are applicable to VLAW are included in Appendix A.

1.3 Purpose of the TER

DOE requested, by letter dated April 24, 2020 (DOE, 2020b), that the NRC staff conduct a consultative review of its Draft WIR Evaluation for VLAW, dated April 23, 2020 (DOE, 2020a). DOE also submitted the IDF PA, dated August 2018 (RPP-RPT-59958, 2018), to the NRC along with other supporting documents. In a letter dated November 6, 2020, NRC requested additional information from DOE on several technical topics (NRC, 2020a). In a submittal dated July 29, 2021, DOE responded to NRC's request for additional information (RAI) and included additional references, as discussed during several teleconferences between NRC and DOE in September 2021 (DOE, 2021c; NRC, 2021a-d).

The purpose of the Draft WIR Evaluation for VLAW and its supporting documents is to show that DOE's actions or proposed actions for managing DFLAW resulting from the processing of waste removed from underground storage tanks at Hanford will satisfy the criteria in DOE Manual 435.1-1. These criteria must be met to determine that DFLAW WIR is not HLW and may be managed as LLW.

The NRC staff's independent review of the Draft WIR Evaluation for VLAW and the supporting IDF PA was conducted in accordance with the IA between the DOE and the NRC. In the IA, the DOE requested NRC emphasis on DOE Manual 435.1-1 Criterion B (i.e., meeting safety standards comparable to the performance objectives set out in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 61 Subpart C)[8] over DOE Manual 435.1-1 Criterion A (i.e., the removal of key radionuclides). The NRC staff placed emphasis on Criterion B, however, evaluated all three criteria for completeness and transparency for other stakeholders. DOE

[7] The documentation reviewed by NRC staff is much more extensive than the Draft WIR Evaluation for VLAW and IDF PA. Those documents are frequently cited in this TER, however, the NRC staff reviewed additional supporting information to the extent necessary to make an independent determination.

[8] The performance objectives in 10 CFR Part 61, Subpart C were established to provide reasonable assurance that a LLW disposal site would be designed, operated, and closed in a way that is protective of human health and safety. The compliance period is not defined in 10 CFR Part 61, though in practice for commercial LLW disposal, a period is selected on a site-specific basis such that the impacts to public health and safety are assessed.

requested consultation for the reasonable expectation[9] of compliance with the performance objectives for a compliance period of 1,000 years.

The NRC staff's review of the Draft WIR Evaluation for VLAW[10] and the IDF PA is documented in this TER. DOE has stated it will consider the information in this TER and the comments from stakeholders before releasing the Final WIR Evaluation for VLAW.

1.4 Scope of the TER

DOE stated that the scope of the Draft WIR Evaluation for VLAW submitted for the NRC staff's consultative review was limited to the DFLAW fraction of the waste (DOE, 2020a). During waste processing, such as vitrification, some radionuclides are separated or partitioned selectively into different output waste streams intended for disposal at the IDF. DOE indicated that it considered these wastes to be "secondary" to processing the original wastes (i.e., DFLAW) and that the secondary waste did not require a waste determination to conclude that the waste was not HLW (DOE, 2021c). DOE indicated that the inventory evaluated in the IDF PA modeling included all wastes (i.e., DFLAW, secondary wastes, and ancillary waste). Figure 1-1 is a simplified representation of the processing, treatment, and disposal of waste from underground storage tanks at Hanford and shows the scope of the Draft WIR Evaluation for VLAW.

In the NRC staff's RAI (NRC, 2020a), staff indicated that they examined the scope of the Draft WIR Evaluation for VLAW in their acceptance review and concluded that DOE's approach described above was not consistent with the NRC staff's understanding of the intent of the incidental waste process. DOE's selection of vitrification as the primary waste production process results in some key radionuclides that are volatized and separated from the original waste (e.g., ^{129}I), or are removed in other processing steps. The NRC staff stated that if most of that activity that is separated or removed will be placed in a near-surface disposal facility (i.e., as other than HLW), then the resulting wasteforms and waste streams should be 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 at the IDF. As a result, the NRC staff included secondary wastes within the scope of the review included in this TER.

In response to the NRC staff's RAI, DOE stated that the DFLAW approach was designed to separate and pretreat waste with comparatively lower concentrations of key radionuclides. The DFLAW pretreatment approach entails in-tank settling, decanting, filtration, and cesium ion exchange removal. Cesium ion exchange will use the Tank Side Cesium Removal (TSCR) System for Phase 1 and either a second TSCR unit or a filtration and cesium removal facility for Phase 2. The processes will remove over 99 percent of the cesium, as well as other key radionuclides. During vitrification of the LAW, some radionuclides, including ^{99}Tc and ^{129}I, will volatize. DOE stated that the LAW Vitrification Facility will, by design, maximize the capture of the volatized ^{99}Tc and ^{129}I into the VLAW. DOE explained that since the completion of the IDF PA, the latest flowsheet modeling showed that approximately 98 percent of the ^{99}Tc and

[9] DOE's use of "reasonable expectation" when compared to NRC's use of "reasonable assurance" is not materially different from a technical perspective.

[10] Vitrified low-activity waste (VLAW) is the portion of low-activity waste (LAW) that will be vitrified. DOE also uses the term immobilized low-activity waste (ILAW) to describe LAW that has been treated. ILAW Glass and VLAW are used interchangeably by DOE. DFLAW is the portion of the LAW that will be treated under a phased approach by DOE. See Figure 1-1.

approximately 96 percent of the ^{129}I would be captured in the VLAW glass, and approximately 99 percent of all radioactivity in the pretreated LAW will be incorporated into the VLAW.

The NRC staff evaluated the information DOE provided in response to the RAI. However, the NRC staff believe that regardless of partitioning fractions that result from downstream processing, waste should not be segmented in the WIR evaluation process if it is to be disposed in the same facility. For example, assume downstream processing partitions a waste into two components (A and B) and most of the activity remains in Fraction A. In this example, Fraction B may not be a significant contributor to risk. However, if the wasteforms developed for each component have significantly different performance, then even though Fraction B may contain a much smaller fraction of the activity, it could be a significant contributor to risk. This issue has an additional challenge of considering which radionuclides end up in which waste fraction. In general, bulk activity (total becquerels or curies) cannot be used reliably to make risk-informed decisions for waste disposal.

In addition, regarding the scope of this TER, some technical aspects (e.g., receptors and representation of the biosphere) of the Draft WIR Evaluation for VLAW (DOE, 2020a) are nearly identical to the information DOE provided with respect to the Draft WIR Evaluation for WMA C (DOE, 2018) that the NRC staff previously reviewed (NRC, 2020b). The NRC staff reviewed the information provided for VLAW to identify if there were significant differences, and discusses those differences in this TER. Where similarities exist, only a summary discussion of the DOE analyses and the NRC review of the DOE analyses are provided in this TER. For additional information the reader may consult the TER for WMA C (NRC, 2020b). The recommendations from the WMA C TER that are applicable to VLAW are included in Appendix A.

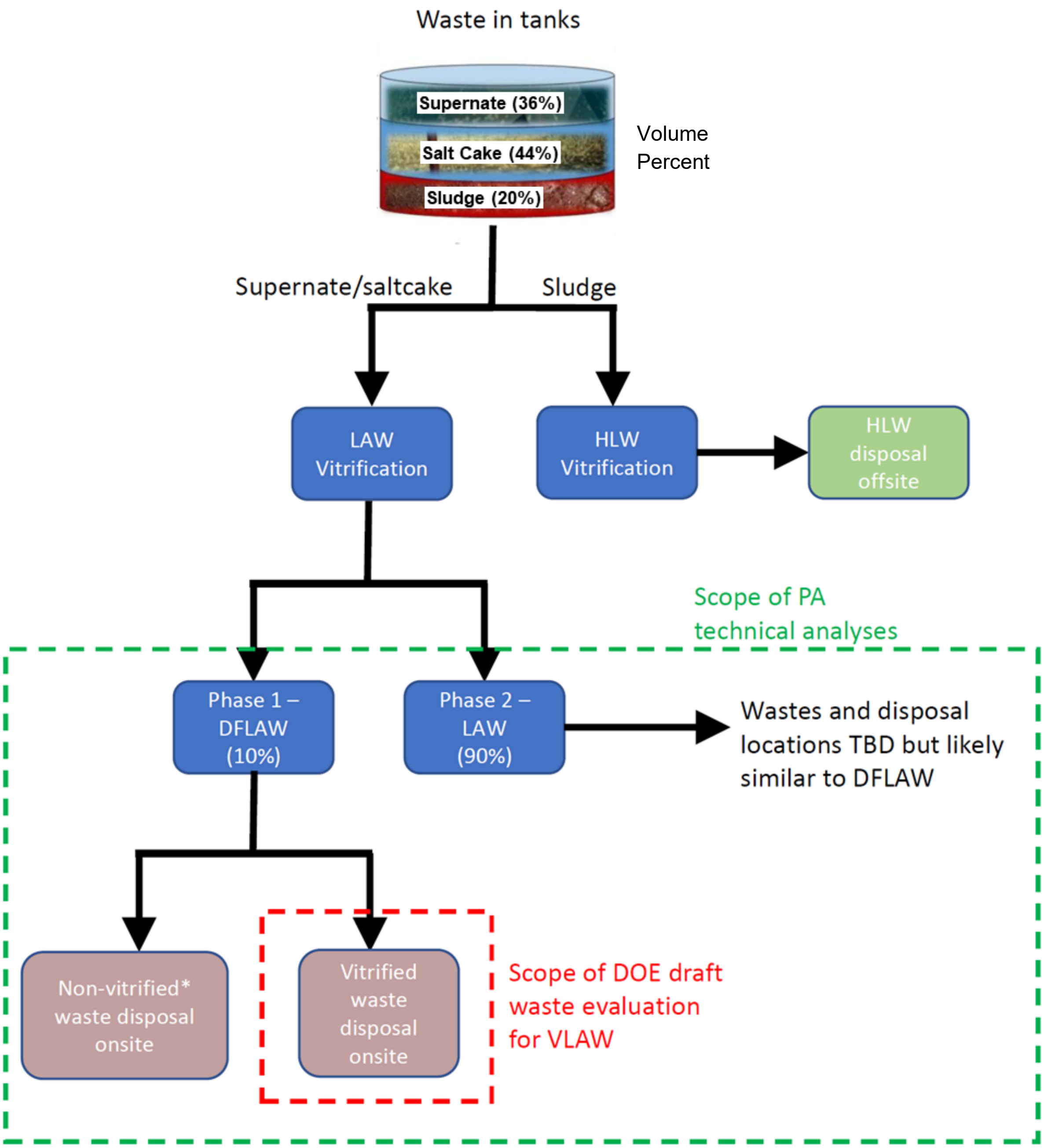


Figure 1-1 Simplified Representation of Tank Waste Processing at Hanford

* The non-vitrified waste includes secondary solid waste (SSW) produced during glass processing at the WTP and waste treatment at the effluent treatment facility (ETF) during glass processing. See Figure 2-5.

2 REVIEW CONTEXT

This section provides context for the NRC staff's review in this TER. The description of the disposal site provides an overview of the site location, natural environment, geology, and hydrogeology. The description of the disposal facility details the man-made barriers put in place to limit the release of radioactivity from the disposed waste. Section 2.3 provides an overview of the wastes that are proposed for disposal. And finally, the demography, natural resources, and present and future land use of the site are summarized.

2.1 Disposal Site Description

DOE's Hanford Site lies within the Columbia Plateau in the southeastern part of Washington. The Hanford Site is semi-arid and is situated north of the city of Richland at the confluence of the Yakima and Columbia Rivers (see Figure 2-1). The Hanford Site measures roughly 50 km (31 mi) north to south and 40 km (25 mi) east to west. Much of the site's approximately 1,500 km^2 (575 mi^2) area is restricted from public access. Restriction of public access to the site helps to protect the public from the nuclear materials storage, waste storage, and waste disposal areas located within the site. About 6 percent of the of shrub-steppe and grasslands covered land has been disturbed and is actively used.

2.1.1 Geography

The Columbia River runs through the Hanford Site. This area is characterized by generally low-relief hills with incised river drainage. The Hanford Site is an area of generally low relief, ranging from 120 m (390 ft) above mean sea level at the Columbia River to 230 m (750 ft) above mean sea level. Cataclysmic flooding shaped the topography of the Hanford Site when ice dams holding back large glacial lakes were abruptly breached. Much of the site was stripped of soils and sediments and basalt bedrock was initially scoured due to the massive floods. Deposition of soils and sediments occurred with the recession of the floodwaters. Winds have locally reworked the flood sediments, resulting in loess (windblown silt) and dunes (sands) being deposited in the lower elevations.

2.1.2 Meteorology and Climate

Climatological data for the Hanford Site is collected and processed at monitoring sites. Since the early 1980s, key information has been transmitted to a meteorology station every 15 minutes. Based on data collected from 1946 through 2001, the average monthly temperatures range from a low of -0.7°C (31°F) in January to a high of 24.7°C (76°F) in July, and daily maximum temperatures vary from an average of 2°C (35°F) in late December and early January to 36°C (96°F) in late July. Average annual precipitation at the Hanford Site is 17 cm (6.8 in.). Current annual precipitation results in a semi-arid climate characterization. Most precipitation occurs during the late autumn and winter. Approximately 50 percent of total rainfall occurs from November through February with snowfall accounting for about a third of that amount. The wettest recorded year for the site was 1995 with 31.3 cm (12.3 in.) of precipitation (DOE, 2004). Both snowmelt and rainfall can be episodic and are characterized by large events within the overall context of low annual average precipitation.

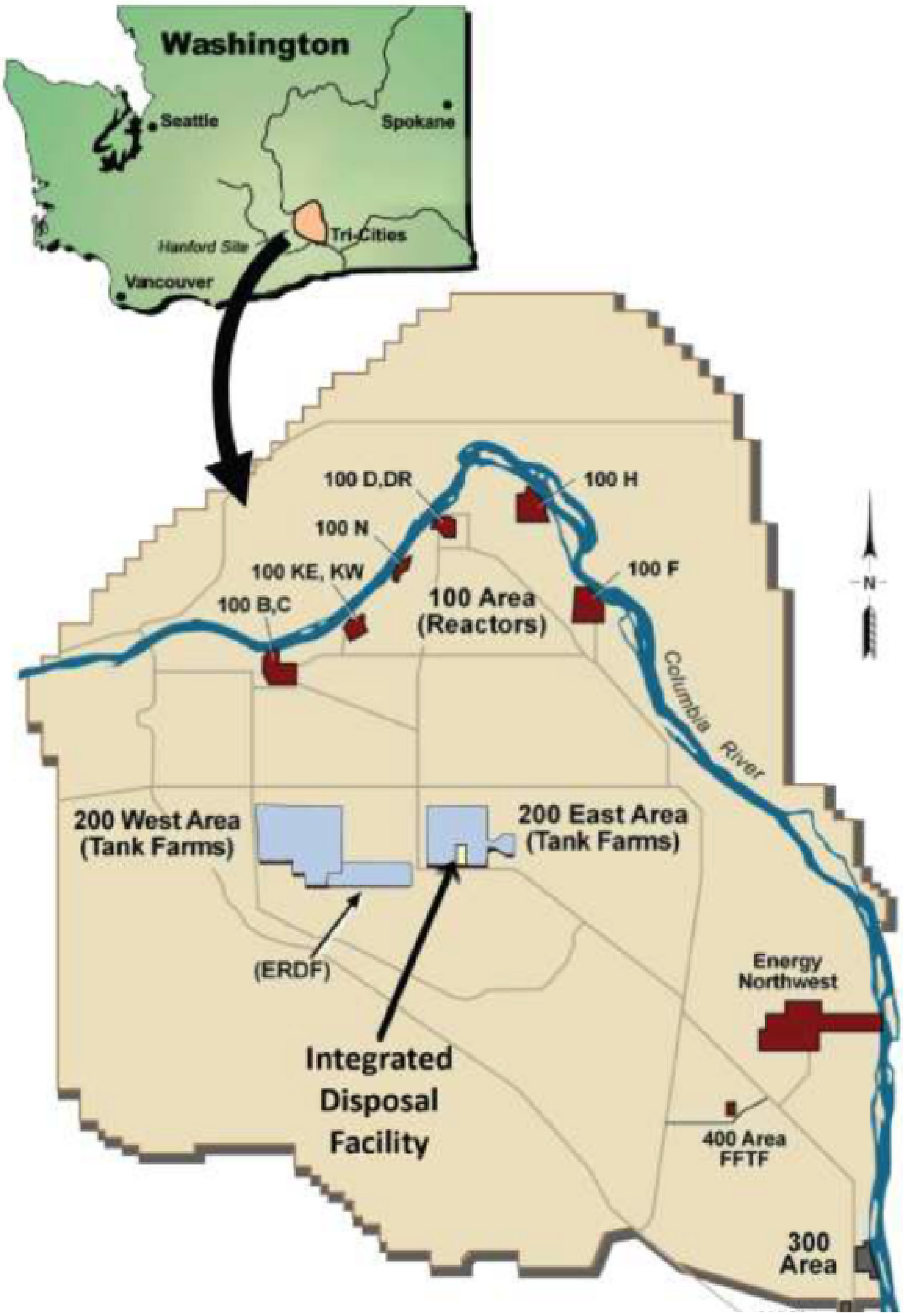


Figure 2-1 The Hanford Site
[Figure 2-1a in the IDF PA (RPP-RPT-59958,2018)]

The Cascade Mountain range in the western part of Washington influences the climate of the Hanford Site by means of its rain shadow effect. The rain shadow effect is when storms and wet weather are blocked by a mountain range. Summers are warm and dry while winters are cool with occasional precipitation. The Cascades also affect the wind regime at the Hanford Site. Prevailing wind directions near the surface are predominantly from the northwest in all months of the year with higher average wind speeds in the summer [3.6 to 4.0 m/s (8 to 9 mph)] compared to that of the winter months [2.7 to 3.1 m/s (6 to 7 mph)]. Intense low-pressure systems can generate winds of near hurricane force on rare occasions, but most high-speed winds are more commonly associated with the passage of strong cold fronts.

On average, ten thunderstorms occur in the central area at the Hanford Site each year; only a small percent of these are classified as severe based on wind speed or the presence of hail. Eighteen tornadoes were recorded from 1950 through March 2001 in ten counties adjacent to the Hanford Site. Maximum wind speeds in the range of 51 to 71 m/s (113 to 157 mph) were recorded for three of these tornadoes; the rest had lower speeds.

2.1.3 Geology

The Pasco Basin, in which the Hanford Site is located, is bounded by the Gable Mountain anticline to the north and the Cold Creek syncline to the south. The 200 East Area sits on the northern flank of the Cold Creek syncline.

The Gable Mountain anticline influences the hydrogeological flow regime beneath the Hanford Site since this anticline has been uplifted to a point where portions of the basalt are above the current water table. Due to low hydraulic conductivity, this basalt acts as a barrier to horizontal groundwater flow in the unconfined aquifer. The basalt thickness is 3,000 m (10,000 ft) or more, and the top of the basalt unit slopes gently to the southwest. The geologic units comprising the vadose (unsaturated) zone in the vicinity of the IDF are approximately 100 m (300 ft) thick. Though there is heterogeneity and variability in the properties of the geologic materials, DOE simplified the geologic representation into three layers (from the top down) of the H1, H2, and H3. From the bottom of the disposal facility the H2 unit extends 70 m (230 ft) and the H3 unit was represented as being 9 to 40 m (30 to 130 ft) thick with about 9 to 30 m being unsaturated (30 to 100 ft).

An undifferentiated Hanford H3 gravel, Cold Creek, and Ringold Unit exists above the basalt unit. A paleochannel in the 200 East Area eroded many of the previous formations above the basalt which probably included the Ringold formation (RF), the Cold Creek unit (CCU), and Hanford H3 gravel formation (H3). Today, these layers are indistinguishable from one another, having been reworked and redeposited to form a coarse-grained gravel to sandy gravel undifferentiated unit designated as the H3/CCU/RF unit or commonly referred to as the H3. Thicknesses of the H3 unit in the vicinity of the IDF are 14 to 21 m (46 to 69 ft).

The Hanford H2 sand formation lies above the H3 unit, and thicknesses in the vicinity of the IDF for the H2 unit are 80 to 85 m (260 to 280 ft). Silt lenses (<0.3 m [1 ft]) and thinly interbedded zones of silt and sand are common but are not abundant in the H2 unit and appear to be discontinuous. The upper portion of H2 unit may have been eroded during Ice Age flooding and the overlying gravelly H1 unit was subsequently deposited. The H1 has a thickness of between 9 m and 15 m (30 ft and 50 ft) in the vicinity of the IDF. However, the IDF is a large near-surface trench with the removal of the H1 and some of the H2, making the thickness of the H1 insignificant from a performance modeling standpoint. Backfill will be used to fill void space. A thick, engineered cover will be placed over the trenches at closure. The exact materials of the engineered cover have not been finalized as the design will be completed at a future date.

During excavation of the IDF, clastic dikes were found (PNNL-15237, 2005). The extent, properties, and therefore, the significance of the clastic dikes on performance are uncertain. Clastic dikes consist of multiple vertical layers of unconsolidated sand, silt, clay, and minor gravel. Clastic dikes have been documented at the Hanford Site and can range in vertical extent from 0.3 m to 55 m (1 ft to 180 ft) and range in thickness from 1 mm to 1.8 m (0.04 in to 5.91 ft).

2.1.4 Hydrology and Hydrogeology

The largest river at the Hanford Site is the Columbia River. The Yakima River forms the southern boundary of the Hanford Site before merging with the Columbia River in Richland, WA. The nearest dam to the Hanford Site is the Priest Rapids Dam, a few miles upstream on the Columbia River. Estimates of the Columbia River probable maximum flood (PMF), which is determined from the upper limit of precipitation falling on the drainage area and other hydrologic factors (e.g., snowmelt), indicate that the PMF would inundate parts of the areas located adjacent to the Columbia River, but the central region of the Hanford Site, known as the Central Plateau, would remain unaffected (DOE, 1986).

The unsaturated zone, or vadose zone, includes sediments or rocks that are not saturated with water and extends down from the ground surface to the water table, or the top of the saturated zone. The unsaturated zone can significantly delay the transport of radionuclides to a potential receptor. Unconsolidated glacio-fluvial sands and gravels of the Hanford H1 and H2 formations make up most of the unsaturated zone. The unsaturated zone is relatively thick in the vicinity of the IDF, approximately 100 m (300 ft). The aquifer underlying the IDF is very transmissive such that it takes very large changes in natural recharge rates to significantly change the water table elevation. Water was liberally used during operations for various purposes. Recharge rates to the aquifer were considerably higher for decades before site cleanup efforts began because of anthropogenic water. Most of the previous man-made recharge that caused the water table to rise in the Central Plateau area ended in the mid-1990s. Natural recharge is highly dependent on the soil type and the presence of vegetation. Fine-grained sediments enhance storage and evaporation whereas coarse-grained sediments enhance recharge. Natural recharge is estimated to be in the range of 1 to 10 millimeters (0.04 to 0.4 in) per year.

Tritium (^{3}H), ^{99}Tc, and ^{129}I are some of the more mobile radionuclides that can move relatively quickly through the unsaturated zone, while ^{60}Co, ^{137}Cs, and uranium isotopes are some of the more significant radionuclides that move more slowly in the unsaturated sediments. Under ambient recharge rates, the travel time of mobile radionuclides through the vadose zone is estimated to be greater than 500 years. Sediment in the unsaturated zone in some portions of the 200 Areas is contaminated due to the release or discharge of radioactive liquid waste by different sources. DOE estimated that 1.5 to 1.7 billion cubic meters (m^3) (396 to 449 billion gallons (gal)) of effluent were released to the Hanford Site soils (PNNL-SA-32152, 1999). The area where the IDF is located did not receive past releases.

The saturated zone beneath the Hanford Site consists of the upper unconfined aquifer and the deeper basalt-confined aquifer. The saturated zone (from dilution) can significantly reduce the concentrations of radionuclides to which a receptor may be exposed. The basalt-confined aquifer consists of less permeable basalt flows but also contains relatively permeable sedimentary interbeds. The horizontal hydraulic conductivities of the interbeds can be about five orders of magnitude higher than most of the interior basalt flow which can range between 10^{-9} m/s to 10^{-15} m/s (3×10^{-9} ft/s to 3×10^{-15} ft/s). Exposures at the margins of the Pasco Basin is the likely source of recharge to the basalt-confined aquifer. The basalt-confined aquifer generally flows toward the Columbia River. Groundwater information indicates some vertical communication of the basalt-confined aquifer with the unconfined aquifer system above.

The unconfined aquifer system is within the undifferentiated H3/CCU/RF unit that overlies the basalt bedrock. The saturated thickness of the unconfined aquifer on the Hanford Site can range from greater than 60 m (~200 ft) to 0 m (0 ft) where it pinches out along the flanks of the basalt ridges. Long-term aquifer thickness in the vicinity of the IDF is around 10 m (30 ft). The unconfined aquifer at the Hanford Site is recharged in the elevated regions near the western boundary of the Hanford Site, and water generally flows in an eastern and northern direction towards the Columbia River. The Columbia River is the primary discharge area for groundwater. The natural direction of flow beneath the IDF is toward the southeast; however, in the past, groundwater flowed in a northern direction due to water mounding from artificial recharge during operations at the Hanford Site. The gradient is predicted to remain very flat under the IDF (approximately 2×10^{-5} m/m). The gravels and sands of the H3/CCU/RF unit have relatively high horizontal saturated hydraulic conductivity values in the range of thousands of meters per day such that groundwater flow velocities are high compared to flow velocities in the vadose zone.

The groundwater quality of the Hanford Site has been impacted by radiological and chemical contaminants resulting from past operations. Wastewater discharge from cribs and ponds, ditches, injection wells, spills, leaking waste tanks, and burial grounds have impacted the groundwater quality. Radioactive decay, chemical degradation, and dispersion will reduce the concentration of these contaminants. However, less mobile contaminants are present in the unsaturated zone and will eventually move downward into the saturated zone. Contaminants could migrate in the aquifer from other releases or sources. DOE has a program that is addressing groundwater cleanup that is outside the scope of this TER. There is little to no contamination present in the vadose zone soils under the IDF.

2.1.5 The 200 Areas in the Central Plateau of the Hanford Site

The Hanford Site has occupied 1,520 km^2 (586 mi^2) along the Columbia River near Richland, Washington since 1943. Operations to make the raw materials for nuclear weapons for national defense continued until the late 1980s. In 1989, DOE's mission at the Hanford Site shifted from weapons material production to waste management and environmental cleanup.

The cleanup of the Hanford Site involves management of more than 200 million liters (L) (53 million gal) of radioactive and chemically hazardous waste in 177 underground storage tanks, about 750,000 m^3 (25 million ft^3) of buried or stored solid waste, as well as spent nuclear fuel and plutonium in various forms. The massive underground storage tanks were built throughout Hanford's 200 Areas in a series of groups (known as tank farms) to hold the wastes, ranging in capacity from 208,200 liters (55,000 gallons) to more than 3,785,000 liters (1,000,000 gallons). Most tanks included a carbon steel shell surrounded by reinforced concrete. The materials inside waste tanks consist of liquids, gases, semi-solids, and solids. No new waste from plutonium production has been added to the tanks in many years, but many of the tanks remain in use today. Eighty-three single-shell tanks are in the 200 West Area and another 66 single-shell tanks are found in the 200 East Area, including the 16 single-shell tanks in the tank farm at WMA C. An estimated 67 of these tanks leaked some of their contents into the ground, and some of this liquid waste migrated through the vadose zone and has reached the groundwater. Since the single-shelled tanks have been shown to leak, priority has been given to transferring waste out of the single-shelled tanks with some of the wastes going into double-shell tanks.

The Separations Area encompasses the 200 East and 200 West Areas which occupy approximately 51 km^2 (20 mi^2) in the Central Plateau, near the center of the Hanford Site. The WTP is currently under construction within the 200 East Area. Waste recovered from the 200 Area tank farms will be separated into three major categories: (1) HLW that transferred to the WTP for vitrification and disposal in a geologic repository or other acceptable facility, (2) LAW that is vitrified into glass (i.e., VLAW) and will be disposed in the near surface in the IDF, and (3) supernatant that will be treated using a process demonstrated by the TBI.

2.2 Disposal Facility Description

This section provides a summary of the IDF design including the key design characteristics and safety functions. The performance of the IDF is balanced between the disposal site (natural) and the disposal facility (man-made). The IDF is in the central portion of the site in an area referred to as the 200 East Area of the Hanford Site. The IDF facility occupies only a fraction of the area of the 200 East Area. The IDF facility is a large, near-surface disposal facility with sloped sides, a liner and leachate collection system, and a planned surface barrier that will be installed after closure of the facility. The facility is constructed over the thick unsaturated zone in the 200 East Area. The performance of the facility is derived from a combination of natural site characteristics and engineered barriers. Figure 2-2 is a photograph of the IDF facility.

2.2.1 Integrated Disposal Facility

Currently the IDF has two disposal cells: one for LLW and one for mixed waste. The facility may be expanded in the southern direction to eventually include as many as six cells. After expansion, the length of the facility in the north-south direction at the floor of the cells will be approximately 420 m (1,390 ft). The current trench bottom is 110 m (360 ft) in length. The width of the facility in the east-west direction at the floor of the cells is approximately 330 m (1,090 ft). DOE provided the technical specifications for the design (RPP-18489, 2004). Based on the dimensions, the disposal facility could accommodate up to 900,000 m^3 (32,000,000 ft^3) of waste. The disposal cells are approximately 20 m (70 ft) thick and will be covered by up to 15 m (50 ft) (in the center) of soil and other materials to provide an engineered cover at closure.

Each disposal cell has liners and a leachate collection system to collect potential releases during operations. The bottom of the facility has a primary and secondary geotextile liner. There is an admix layer beneath the secondary liner. The primary liner is covered by a compacted operational layer. A system of liners, gravel drainage layers, and piping will collect leachate at sumps. Primary and secondary drainage layers utilize geocomposite materials on top of high-density polyethylene (HDPE). Figure 3-104 on page 3-202 of the IDF PA shows detailed design drawings of the liner system (RPP-RPT-59958, 2018). The piping is 30 cm (12 in) diameter slotted pipes located along the north-south centerline of each cell. Slopes of 1% are used to ensure that leachate flows to the sumps. Pumps will then be used to remove the

Figure 2-2 Aerial Photograph of the IDF
[Figure 2-6 in the IDF PA (RPP-RPT-59958,2018)]

leachate to storage tanks for temporary holding prior to disposal. See Figure 2-2 for an aerial view of the storage tanks.

Waste will be placed inside the disposal cells in layers with soil or sand used as backfill to provide structural support, to help reduce moisture contact with the waste containers through capillary action, and to provide shielding for workers as more waste is placed in the facility. Different waste containers will be used for the disposal of various waste streams[11]. The main waste types (see Section 2.3) are vitrified LAW glass from the WTP, failed glass melters from processing LAW, solid LLW, and solid mixed low-level waste (MLLW). The solid LLW and solid MLLW are secondary waste streams generated by waste processing that are envisioned to be solidified or encapsulated with cement.

Glass will be produced in stainless-steel cylindrical containers that are 1.22 m (4.00 ft) in diameter and 2.29 m (7.51 ft) tall. The containers have necks at the top such that the total cylindrical volume is not available for waste nor is that volume filled 100% with glass. The estimated volume of glass held by each container is 2.17 m^3 (76.6 ft^3). Glass melters, due in part to the high temperatures and chemical environment, do not have an infinite lifetime. Melters periodically fail or need to be replaced to operate efficiently. The melter cavity itself is somewhat smaller than the final glass tank size for disposal which is approximately 5.64 m (222 in) long by 2.74 m (108 in) wide by 1.12 m (44 in) high. Melter overpacks are carbon steel

[11] The specific container designs had not been finalized at the time the PA was developed, however, preliminary design information was provided by DOE. Final designs may differ from those described.

containers designed to provide shielding, contamination control, and structural support to allow direct burial of spent melters. The dimensions of the melter overpacks are considerably larger than the melters as it is assumed the melters will be grouted in the overpacks.

The container type for the LLW and MLLW are expected to be 55-gallon drums and B-25 boxes, which are fabricated from carbon steel and may be painted or galvanized. The void fraction in loaded waste containers is required to be less than 10 percent. To support potential grouting of LLW and MLLW containers, ¼ height International Standards Organization (ISO) disposal containers may be used. Waste will be loaded into the disposal cells in layers called “lifts”. Waste containers are placed as close as practical to each other (10 to 20 cm (4 to 8 in)). A soil layer ~ 1 m (3 ft) thick compacted at or below the optimum moisture content will be placed on top of each layer of waste packages. Concrete shield blocks used during waste emplacement would remain in the trenches at closure.

After waste emplacement is completed, DOE will use a closure cap system. The specific design for the closure cap has not been completed; however, DOE has experience developing and testing surface barriers at the Hanford Site (PNNL-18845, 2011). A modified Resource Conservation and Recovery Act (RCRA) Subtitle C barrier design is the baseline design. The barrier will have multiple design goals including to limit intrusion into the waste, to provide hydrologic containment for the waste, and to limit release by other surface processes such as erosion. The specific choice of barrier materials, barrier thickness, and other design parameters and features have not yet been selected. The minimum depth from the cover surface to the waste will be 5 m (16 ft), though preliminary designs suggest a thickness of the cover that could be as much as 16 m (50 ft) at the center to accommodate a proper slope of the cover. A barrier overhang will be used to control potential water infiltration problems at the edge of the cover. DOE plans to use many layers of differing materials to accomplish the design goals.

2.2.2 Key Design Characteristics and Safety Functions

The overall performance of the disposal system is a function of natural and engineered barriers. Figure 2-3 shows the key design features of the engineered system as well as the relevant safety functions of the natural and engineered systems. Low flow through the waste, slow transport to the aquifer, and dilution of contaminants when they reach the aquifer are the key safety functions. The engineered wasteforms have a significant role in limiting future release of radioactivity (See Figure 2-4). In addition, the characteristics of potential receptors and the likelihood that, in the future, a receptor will contact waste or radioactivity released from the waste are also important characteristics to consider. Demography and land use are discussed in Section 2.4.

Because of the diversity of the wastes that have been generated and will be generated in the future, a variety of different wasteforms are necessary. The glass wasteform is designed to provide mechanical stability, low permeability, and a slow rate of dissolution of the glass matrix. The glass matrix is the solidified borosilicate glass that contains the radioactivity from the liquid waste. Though the glass is held inside stainless-steel containers, DOE did not credit the stainless steel as a barrier in their evaluation. Some wastes, such as secondary solid wastes, will be cementitious wasteforms. The cementitious wasteforms provide mechanical stability, low permeability, low diffusion rates, and high sorption of select radionuclides.

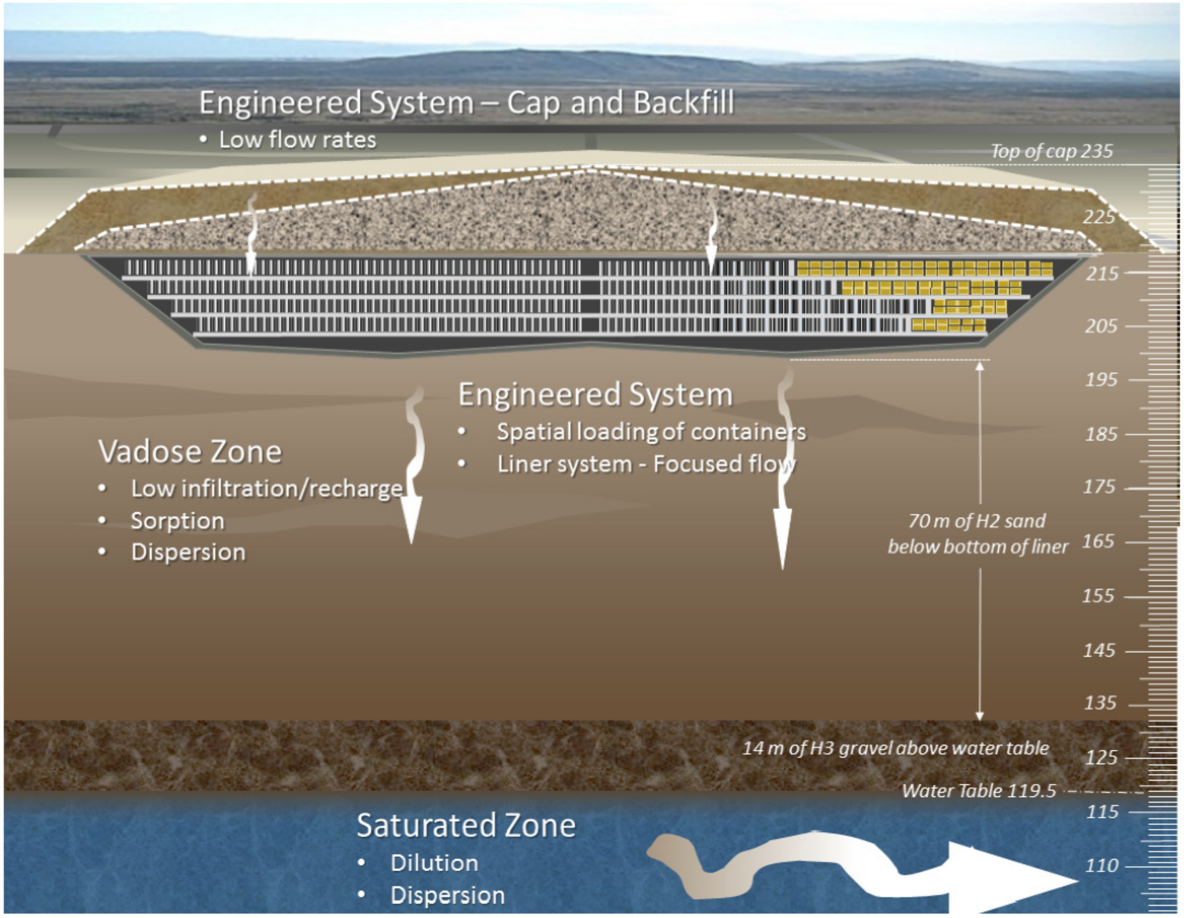


Figure 2-3 Cross-Sectional View of the IDF and Disposal Site
[Figure 4-2 in the IDF PA (RPP-RPT-59958,2018)]

2.3 Overview of Wastes to be Disposed

The Hanford Site has 177 underground storage tanks with more than 200 million L (53 million gal) of radioactive and chemically hazardous waste. To close the storage tanks, the waste must be removed and treated. DOE intends to dispose of the LAW stream from the tanks at the IDF.

The total amount of radioactivity remaining in the 177 underground storage tanks is approximately 5.1 x 10^{18} Bq (1.39 x 10^{8} Ci) as of 2017, when the short-lived equilibrium decay products for ^{137}Cs and ^{90}Sr are included. Cesium and strontium (and their decay products) comprise approximately 99 percent of the radioactivity. The different waste phases contain differing amounts of each isotope. Radionuclides that are soluble in-tank waste tend to be present in higher amounts in the supernate and saltcake, whereas the insoluble radionuclides tend to be present in higher amounts in the tank sludge. Whereas the fraction of ^{99}Tc, ^{129}I, and ^{137}Cs present in the supernate is 0.42, 0.45, and 0.55, respectively, the fraction of ^{90}Sr, ^{238}Pu,

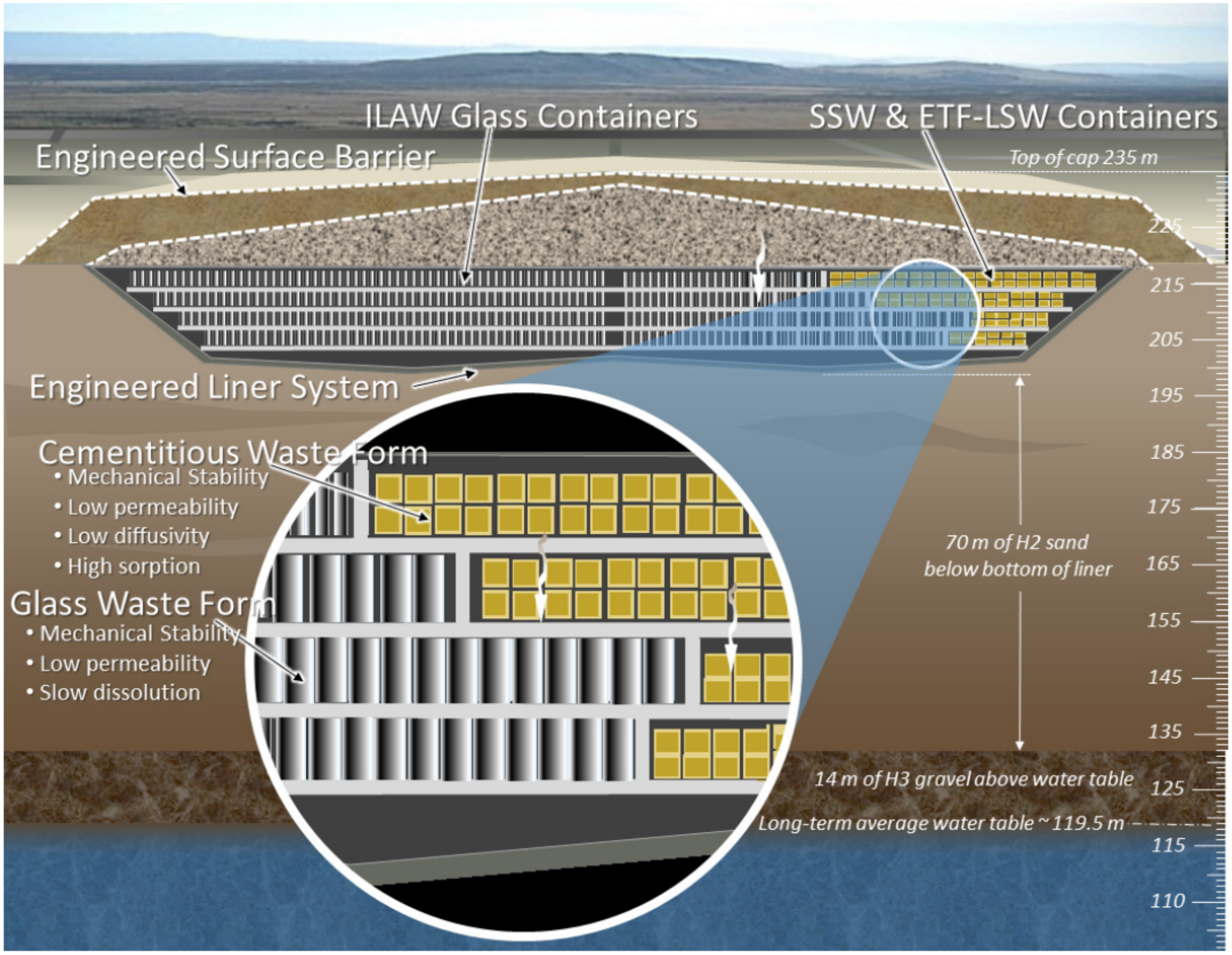


Figure 2-4 Cross-Sectional View of the Integrated Disposal Facility Showing the Conceptual Design for Waste Emplacement
[Figure 3-111 in the IDF PA (RPP-RPT-59958,2018)]

and ^{237}Np present in the supernate are much lower at 0.01, 0.02, and 0.05, respectively. DOE maintains an inventory of waste in a database referred to as the Best Basis Inventory (BBI). The inventory estimates are based off the best available information from many different sources including sampling, computer models, and process knowledge.

DOE describes the type and quantity of waste to be disposed in the IDF in the Draft WIR Evaluation for VLAW (DOE, 2020a) and the IDF PA (RPP-RPT-59958, 2018). The IDF PA states that the IDF will receive multiple waste streams generated by the WTP from the vitrification process. These waste streams include LAW glass from the WTP, failed glass melters from processing the LAW, secondary solid waste (SSW) produced during glass processing at the WTP, and SSW generated from the effluent treatment facility (ETF). The IDF PA also notes that DOE intends to dispose of additional waste streams generated from processes other than the WTP process at the Hanford Site at the IDF. The additional waste streams include the Fast Flux Test Facility (FFTF) decommissioning waste, secondary waste management LLW and MLLW, and onsite non-CERCLA (Comprehensive Environmental

Response, Compensation, and Liability Act), non-tank LLW, and MLLW. An overview of the different types of waste that will result from processing the waste in the tanks is shown in Figure 2-5. In the IDF PA analyses, DOE evaluated disposal of 130,000 glass canisters as well as the other waste streams described above. Table 2-1 is an overview of the wastes volumes that will be disposed at the IDF.

The Draft WIR evaluation for VLAW evaluated only a subset of the waste to be disposed in the IDF (i.e., direct-feed low-activity waste [DFLAW]). The DFLAW waste consists of low-activity waste from the Hanford tanks that is sent directly from the waste tanks to the vitrification facility. In the DFLAW approach, the supernate in the tanks (i.e., the upper-most layer of tank waste that has lower concentrations of long-lived radionuclides) will be separated and pretreated to generate a LAW stream. The first step in the DFLAW process includes in-tank settling and removal of the supernate (including dissolved saltcake and interstitial liquids) from the tanks by decanting. That liquid will then be then passed through a tank side cesium removal (TSCR) unit, and the resultant liquid will be vitrified into glass. Approximately 8.9×10^7 L (2.3×10^7 gallons) of liquid will be processed under the DFLAW process producing approximately 13,500 glass canisters. DFLAW generated vitrified waste represents about 10% of the vitrified waste that will be generated from processing all LAW.

While DOE only included DFLAW in the WIR determination, the NRC staff considered all waste streams resulting from processing that can produce significant contribution to doses from disposal at the IDF to be within the scope of the evaluation in this TER. See Section 1.4 for a more detailed discussion of the scope of the waste streams included in this TER.

2.4 Demography, Natural Resources, and Land Use

The demography, presence of natural resources, and past, current, and future land use are important factors in the evaluation of a site for radioactive waste disposal. Historically, Native Americans fished, hunted, and settled along the Columbia River and in the Hanford area for thousands of years. People that lived on the present-day Hanford Site were relocated when the United States designated Hanford as a location essential to the development of atomic bombs during World War II. Today, most of the land south of the Hanford Site is urban and the nearest population centers are the three cities of Richland, Kennewick, and Pasco (frequently called the Tri-Cities). The cities of Kennewick, Richland, and West Richland and most of the Hanford Site are within Benton County, which has increased in population from 112,560 in 1990 to 142,475 in 2000, a 26.6% increase in 10 years. The unincorporated population of Benton County was 33,227 in 2000. Population density and population growth rates are important because they may influence the likelihood that a closed waste disposal site is inadvertently used in the future.

The presence of valuable natural resources increases the likelihood that someone may inadvertently contact buried waste or contamination while trying to recover those resources. When waste or contamination is deeper than 5 m (16 ft) below the present-day land surface, mining, oil and gas recovery, and water extraction are the primary activities that could result in the inadvertent extraction of buried waste. Crushed rock, gravel, sand, and silt are currently the most commercially viable mineral resources in the land surrounding the Hanford Site. Deep natural gas has not yet been successfully produced in the vicinity of the Hanford Site. Installation of wells for groundwater is relatively expensive because groundwater is deep. Wells are installed for irrigation; however, the neighboring rivers are the preferable source for water.

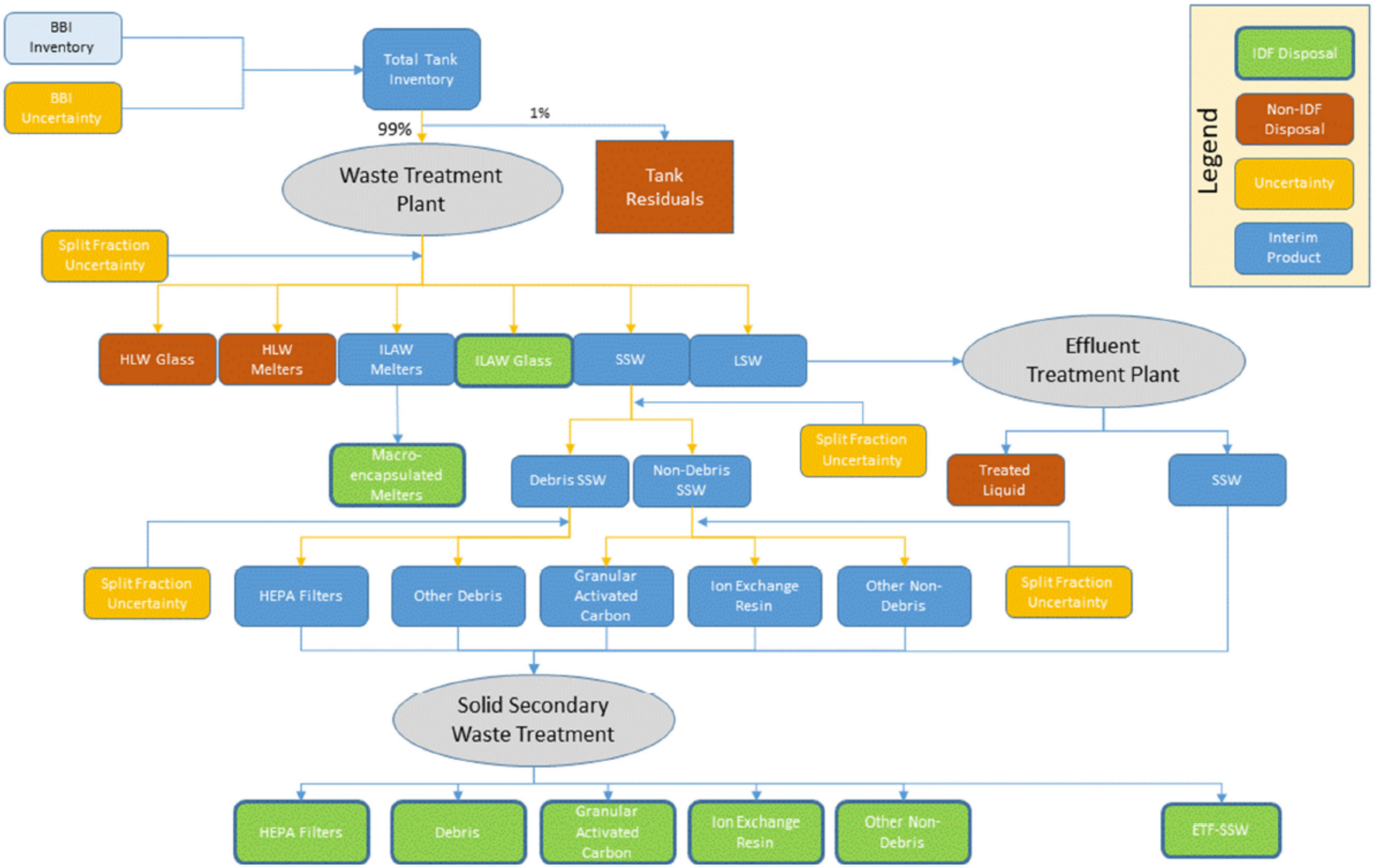


Figure 2-5 Overview of Wastes Produced from Management of Wastes Remaining in Storage Tanks
Note: DOE uses the terms "ILAW Glass" and "VLAW" interchangeably, DFLAW is 10% of VLAW
[Figure 3-117 in IDF PA (RPP-RPT-59958, 2018)]

Table 2-1 Waste Types and Volumes to be Disposed at IDF

Waste Stream	Container Type	Number of Containers	Waste Volume (m^3)
Immobilized Low Activity Waste (ILAW) glass	Stainless-steel	130,584	278,797
ILAW melters and overpack	Steel	49	373
ETF-generated LSW	Various		18,900
Total SSW (see below)	Various		41,447
High-efficiency particulate air (HEPA) filters	Steel drum	7,130	1,832
Ion-exchange resins	High integrity container	112	686
LAW granular activated carbon (GAC)	Steel drum	4,424	1,137
Silver mordenite	Steel drum	403	104
Other debris	Steel drum and boxes	60,309	26,546
Secondary waste management	Various		9,489
Fast Flux Test Facility	Various		1,030
Non-CERCLA, non-tank	Various		623

[adapted from Table 3-26 in the IDF PA (RPP-RPT-59958, 2018)]

The land use classification around the Hanford Site varies. At the Hanford Site, storage of radioactive material along with cleanup of radioactive waste in facilities, soils, and groundwater are major activities. Adjoining lands to the west, north, and east of the Hanford Site are principally range and agricultural land. Much of the land to the north and east is irrigated cropland. The Columbia River is a large natural water resource for the area. A reclamation project provides water that is transported via canals to the areas north and east of the Columbia River. Near the Yakima River and west of the Hanford Site, land is also used for irrigated agriculture. Columbia River water is used by various facilities at the Hanford Site and the cities of Richland, Pasco, and Kennewick.

DOE intends to continue to control access to the Hanford Site for the foreseeable future. DOE developed a comprehensive land use plan in 1999 (DOE, 1999c). The plan provides land use maps, land use designations for permissible use, land use policies, and implementing procedures that would govern the review and approval of future land uses. The area where the IDF is located was designated Industrial-Exclusive. In the plan, DOE describes five types of institutional controls including warning notices, entry restrictions, fencing, land use management restrictions, and groundwater use management restrictions. For evaluation of potential impacts under DOE Order 435.1, DOE assumed the institutional controls will not be effective beyond 100 years after site closure, although DOE indicated they have no intention of releasing site control at that time.

3 CRITERION A – Key Radionuclides Removed

Demonstrating that key radionuclides have been or will be removed is the first step in the WIR determination process, or Criterion A. First, the "key" radionuclides present in the waste need to be identified, with the understanding that the list of key radionuclides may be longer than the list of the risk-significant radionuclides. Next, DOE may demonstrate that removal of key radionuclides to the maximum extent practical will be achieved.

3.1 Key Radionuclides

The criterion associated with key radionuclide removal in DOE Manual 435.1-1 states that wastes:

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

The identification of key radionuclides is important to establish which radionuclides must be processed or removed to achieve protection of public health and safety.

3.1.1 DOE Identification of Key Radionuclides

DOE viewed key radionuclides to be those that, using a risk-informed approach, contribute most significantly to radiological dose to workers, the public, and the environment. To identify key radionuclides applicable to VLAW wastes, DOE included those radionuclides identified in the IDF PA (RPP-RPT-59958, 2018) as important to demonstrating the performance objectives of 10 CFR Part 61, Subpart C as well as those isotopes identified in Table 1 and Table 2 of 10 CFR 61.55. DOE developed the list of key radionuclides by examining PA modeling results for the air, groundwater, and inadvertent intruder pathways. DOE also considered all pathways combined.

The air pathway is the primary contributor to dose during the initial 1,000-year post-closure period, with ^{129}I being the principal radionuclide. The air pathway analyses consider the effects of gaseous radionuclides (that are tracked in the tank waste inventory) after they are disposed in the IDF. These include tritium (^{3}H), ^{14}C, and ^{129}I, and specifically excludes the effects of radon and its progeny in air. The peak air pathway dose was calculated to be 1.9 x 10^{-3} mSv/yr (1.9 x 10^{-1} mrem/yr). Radon is excluded from the air pathway because it has a separate performance criterion. The DOE requirements in DOE Manual 435.1-1, Chapter IV.P.(1), read as follows for radon flux:

> *"Release of radon shall be less than an average flux of 20 pCi/m^2/s (0.74 Bq/m^2/s) at the surface of the disposal facility. Alternatively, a limit of 0.5 pCi/l (0.0185 Bq/l) of air may be applied at the boundary of the facility."*

The groundwater pathway contributes a negligible amount to dose during the first 1,000 years and then provides the dominant contribution during the 1,000- to 10,000-year post-closure period, with ^{99}Tc and ^{129}I contributing the most. The peak dose from the groundwater pathway during this period was less than 2 x 10^{-2} mSv/yr (2 mrem/yr).

The key radionuclides for the acute and chronic inadvertent intruder pathway are ^{137}Cs, ^{3}H, and ^{90}Sr, though other radionuclides may contribute depending on the scenario and pathway analyzed (please see Section 4.11 for discussion of inadvertent intruders). The projected acute dose to the well driller that intrudes into the facility 100 years after closure was 0.093 mSv (9.3 mrem). The projected chronic dose to the intruder 100 years after IDF closure under the rural pasture resident scenario was 0.433 mSv/yr (43.3 mrem/yr). Consideration of all-pathways did not identify additional radionuclides because the doses from the different scenarios did not occur at the same time to a significant extent.

Table 1 and 2 of 10 CFR 61.55 provide long- and short-lived radionuclides used to classify LLW. DOE indicated that the radionuclides provided in Table 1 and Table 2 of 10 CFR 61.55 were key radionuclides except for ^{94}Nb. DOE indicated that ^{94}Nb does not exhibit significant activity in Hanford tank waste and, therefore, was not considered to be a key radionuclide. Table 3-1 provides the radionuclides identified as key radionuclides by DOE (DOE, 2020a).

3.1.2 NRC Evaluation of Identification of Key Radionuclides

DOE's approach to identify key radionuclides was consistent with the NRC staff's interpretation of key radionuclides. DOE considered key radionuclides to be those that contribute most significantly to radiological dose to workers, the public, and the environment. DOE summarized their PA results for different pathways and exposure scenarios and used this information to develop their list of key radionuclides. The list was then supplemented by any additional radionuclides found in the 10 CFR Part 61.55 waste classification tables (Table 1 and 2).

The NRC staff evaluated DOE's approach to identifying key radionuclides and conclude that the approach is reasonable. Additional considerations with respect to identifying key radionuclides are found below. These considerations do not impact the NRC staff's conclusion that DOE's approach was reasonable as applied to waste to be disposed at the IDF.

The DOE approach identified radionuclides that may impact the groundwater and air pathways. The radiological doses to hypothetical receptors during the DOE compliance period (1,000 years) are small fractions of the dose limits. The radionuclide ^{129}I is the most significant radionuclide for the air pathway, but as discussed later in this document the air pathway analyses were conservative. If the radiological doses are very small and there is confidence in the magnitude of the doses, then an argument could be made that none of the radionuclides are significant for the air pathway. Including additional key radionuclides does not have a detrimental impact on public health and safety but, in some cases, it could result in misperceptions and the misapplication of resources for additional waste removal or remediation.

DOE conservatively included radionuclides that had the most significant impacts in the 1,000 year to 10,000-year timeframe. This is good practice as it helps to account for uncertainty in the projected timing of future radiological doses. Radionuclides that are identified as significant to the results of sensitivity analyses should be included on a case-by-case basis. Some sensitivity analyses may represent a range of expected results whereas others are very low probability scenarios. It is appropriate for DOE to use subjective judgment to include additional key radionuclides based on the results of sensitivity analyses.

Table 3-1 Key Radionuclides Identified by DOE

Radionuclide	10 CFR 61.55 Long-Lived Radionuclides	10 CFR 61.55 Short-Lived Radionuclides	Radionuclides Important to Performance Assessment [a]
^{3}H[b]		X	
^{14}C[b]	X		
^{60}Co		X	
^{59}Ni	X		
^{63}Ni		X	
^{90}Sr		X	X
^{99}Tc[c]	X		X
^{129}I[c]	X		X
^{137}Cs		X	X
^{228}Rn			X
^{229}Th			X
^{232}Th			X
^{234}U			X
^{238}U			X
^{237}Np	X		X
^{238}Pu	X		
^{239}Pu	X		X
^{240}Pu	X		
^{241}Pu	X		
^{242}Pu	X		
^{241}Am	X		
^{243}Am	X		
^{242}Cm	X		
^{243}Cm	X		
^{244}Cm	X		

[a] The IDF PA (RPP-RPT-59958, *2018)* encompasses other waste in addition to the DFLAW addressed in the *Draft WIR Evaluation for VLAW*.

[b] ^{3}H and ^{14}C are contained in Hanford tank waste but are not present in VLAW. These isotopes partition into secondary waste during vitrification (IDF PA, Table 3-27).

[c] ^{99}Tc and ^{129}I are contained in Hanford tank waste. The LAW Vitrification Facility is designed to maximize the capture of these radionuclides in the vitrified wasteform. The LAW Vitrification Facility off-gas system is designed to recycle and/or capture a portion of volatile radionuclides (including ^{99}Tc and ^{129}I) which are volatilized.

DOE indicated that ^{94}Nb was not included as part of the key radionuclides because it did not exhibit significant activity in Hanford tank farm wastes. The NRC staff evaluated the inventory of ^{94}Nb and how it was derived. While the inventory of ^{94}Nb is low relative to other isotopes, it does have a long half-life of 20,000 years. The limit that NRC provides in 10 CFR 61.55 is for ^{94}Nb in activated metals with no limit provided for ^{94}Nb that is not in activated metals. At the time the limit was developed, NRC anticipated that commercial disposal (without reprocessing) of ^{94}Nb would not occur unless the ^{94}Nb was in activated metals. The limits for isotopes that are not in a metal form were set to be a factor of 10 less than isotopes that were in activated metals. DOE stated that the total amount of ^{94}Nb created from 1944 to 1989 in all Hanford reactors is about 3.7 x 10^{-3} MBq (0.1 Ci). The isotope ^{94}Nb is primarily produced in reactors from activation of natural niobium in stainless steel and Inconel, neither of which were used at Hanford in the fuels that were reprocessed. DOE determined that ^{94}Nb is not a key radionuclide in VLAW. The NRC staff reviewed the reference RPP-13489, *Activity of Fuel Batches Processed Through Hanford Separations Plants, 1944 Through 1989, Table H-1* to verify the 3.7 x 10^3 MBq (0.1 Ci) estimate of ^{94}Nb for all fuel batches (RPP-13489, 2002). Because the total activity of ^{94}Nb is estimated to be small due to the known type of stainless steel used in the reprocessed fuel, it is reasonable to not include ^{94}Nb as a key radionuclide. The NRC staff note that ^{94}Nb was not included in the Best Basis Inventory (BBI) because DOE did not have the analytical capability to measure it.

The NRC staff's conclusions on the identification of key radionuclides are based on the results of the technical analysis (e.g., doses to offsite receptors and inadvertent intruders) and the associated assumptions with respect to that analysis. Identification of key radionuclides is an iterative process that DOE would need to revisit if the results of the technical analysis were to change significantly.

3.2 Removal to the Maximum Extent Practical

After key radionuclides are identified, then a demonstration that they have been removed to the maximum extent technically and economically practical must be provided to determine that the wastes may be managed as LLW. DOE plans to remove key radionuclides from VLAW using a series of steps which employ a variety of methods and technologies.

3.2.1 DOE Analyses of Removal to the Maximum Extent Practical

DOE's analyses for the removal of key radionuclides to the maximum extent practical included a description of the technologies and their effectiveness. DOE discussed the technical practicality of the technologies that were or will be used and their economic practicality, as well as the scope of technologies that were considered.

3.2.1.1 Waste Removal Processes and Performance

DOE will rely on settling, decanting, filtration, and ion exchange to remove most of the key radionuclides. Under the long-term caustic storage conditions within the underground storage tanks, the waste has separated into insoluble solids, soluble solids, and liquids (Figure 3-1). Figure 3-1 shows the volume of waste remaining in the tanks in each waste component as well as the associated total radioactivity (as of July 2017, to convert to MBq multiply Ci by 37,000). DOE will dissolve the saltcake that remains in the single-shell tanks and transfer it along with interstitial liquid into the double-shell tank system. The supernate remaining in single-shell tanks has been moved to double-shell tanks. In the DFLAW approach, DOE will treat the supernate (including dissolved saltcake and interstitial liquids) that contains mostly short-lived radionuclides (primarily ^{137}Cs and its progeny ^{137m}Ba) but also some long-lived radionuclides. The supernate and saltcake (soluble solid phase) are roughly 80 percent by volume of the tank waste. These phases contain a lower fraction, on a relative basis, of the total radioactivity at approximately 44 percent.

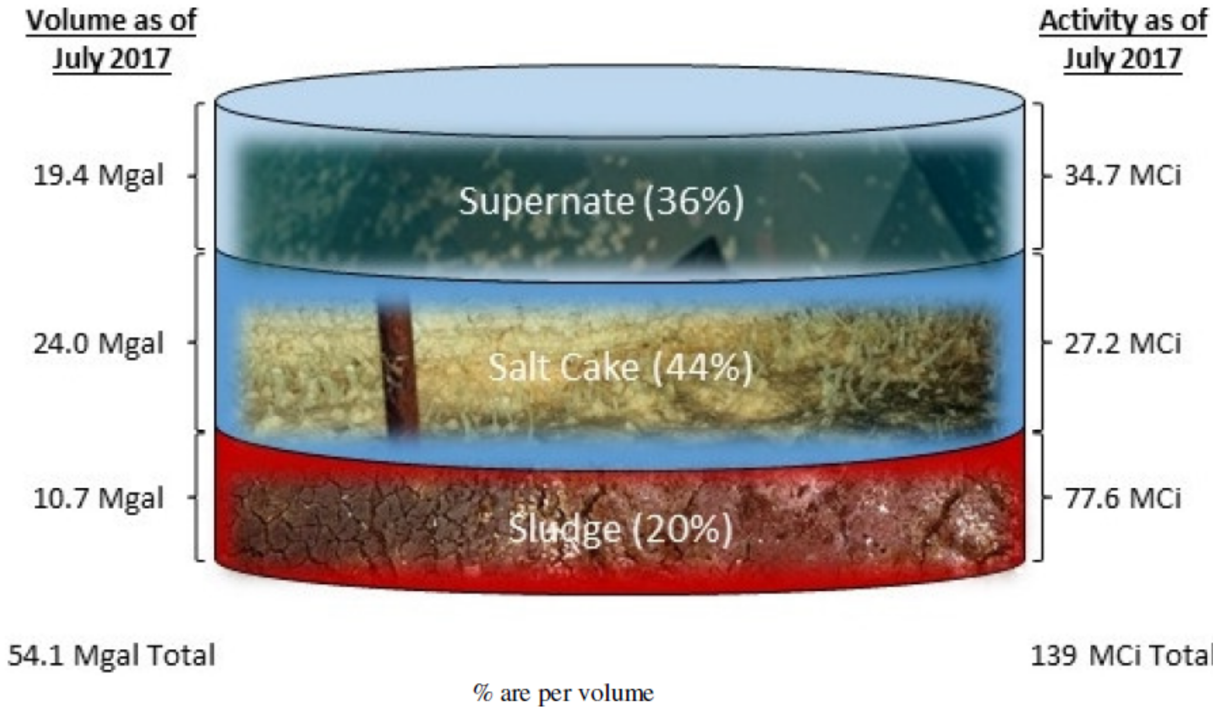


Figure 3-1 Tank Waste Volumes (by form) and Estimated Total Radioactivity
[Figure 4-1 in (DOE, 2020a)]

Waste that has been moved to the double-shell tanks will undergo additional settling and decanting to separate supernate and dissolved saltcake from the insoluble solids. The insoluble solids tend to settle to the bottom of the tanks and contain higher proportions of long-lived actinides. The majority of the ^{137}Cs, ^{99}Tc, ^{129}I, ^{14}C, and ^{3}H is contained in the soluble fraction of the tank waste. The other soluble radionuclides are ^{60}Co, ^{59}Ni, and potentially ^{90}Sr. The radionuclide ^{60}Co has a short half-life (approximately 5 years) and, therefore, is not a contributor to dose after closure of the IDF. The radionuclide ^{59}Ni, with a half-life of 1.01×10^5 years, is present in very low concentrations in the tank waste and is also an insignificant contributor to dose after IDF closure (see Table 7-13 of the IDF PA).

The liquid waste will be filtered to remove remaining insoluble radionuclides to the extent practical. Most of the radionuclides present in the liquid resulting after filtration will be those radionuclides that are partially or completely soluble, including ^{137}Cs, ^{99}Tc, ^{129}I, and possibly ^{90}Sr. In the Draft WIR evaluation for VLAW, DOE stated that most of the ^{90}Sr is insoluble, but it can be soluble in some tanks with a higher organic content (DOE, 2020a). Tanks with soluble ^{90}Sr are not currently planned to be part of the DFLAW campaigns but may be added in the future. In the IDF PA modeling, DOE included VLAW from tanks with soluble ^{90}Sr.

Following filtration, the ^{137}Cs, as well as any large fractions of Ca, U, ^{90}Sr, Np, and Pu, if present in soluble form, will be removed by passing the waste through three ion-exchange (IX) columns. The ^{137}Cs will be removed using three IX columns arranged in a lead, lag, polish configuration. In the Draft WIR Evaluation for VLAW, DOE provided the average decontamination factor for

^{137}Cs. The decontamination factor was calculated by dividing the total amount of ^{137}Cs that will enter the lead IX column by the total amount of ^{137}Cs that will exit the polish IX column for each DFLAW campaign. After use, the IX column media will be stored onsite until future disposition.

During wasteform production (i.e., vitrification), some radionuclides are further partitioned into different wasteforms. DOE indicated that ^{3}H and ^{14}C are not present in significant quantities in the VLAW as they partition to solid secondary waste during the vitrification process. In the Draft WIR Evaluation for VLAW, 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 wasteform. 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)" (DOE, 2020a). During the LAW vitrification process, the volatile components will be transported through the melter off-gas treatment system, 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 ^{99}Tc and ^{129}I in the liquid condensate from the SBS and WESP resulting from the off-gas system can be routed in three ways:

1) Recycle back to the LAW Vitrification Facility for blending with incoming waste feed
2) Return to the Hanford tank farms double-shell tanks (DSTs)
3) Purge via a tanker truck load-out station (RPP-RPT-58971, 2020)

The Effluent Management Facility (EMF) evaporator condensate will be sent to the Effluent Treatment Facility (ETF) for further treatment prior to disposal. The liquid effluent resulting from the treatment process will be discharged to a State-approved land disposal site. The removed solids will be incorporated into a cementitious wasteform and disposed at the IDF as secondary solid waste (SSW).

In Table 3-29 in the IDF PA DOE provided a summary of radionuclide inventories and how the inventory is distributed between different wasteforms for five different cases (the numbering of these cases started with Case 7) (RPP-RPT-59958, 2018). The two radionuclides that contribute most of the dose for releases to the groundwater pathway are ^{99}Tc and ^{129}I. DOE developed different inventory cases to examine different options for producing vitrified glass. The two aspects evaluated were the single-pass retention of volatile species and if recycling of the off-gas was used. In Case 7 (the base case), 99 percent of the ^{99}Tc is assumed to end up in the VLAW glass. The other inventory cases examined variations from the base case. Figures 3-2 and 3-3 provide the fraction of ^{99}Tc and ^{129}I for different inventory cases and wasteforms. As shown in Figure 3-2, in three of the cases (Case 7, Case 8B, and Case 9), about 40 percent of the inventory of ^{129}I ends up in the SSW. In Case 10A and Case 10B, nearly 80 percent of the ^{129}I ends up in either the ETF-Generated SSW or the SSW.

In response to the NRC staff's RAI, DOE explained that the inventory cases were completed to examine design choices and that based in part on the results of the analyses, that DOE will be using recycle of the off-gas from vitrification. This enhances the retention of ^{99}Tc and ^{129}I in the

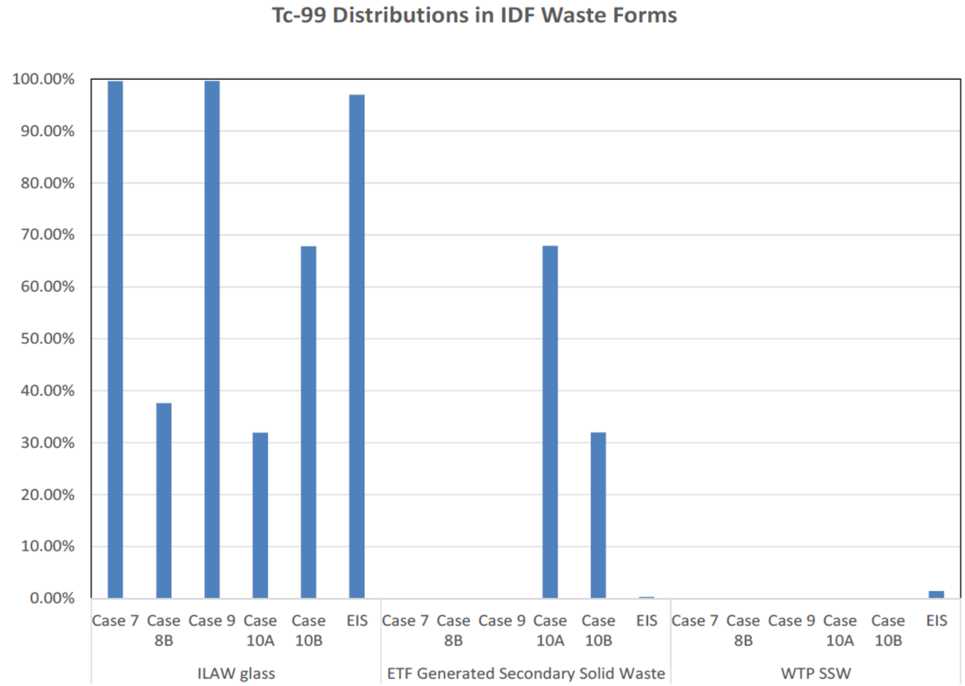


Figure 3-2 Distribution of ^{99}Tc for Different Inventory Cases and Wasteforms
[Figure 3-121 of the IDF PA (RPP-RPT-59958, 2018)]

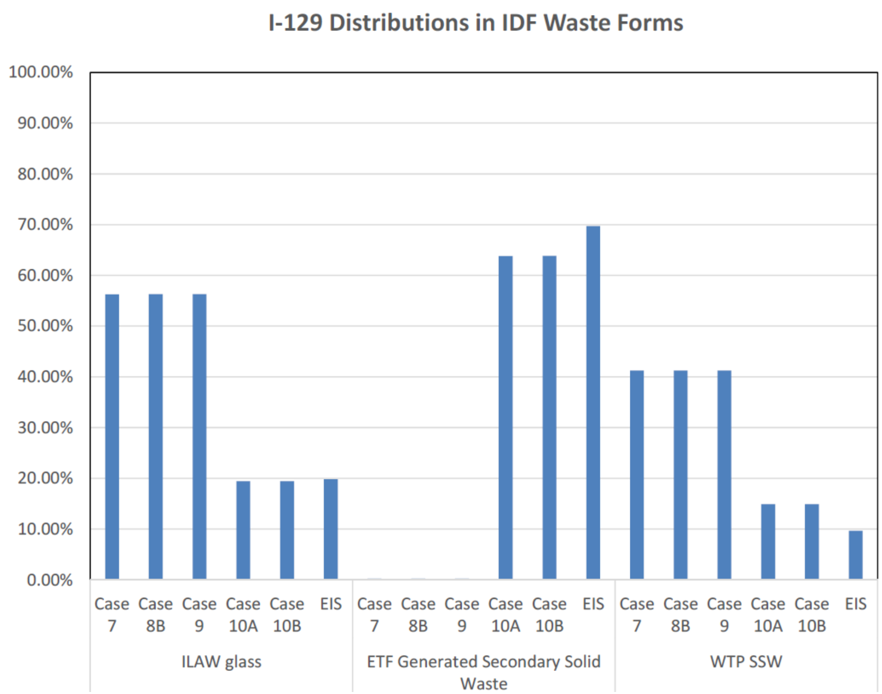


Figure 3-3 Distribution of ^{129}I for Different Inventory Cases and Wasteforms
[Figure 3-119 of the IDF PA (RPP-RPT-59958, 2018)]

vitrified glass. DOE also completed additional research to support higher single-pass retention values for volatile species (DOE, 2021c).

3.2.1.2 Alternative Treatment Technologies

DOE has previously conducted studies to examine technologies for removal of radionuclides from the Hanford tank wastes. These studies were mostly focused on technologies for removing radioactivity or select radionuclides from the waste in the tanks rather than on using different technologies or processing methodologies on the waste after it was removed from the tanks. These studies were summarized in a report referred to as the TBR (WHC-SD-WM-TI-699, 1996).

The evaluation DOE described in the TBR consisted of:

- Identifying individual technology options for radionuclide separations processes,
- Identifying the status of the technology,
- Defining the radionuclide removal efficiency, and
- Determining the cost of implementing the technology.

DOE indicated that economic practicality focuses on whether further radionuclide removal would be useful and sensible considering the overall benefit to human health, safety and the environment while compared to the costs of additional removal of key radionuclides.

DOE summarized the costs for the technology options identified in the TBR to remove radionuclides in Table 4-7 of the Draft WIR Evaluation for VLAW. The costs had not been adjusted to 2020 dollars. DOE found that Single-Cycle Cation Ion Exchange, and Selective Removal/Hydroxide Precipitation for TRU and ^{90}Sr were economically practical, but that the other alternative treatment technologies considered were not economically practical. The summary of costs for technically practical radionuclide removal technologies are shown in Table 3-2.

In October 2017, DOE conducted an external peer review of the Low-Activity Waste Pretreatment System (LAWPS)[12]. Consistent with the recommendations in the external review, DOE decided to pursue the DFLAW approach (84 FR 424, pp. 425). Various pretreatment approaches for DFLAW were analyzed and compared. Table 3-3 provides the comparison of economic practicality considerations between the prior approach for LAWPS and the TSCR approach. The first phase of the DFLAW approach will deploy a TSCR system utilizing crystalline silicotitanate (CST).

[12] RPP-RPT-60405, External Expert Review of the Low-Activity Waste Pretreatment System (LAWPS) Project

Table 3-2 Summary of Costs for Technically Practical Radionuclide Removal Technology Options

Technology	Economically Practical	Cost $/Ci
Single-Cycle Cation Ion Exchange, Selective Removal (^{137}Cs concentration > 0.05 Ci/L)	Yes	25
Single-Cycle Cation Ion Exchange, Selective Removal (^{137}Cs concentration < 0.05 Ci/L)	No	65
Single-Cycle Cation Ion Exchange	No	30
Second-Cycle Cation Ion Exchange	No	420
Hydroxide Precipitation for TRU and ^{90}Sr, Selective Treatment	Yes	63-128
Ferric Hydroxide Precipitation for TRU and ^{90}Sr, Selective Treatment	No	140-570
Solvent Extraction, TRUEX, PUREX	No	800,000

Table 3-3 Comparison of Low-Activity Waste Pretreatment Options

Economic Practicality Considerations	LAWPS Approach	TSCR Approach
Settling to remove insoluble long-lived radionuclides > 50%	Yes	Yes
Separation of supernate by decanting	Yes	Yes
Filtration	Yes – crossflow	Yes – dead end
Cs removal > 99%	Yes – elutable	Yes – nonelutable
Meets LAW vitrification throughput	Yes	Yes
Avoid Cs return to tank farm	No	Yes
Avoid adding elution chemicals to DSTs	No	Yes
Avoid project delay/support December 2021 target date for feed delivery to LAW Vitrification Facility	No	Yes
Meets Amended Consent Decree (CD) milestone for LAW Vitrification Facility hot commissioning (December 2023)	No	Yes
Frees DST space to support SST retrievals/avoid building new DSTs	No	Yes
Meets CD-1 total project cost range	No – $790M	Yes – $220M to $470M
Requires interim storage of spent media columns	No	Yes

3.2.2 NRC Evaluation of Removal to the Maximum Extent Practical

The NRC staff performed a risk-informed review of the information DOE provided in the Draft WIR Evaluation for VLAW, the IDF PA, as well as numerous other supporting documents. The appropriateness of an approach to achieve removal of key radionuclides to the maximum extent technically and economically practical is dependent on the projected impacts to the public from the key radionuclides. Removal goals or requirements should be risk-informed if public health and safety is to be protected and if taxpayer dollars are to be used efficiently. The NRC staff considered the following questions when performing the review:

- How was the technology selected?
- Was the technology selection complete?
- Was the limit of the technology achieved?
- What operational or system changes may facilitate additional key radionuclide removal?
- Is additional waste removal necessary?

A possible approach to answering these questions would be to answer the last question first and, based on the outcome, evaluate the other questions. In a risk-based approach, this may in fact be the order in which the questions are evaluated. However, removal of key radionuclides to the maximum extent technically and economically practical is a concept analogous to NRC's ALARA standard and it is implemented using a risk-informed approach rather than a risk-based approach. The risk-informed approach uses risk information, considering uncertainties, to inform the overall decision-making process to make prudent and practical decisions, while erring on the side of protection of public safety.

3.2.2.1 NRC Evaluation of Waste Removal Processes

The NRC staff's examination of waste removal processes considered the final disposition of key radionuclides in the IDF. If the radioactivity is disposed in the same facility, then it may impact a member of the public even if the radioactivity is disposed in different wasteforms. The NRC staff note that ^{99}Tc and ^{129}I, which are volatile radionuclides, are key radionuclides for the groundwater pathway. The volatile radionuclides ^{3}H, ^{14}C, and ^{129}I are key radionuclides for the air pathway. In the Draft WIR Evaluation for VLAW, DOE stated that the LAW Vitrification Facility off-gas system is "designed to recycle and/or capture those volatile radionuclides". In Section 2.5.3 of the Draft WIR Evaluation for VLAW DOE discussed how the volatile components will be transported through the off-gas treatment system (DOE, 2020a). Liquid condensate from the SBS and WESP will be transferred to the EMF. The non-recycled portion of the treated off-gas is sent to the ETF for further treatment prior to disposal. The solids removed will be incorporated into a cementitious wasteform and disposed at the IDF as SSW.

DOE stated the inventory of the 28 contaminants in each wasteform to be disposed at the IDF was uncertain (RPP-RPT-59958, 2018). DOE completed an iterative evaluation and assessment process to determine what primary treatment and secondary treatment processes were necessary. DOE proposed a range of inventory cases that might occur. These cases were labeled Case 7, 8b, 9, 10a, and 10b. These hypothetical inventory cases were meant to help inform whether it may be desirable to increase the amounts of key radionuclides in higher performance (i.e., lower release rate) wasteforms. The NRC staff reviewed the inventory cases and found them to appropriately represent different treatment options associated with recycling of off-gas.

At the time this TER was prepared, the single-pass retention rates of volatile species in vitrified glass and the effectiveness of recycling are uncertain. The NRC staff asked for additional information about the retention of volatile species in its RAI (NRC, 2020a). As discussed in Section 4.6.2.3, DOE has not demonstrated the higher retention values asserted in response to the RAI (DOE, 2021c). Depending on the processes and separations used, a moderate to significant amount of ^{129}I and ^{99}Tc may end up in SSW or ETF-LSW. DOE stated that the SSW is a newly generated waste stream and will include a wide variety of waste (e.g., HEPA filters, granular activated carbon (GAC), silver mordenite) that will be generated during and after the LAW has been vitrified. DOE considered SSW to be outside the scope of the WIR evaluation, but DOE included the SSW as part of IDF PA. The SSW will be classified by DOE to ensure it meets the Waste Acceptance Criteria (WAC) for the IDF. DOE stated that Criterion A only applies to the removal and pretreatment of waste from the tanks prior to developing wasteforms (e.g., glass, cements).

DOE has proposed to use in-tank settling, separation of the supernate, filtration and cesium removal to achieve removal of key radionuclides in the DFLAW approach. The DOE focused on the supernate in the tanks because much of the long-lived radioactivity is associated with sludge. Based on technical reports, about 20-25 percent of the ^{238}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, and ^{244}Cm is not in the sludge. DOE stated that filtering will remove insoluble radionuclides. DOE expects that no visibly detectable solids will be present following filtration. The radionuclides present in the resulting liquid will be those radionuclides that are partially or completely soluble, including ^{137}Cs, ^{99}Tc, ^{129}I, and possibly ^{90}Sr. The NRC staff agrees that the settling, separation, and filtration processes will remove most of the insoluble radionuclides. DOE would then use ion-exchange columns to remove most of the ^{137}Cs.

The DFLAW approach is designed to target a component of the waste in the underground storage tanks that has less insoluble, long-lived radionuclides such as transuranics (TRU). DOE stated that the DFLAW pretreatment processes will remove over 99 percent of the ^{137}Cs and other radionuclides. The ion-exchange columns are likely to remove over 99 percent of the ^{137}Cs. Removal of ^{137}Cs is important because, if present in original concentrations in the waste, it would likely lead to doses that could exceed the performance objectives for the inadvertent intruder unless an extended institutional control period was used.

The NRC staff asked for additional information on the removal of key radionuclides (NRC, 2020a). In response to the RAI, DOE provided the removal efficiency for key radionuclides using the DFLAW approach. This is the removal of radionuclides prior to producing wasteforms. DOE also provided the fraction of the BBI that would not be disposed in the IDF (NRC staff calculated percent disposed in the IDF). Table 3-4 is the removal efficiency for key radionuclides. Although the total radioactivity removed is high, the removal of select key radionuclides can be much lower. The removal of ^{137}Cs is expected to be nearly 100 percent and the removal of ^{241}Am is expected to be 99.9 percent. On the other hand, the removal of ^{14}C is only 39.7 percent and the removal of ^{129}I is 45.0 percent. The removal of radionuclides that tend to impact inadvertent intruders is generally high, whereas the removal of radionuclides that impact long-term doses either in the groundwater pathway or air pathway tend to be much lower.

Table 3-4 Removal of Key Radionuclides

Radionuclide	Removed DFLAW (%)	IDF (%)	IDF SSW (%)	Radionuclide	Removed DFLAW (%)	IDF (%)	IDF SSW (%)
^{3}H	67.7	0.4	100.0	^{234}U	98.7	9.7	19.5
^{14}C	39.7	*	100.0	^{235}U	98.7	9.9	85.8
^{60}Co	96.7	9.5	8.6	^{238}U	98.7	9.8	21.4
^{59}Ni	90.5	13.0	< 0.1	^{237}Np	95.8	21.6	< 0.1
^{63}Ni	92.9	12.5	< 0.1	^{238}Pu	99.4	14.4	< 0.1
^{79}Se	NP#	100.0	< 0.1	^{239}Pu	99.2	13.0	0.2
^{90}Sr	98.7	10.3	1.1	^{240}Pu	99.3	15.7	1.0
^{99}Tc	55.2	99.9	< 0.1	^{241}Pu	99.7	14.7	< 0.1
^{126}Sn	76.2	100.0	< 0.1	^{242}Pu	85.0	17.9	< 0.1
^{129}I*	45.0	98.0	42.3	^{241}Am	99.9	9.6	< 0.1
^{137}Cs	100	0.8	37.4	^{243}Am	99.9	11.8	< 0.1
^{226}Ra	NP	99.9	< 0.1	^{242}Cm	97.0	99.9	< 0.1
^{229}Th	39.1	4.1	7.9	^{243}Cm	98.5	100.0	< 0.1
^{233}U	98.4	6.6	6.9	^{245}Cm	98.6	100.0	< 0.1

* DOE identified a significant amount of ^{129}I associated with laboratory waste that is not tracked in the BBI system.

NP=not present

DOE's emphasis on total activity removal is an imperfect measure of risk reduction when different receptors and pathways are considered. As shown in Table 3-4, the amount of certain key radionuclides removed using the DFLAW approach can be relatively low. Some radionuclides are retained in high percentages in the secondary wastes that will be disposed in the IDF. The fourth and last columns of Table 3.4 are the percentage of activity that is in SSW compared to vitrified wasteforms. The secondary wastes have only about one seventh of the volume of the total vitrified waste that will be disposed in the IDF. If intruder protection was the only performance objective, then the DFLAW approach could be concluded to be highly effective. However, impacts to water and air must also be considered. In the IDF PA, DOE considered the impacts to water and air for the radionuclides that are not highly removed by DFLAW. The estimated doses (discussed in Section 4.15.2) were calculated to be very small during the compliance period and about an order of magnitude less than the performance objective at later times. However, there are uncertainties associated with the estimated doses. The NRC staff expects that the long-term doses may be higher than calculated by DOE, but the increased doses are very unlikely to occur during the compliance period. Given the costs associated with changing the VLAW process, it is not likely to be economically practical to further remove key radionuclides. The NRC staff believes there is little to no merit in reducing a 0.01 mSv (1 mrem) dose.

The radionuclide ^{90}Sr can be a key contributor to chronic doses for the inadvertent intruder. The NRC staff asked for additional information on the amount of ^{90}Sr in the tanks that would be soluble (NRC, 2020a). NRC noted that only about 0.3 to 0.5% of the ^{90}Sr was included in the inventory for the IDF PA. In response to the NRC, DOE stated that of the total inventory of waste stored in the Hanford single-shell and double-shell tanks, it was estimated that 4 percent of the ^{90}Sr was soluble (DOE, 2021c). DOE indicated that more recent laboratory data showed that most of the soluble ^{90}Sr will be removed by the crystalline silicotitanate (CST) within the TSCR unit (PNNL-28945, 2019). In addition, DOE stated it intends to characterize the amount of ^{90}Sr in the LAW feed after startup by sampling and analyzing every batch received in the LAW Vitrification Facility concentrate receipt vessels and tracking it through the vitrification process. Through the combination of new research and the plan for characterization and verification, DOE has addressed the amount of soluble ^{90}Sr that will be in the VLAW.

3.2.2.2 NRC Evaluation of Alternative Treatment Technologies

The NRC staff's scope of evaluating alternative treatment technologies was limited to those treatments that are compatible with the decision to vitrify the LAW. Given the cumulative investment made to implement the current approach, the NRC staff did not believe it is likely or practical to implement a new technology. There is a large amount of project inertia associated with the complex, integrated decisions that must be made at the Hanford Site. Changes to the current approach are expensive and time consuming and cannot be examined in isolation without considering how they fit into the broader decision-making context. The scope of the NRC staff's review of the Draft WIR Evaluation for VLAW did not include a review of DOE's decision to pursue a glass wasteform. There are a range of wasteforms that could be used to allow radioactive waste to be disposed as LLW; DOE has chosen vitrified glass in this instance. The NRC staff's review did consider the various alternatives within the overarching decision to develop a glass wasteform, as well as changes within the DFLAW approach.

DOE performed studies on the potential removal of technetium. Elutable IX with SuperLig® 63953 was tested from 1996 to 2003 for deployment in the original Hanford waste treatment plant (RPP-PLAN-54676, 2013). The removal of technetium was technically practical but was deemed to be unnecessary if the technetium was to be placed in a glass wasteform. DOE also considered if iodine could be removed from tank waste by exploring available technology to remove ^{129}I. The concentration of ^{129}I 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. The removal of ^{129}I from tank waste was not considered to be technically practical because technology has not been demonstrated for the relatively low concentrations in the Hanford tank waste (WHC-SD-WM-TI-699, 1996).

The high temperatures of the vitrification process selectively drive certain volatile species (^{99}Tc, ^{129}I) to the off-gas system. These volatile species can be captured in the off-gas system and can be removed to secondary wastes or can be recycled back to the melter to increase retention in the glass. The NRC staff understands that this occurs after waste has been removed from the tanks, but the waste treatment process creates opportunity to selectively remove risk drivers. For instance, silver mordenite can be used to capture ^{129}I which then could be disposed as HLW, thereby decreasing the amount of a key radionuclide that contributes to long-term groundwater impacts at Hanford. As shown in Table 3-4, DOE expects 98 percent of the ^{129}I in the underground tanks will be disposed in the near surface in the IDF. This result shows why NRC considers secondary wastes to be within the scope of the Draft WIR Evaluation for VLAW.

DOE has completed significant research on glass wasteforms including using recycle to increase retention of volatile species. The NRC staff does not have access to the total amount invested in this research or what it would cost to qualify secondary wastes with high concentrations of key radionuclides as HLW rather than WIR managed as LLW. DOE has implemented measures to increase the retention of volatile species in glass (e.g., use of reductants, a cold cap, recycling). However, vitrification very effectively removes the two drivers of groundwater pathways doses from the waste due to volatilization of the species during the vitrification process, and off-gas systems exist to capture those radionuclides. Therefore, separation of the volatile species and disposition as HLW may decrease the future risks from the IDF. It is recommended that DOE formally assess these options from a cost-benefit and risk reduction perspective because there may be unforeseen uncertainties associated with using recycle during production-scale vitrification of diverse wastes that could result in challenges to the treatment approach (Recommendation #1). Based on currently estimated future doses, this recommendation does not need to be implemented, however, there may be unforeseen uncertainties associated with using recycle during vitrification of diverse wastes at the production scale that result in challenges to the treatment approach. In addition, there is additional research on glass performance that DOE has yet to complete. If this research indicates that the potential future doses could be higher than currently estimated, then an alternate approach for managing the volatile radionuclides could be preferable from a cost-benefit or risk reduction perspective.

The Government Accountability Office (GAO) examined Hanford's pretreatment technology (GAO-20-363, 2020). GAO evaluated the cost of pretreatment efforts from fiscal year 2013 through fiscal year 2018, the status of the technical challenges facing the pretreatment facility, and the steps DOE was taking to start treating waste by 2023. GAO made two recommendations, including that DOE ensure that its analysis of alternatives for pretreatment of HLW include a mission need statement and a life-cycle cost estimate for the baseline alternative. DOE began work on a strategy to bypass the originally planned pretreatment facility in 2013 (DOE, 2013). In 2017, an external review group recommended the current direct-feed approach with tank side cesium removal (RPP-RPT-60405, 2017).

Construction of new facilities and modification of existing facilities has occurred under the DFLAW approach. For example, the EMF will be used by DOE to manage the contaminated liquid generated through the processing of LAW, which was originally intended to be part of the pretreatment facility. From 2014 to 2018, DOE spent resources on designing a separate Low-Activity Waste Pretreatment System and facility which was to separate out the less radioactive portion of waste from the tanks in preparation for direct-feed to the LAW facility. Work on this facility was suspended in 2017. After suspending the LAW Pretreatment System work, DOE began work on a demonstration of the tank side cesium removal technology in 2018. DOE could construct new facilities within the scope of DFLAW.

Table 3-5 provides a summary of the removal technology DOE will rely on to achieve removal of key radionuclides. The removal technologies will be highly effective for some key radionuclides and not as effective for others. The NRC staff believes that DOE considered alternatives in arriving at the current DFLAW approach and the technologies considered were reasonable. DOE also continues to evaluate changes to the DFLAW treatment processes as new information is developed. If projected groundwater impacts were to increase significantly and be expected to occur within the DOE compliance period, then examination of alternatives to the current approach may be warranted. Based on the current assessment results, the DFLAW approach is sufficient to achieve removal of key radionuclides to the extent technically and economically practical.

Table 3-5 Summary of Removal Technologies for Key Radionuclides

Radionuclide	Soluble or Insoluble/Form	Removed By
^{3}H, ^{14}C	Soluble	Partition to SSW during the melting process; disposed of at IDF
^{60}Co	Soluble	Short half-life (5 years)
^{59}Ni	Soluble	Present in very low concentrations in the vitrified LAW; insignificant contributor to dose after IDF closure
^{63}Ni	Soluble	Short half-life (100 years)
^{90}Sr	Insoluble/Soluble	Insoluble part remains in sludge or is removed by filtering; Soluble fraction removed by passing the waste through three ion-exchange columns.
^{99}Tc	Soluble/Volatile	Recycled into vitrified waste; or sent to SSW
^{129}I	Soluble/Volatile	Recycled into vitrified waste; adsorbed by activated carbon beds or sent to SSW
^{137}Cs	Soluble	Removed by passing the waste through three ion-exchange columns
^{228}Rn	Gas	Flux controlled by natural engineered system
^{229Th}, ^{232}Th	Insoluble	Settling; Decanting; Remaining Insoluble fraction is filtered
^{234}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu	Depends on chemical form	Insoluble fraction filtered; soluble fraction removed by passing the waste through three ion-exchange columns
^{241}Am, ^{243}Am	Insoluble	Settling; Decanting; Remaining Insoluble fraction is filtered
^{242}Cm, ^{243}Cm, ^{244}Cm	Insoluble	Settling; Decanting; Remaining Insoluble fraction is filtered

3.3 NRC Conclusions for Criterion A

The NRC staff evaluated DOE's demonstration that wastes have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical, by evaluating the identification and the removal of key radionuclides. The latter aspect included assessing waste removal processes, termination of waste removal, and DOE's consideration of alternative treatment technologies. The NRC staff performed a risk-informed review of the information provided. The acceptability of removal of key radionuclides to the maximum extent technically and economically practical is conditional on the technical acceptability of the PA and other supporting analyses.

The NRC staff's conclusions related to Criterion A are as follows:

- DOE properly identified key radionuclides.
- DOE demonstrated the removal of key radionuclides to the maximum extent technically and economically practical for those radionuclides most likely to impact the inadvertent intruder.
- The removal of ^{137}Cs will likely be highly effective.
- DOE demonstrated the removal of key radionuclides with respect to groundwater pathway doses because those impacts are expected to occur after the compliance period and are anticipated to be low with respect to the performance objective.
- A minimal amount of ^{99}Tc and ^{129}I is expected to be incorporated into waste that will not be disposed in the IDF.

The following assumptions apply to the NRC staff's conclusions:

- The PA results for the estimated doses from the air and groundwater pathways for an offsite member of the public do not change significantly in the future.
- The PA results for the estimated doses to an inadvertent intruder do not change significantly in the future.

The NRC staff has the following recommendation associated with Criterion A:

- From a cost-benefit and risk reduction perspective, DOE should formally assess the separation of volatile species and disposition as HLW. Because vitrification very effectively removes the two drivers of groundwater pathways doses from the waste, and off-gas systems exist to capture those radionuclides, disposition as HLW may decrease the future risks from the IDF. Based on currently estimated future doses, this recommendation does not need to be implemented, however, there may be unforeseen uncertainties associated with using recycle during vitrification of diverse wastes at the production scale that result in challenges to the treatment approach. (Recommendation #1)

4 CRITERION B – Compliance with 10 CFR Part 61 Performance Objectives

This section summarizes the NRC staff's review of the information DOE submitted with respect to Criterion B of DOE Manual 435.1-1, which is demonstrating compliance with the 10 CFR Part 61, Subpart C performance objectives for the disposal of DFLAW at the IDF.

The technical areas evaluated by the staff are described in the following subsections. These subsections are structured to summarize DOE's information submitted followed by the NRC staff's evaluation of DOE's information. Each subsection concludes with a summary of the NRC staff's review that identifies whether the NRC staff found DOE's approach to be reasonable, identifies sources of uncertainty and/or risk drivers, and provides the NRC staff's recommendations. This structure was followed to allow the reader to better understand the context of the NRC staff's conclusions and recommendations.

The NRC staff performed a risk-informed review using the risk insights developed by DOE and independent analysis to complete the evaluation in this TER. DOE developed its risk insights primarily from computer simulations and analyses. Different technical issues that the NRC staff identifies in this TER may not be equivalent in terms of risk-significance. Staff examines the cumulative impacts of issues and uncertainties. In the instances where staff determines that the cumulative impact of issues and uncertainties is not risk-significant and there is reasonable assurance that the requirements will be met, the staff identifies these issues as "recommendations".

The recommendations provided in each subsection are collated in Table 6-1. Table 6-1 identifies three categories of recommendations: (1) "DFLAW" means applicable to the Direct-Feed LAW that is evaluated in the Draft WIR Evaluation for VLAW; (2) "VLAW" means applicable to the remainder of the vitrified LAW or secondary wastes; and (3) "General" means if completed can improve the technical basis for DFLAW, VLAW, and/or future waste evaluations and is considered a best practice for performing waste evaluations. The recommendations are numbered in each subsection (e.g., Recommendation #2) and are indexed to Table 6-1. The recommendations from the WMA C TER that are applicable to VLAW are included in Appendix A. The NRC staff's overall conclusions for Criterion B are provided in Section 4.16.

The staff identifies many technical recommendations in Section 4. Most of the recommendations are not applicable to the Draft WIR Evaluation for disposal of VLAW at IDF because of the low projected risks from vitrified wastes. In addition, DFLAW represents only about 10% of the vitrified waste that will be produced. These recommendations are relevant to future waste evaluations and determinations especially for sites with higher projected risks. If the estimated performance of wasteforms or the disposal facility were to change materially, then the recommendations could be applicable to VLAW. The applicability of a recommendation is dependent on the risk-significance of the topic for the specific waste stream and site conditions.

4.1 Assessment Context

A complex technical evaluation called a PA is used to evaluate the potential future risks to a member of the public from the disposal of vitrified (and other) wasteforms. The PA is a computer model, or models, as well as a large collection of documents, files, data, and supporting information. The IDF PA is documented in an extensive report (RPP-RPT-59958, 2018) as well as additional references.

In this section, the staff evaluates the context of the IDF PA, a term defined as "assessment context". The assessment context is a description of the problem and systems. The assessment context includes the purpose of the PA, the regulatory framework, the overall assessment philosophy or strategy, the endpoints and timeframes for the assessment, and a description of the waste characteristics and disposal system characteristics. The waste characteristics and disposal system characteristics were summarized in Section 2. To develop the assessment context of a PA, the following questions are considered:

What is being assessed?

In the performance assessment, DOE evaluated the disposal of all low-activity wastes anticipated to be generated. In the draft WIR evaluation, DOE assessed the potential impacts from disposal of approximately 13,500 canisters of DFLAW in the IDF at the Hanford. The IDF is a large, near-surface disposal facility with a liner and sump system as well as an engineered multi-layer cover. DOE documented their evaluation in the Draft WIR Eval for VLAW (DOE, 2020a).

Why is it being assessed?

The disposal of the waste in the IDF is being assessed to ensure that public health and safety will be protected in accordance with the requirements provided in DOE Order 435.1 (DOE, 2021a).

What is the scope of the assessment?

DOE assessed potential impacts for offsite members of the public after closure of the disposal facility for periods of time of up to 10,000 years and longer. DOE also evaluated potential impacts to members of the public who may inadvertently use the land overlying the disposed wastes sometime in the future. The scope of the evaluation was determined by a model development process considering features, events, and processes. DOE included uncertainties in the assessment.

4.1.1 DOE's Assessment Context

DOE developed the scope of the IDF PA to provide information demonstrating the protection of long-term human health and the environment. The PA modeling and documentation provided the assessments needed to approve operation (e.g., disposal of different wasteforms) and eventual closure of the IDF. The purpose of the PA was to demonstrate that the relevant regulatory criteria would be met if certain wastes were disposed in the IDF. The regulatory framework, criteria, endpoints, and timeframes are discussed, followed by a summary of DOE's framework and strategy to complete the assessment.

The regulatory context for waste disposal at the IDF, including requirements for the protection of human health and the environment, is regulated by multiple agencies (DOE, Ecology (State of Washington Department of Ecology), and EPA). The laws and regulations which govern closure processes for the disposal facility include the National Environmental Policy Act (NEPA), *Hanford Federal Facility Agreement and Consent Order* (HFFACO) (Ecology et al., 1989), RCRA/*Hazardous Waste Management Act* (HWMA) (*Revised Code of Washington* [RCW] 70.105, "Hazardous Waste Management"), *Atomic Energy Act of 1954* (AEA), and CERCLA.

In concert, these laws and regulations provide the guidelines for the disposal and closure processes. NEPA provides the decision-making structure for Federal agencies. The HFFACO describes closure activities, which are driven by (1) the requirements of the AEA, as amended, regulating the radioactive portion of mixed waste, and (2) RCRA/HWMA as implemented through *Washington Administrative Code* 173-303, "Dangerous Waste Regulations," regulating the non-radioactive dangerous portion of mixed waste. It should be noted that the various laws and regulations for closure create redundant and sometimes conflicting administrative requirements. The HFFACO, in part, was established to address these issues and to also identify the need for analyses that will be approved by Ecology and DOE pursuant to their authorities under RCRA and the AEA. The HFFACO will ensure the actions taken for waste disposal are protective of human health for all contaminants of concern (radiological and non-radiological).

The performance objectives and standards that the results of the PA calculations were evaluated against are found in DOE Order 435.1. The point of assessment (receptor location) and timing assumptions DOE used followed requirements from DOE Order 435.1 and HFFACO. DOE's expectation is that the Hanford Site will undergo cleanup activities for the foreseeable future, however, institutional control and societal memory are assumed to last for only 100 years after closure. Inadvertent human intrusion is assumed to occur after the active institutional control period though DOE may still be present at the site actively managing closure of different facilities. The intruder protection objective is applied consistent with DOE Order 435.1 principles and guidance. The point of assessment for all-pathways (i.e., combined doses for the groundwater and air pathways) and groundwater protection analyses is at the point of maximum concentrations that are 100 m (328 ft) or farther from the downgradient fence line of the IDF, per DOE Guide 435.1-1 (DOE, 1999a). A point of exposure 100 m (328 ft) downgradient of the facility fence line (i.e., at the wellhead of a pumping well) after closure of the facility was used for the all-pathways performance objective. Prior to closure of the facility and during operations the nearest receptor is located 20 km (12 mi) from the facility. Peak concentrations in groundwater were used in the all-pathways analyses. Performance objectives and the standards for all-pathways, atmospheric, radon flux, inadvertent intruder, and groundwater protection analyses are shown in Table 4-1.

The compliance time specified in DOE Order 435.1 for a PA is 1,000 years after closure. In the IDF PA, DOE also included also included a 10,000-year analysis period based on the recommendations in NRC's guidance NUREG-1854 (NRC, 2007) and to provide information to decisionmakers about potential long-term doses. The 1,000 to 10,000-year timeframe is called the post-compliance period and is used for sensitivity analyses. DOE also performed analyses out to 500,000 years to estimate potential peak doses. DOE Manual 435.1-1 and DOE Guide 435.1-1 provide direction that a sensitivity-uncertainty analysis timeframe should include the calculation of the maximum dose regardless of the time at which the maximum occurs to provide a means of increasing confidence in the outcome of the modeling and to increase understanding of the models.

Table 4-1 Exposure Scenarios, Performance Objectives and Measures, and Points of Assessment for the Integrated Disposal Facility Performance Assessment

Exposure Scenario	Performance Objectives and Measures	Point of Assessment	
		Operational and Active Institutional Control Periods	Post-Institutional Control Period
All-pathways	0.25 mSv/yr (25 mrem/yr)	20,000 m (65,600 ft) – nearest offsite receptor in direction of prevailing wind	100 m (328 ft)
Air pathway	0.10 mSv/yr (10 mrem/yr)	20,000 m (65,600 ft) – nearest offsite receptor in direction of prevailing wind	100 m (328 ft)
Radon	20 pCi/m^2/s*	Flux rate at facility surface	Facility surface
	0.5 pCi/L*	Facility boundary	100 m (328 ft)
Water resources	Washington State Department of Ecology requirements on concentrations of radionuclides and hazardous chemicals	At the source and 100 m (328 ft)	
Intruder	1 mSv/yr (100 mrem/yr) Chronic	Not applicable	Facility
	5 mSv/yr (500 mrem/yr) Acute	Not applicable	Facility

* The units provided in the standard are a mix of English and System International (SI)

To meet the DOE Order 435.1 requirements, DOE examined various exposure scenarios (e.g., resident, farmer) to calculate the total effective dose equivalent for comparison to the performance objective of 0.25 mSv (25 mrem). The all-pathways dose combines the dose from both the groundwater pathway and the atmospheric pathway excluding the dose from radon and its progeny in air. In the IDF PA, DOE assumed the individual who receives a dose from the groundwater pathway is a Representative Person (DOE, 2016) who resides near the IDF and draws contaminated water from a well downgradient. DOE assumed that water was used for drinking, irrigation of crops, and to water livestock. For the atmospheric transport pathway, air immersion, dust inhalation, and external exposure were the dose pathways for the receptor residing 100 m (328 ft) downgradient of the facility fence line. Figure 4-1 provides an overview of the conceptual model used to assess the disposal of VLAW and other wastes in the IDF. Intruder receptors are assumed to be located above the facility after closure, whereas offsite receptors (not shown on the figure) are assumed to be located downgradient or downwind.

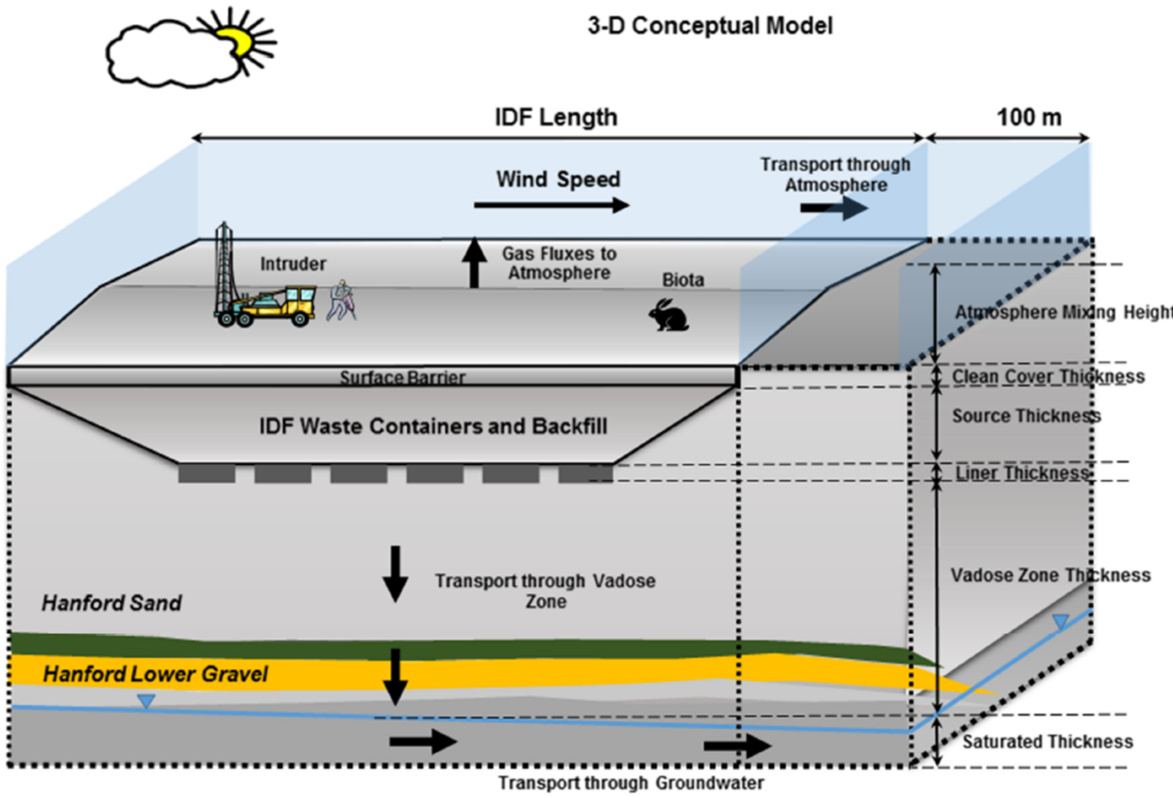


Figure 4-1 Conceptual Model for the IDF PA Showing Key Features
[Figure 2-11 of the IDF PA (RPP-RPT-59958, 2018)]

DOE used numerical models to simulate the potential impacts to future receptors based on the conceptual model shown in Figure 4-1. DOE's strategy was to use expert judgment to screen potential FEPs for potential inclusion in the modeling effort. DOE used available site historical and characterization information to produce the conceptual model. DOE used multiple computational tools to develop, test, evaluate, and summarize the results from numerical representations of the conceptual model. The modeling approach is a component of the assessment context.

DOE's modeling approach in the IDF PA included models for source-term release, contaminant fate and transport along the groundwater pathway, contaminant fate and transport along the air pathway, and exposure and dose analysis. Potential impacts to inadvertent intruders were also evaluated. DOE provided a schematic representation of their overall modeling approach in the IDF PA as shown in Figure 4-2 (RPP-RPT-59958, 2018). The state of knowledge, primarily developed from multiple decades of site operations and scientific studies, was used to develop physical and chemical representations of the systems including their associated uncertainties. Process modeling was performed with tools such as Subsurface Transport Over Multiple Phases (STOMP©, copyrighted by Battelle Memorial Institute, 1996) to simulate flow and transport in the unsaturated and saturated zones and Geochemist's Workbench (PNNL-12030, 2000). DOE used the deterministic models to evaluate the relevant features and processes of the individual components that lead to the release of contaminants from the wasteforms and the subsequent transport of the contaminants by the groundwater pathway to the potential

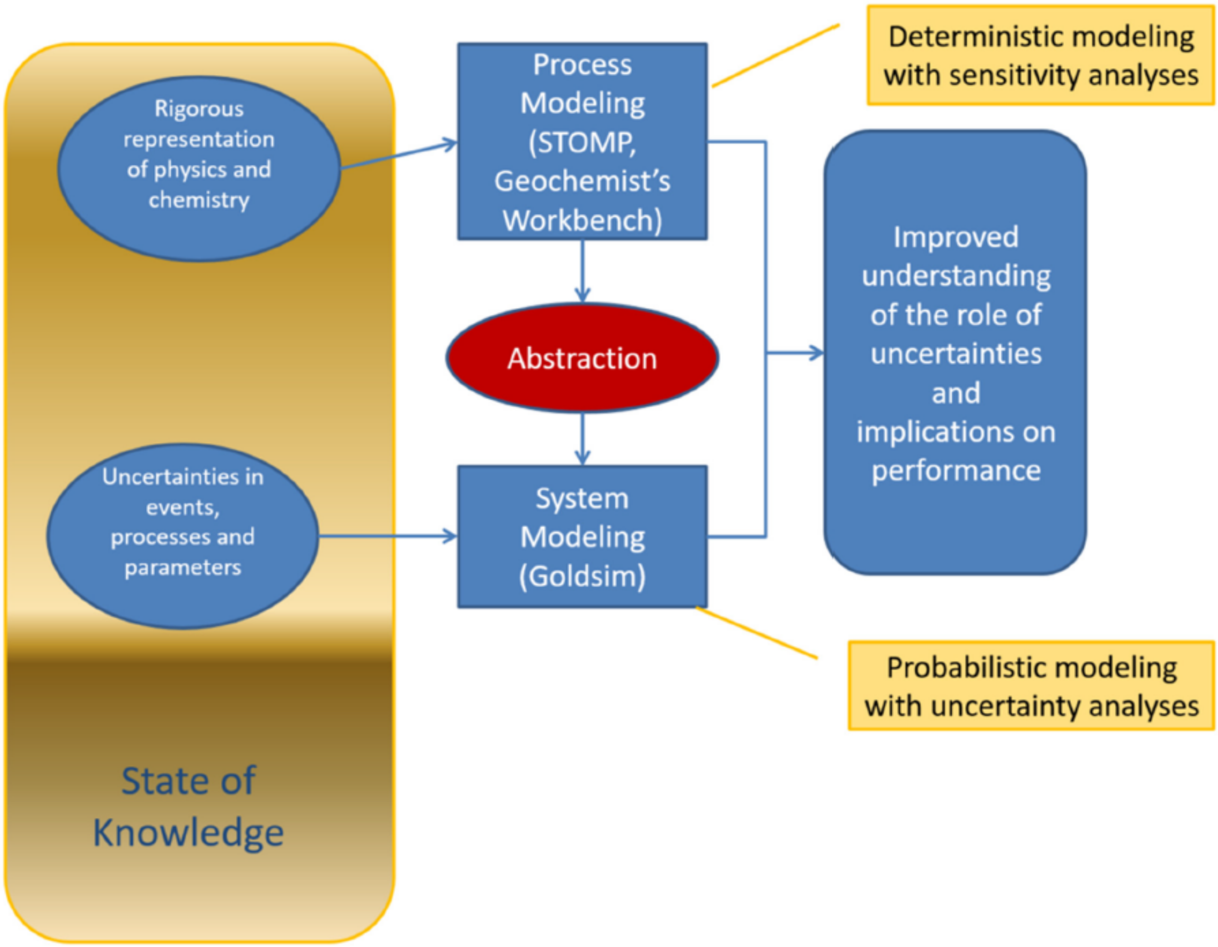


Figure 4-2 Modeling Approach Used in the IDF PA
[Figure 1-11 of the IDF PA (RPP-RPT-59958, 2018)]

receptors. Process models typically include a detailed two- or three-dimensional representation of the relevant processes that can affect the fate and transport of contaminants in and through the engineered and natural barriers of the system. In some cases, the results of process modeling were simplified through a process called abstraction to combine the process models in a system model. Abstractions are simplifications of more detailed modeling (such as a lookup table) that preserves the essential detail of the process modeling for more efficient implementation in a system model.

System modeling was performed with the GoldSim (GoldSim© simulation software is copyrighted by GoldSim Technology Group LLC of Issaquah, Washington) software package. DOE used both deterministic modeling with sensitivity analyses and probabilistic modeling with uncertainty analyses to develop insights of potential future system performance. The integrated system model was used to evaluate the uncertainty in the groundwater pathway dose, the atmospheric pathway dose, radon flux, and the effects of an inadvertent intrusion into the disposal facility. The deterministic process models were used to evaluate the expected fate and transport of contaminants planned for disposal in the disposal facility.

The deterministic models took a long time to execute (e.g., in some cases, days for a single simulation) and DOE elected to run the process models in an uncoupled manner where the results from one calculation were manually transferred to the next calculation. The results from a deterministic case using the most likely or expected or central tendency parameter values represented a reasonably conservative case used for compliance demonstration. DOE termed this result the "base case." The system model was comprised of different submodels. The submodels DOE developed included representations for the source-term, groundwater pathway fate and transport, and biosphere transport and exposure. Source-term models included hydrological flow in the near-field in and around the engineered features, release from the wasteforms, and transport through the engineered barriers to the surrounding geology. The system model was used to calculate the fate and transport of contaminants and dose to the receptor. Uncertainty in the models and parameters that affect the release and transport of the contaminant from the disposal facility was included. DOE stated the result of this process was an improved understanding of the significance of uncertainties.

DOE used the integrated system model for several different purposes: 1) to evaluate the atmospheric pathway and the radon flux performance objectives, as well as the inadvertent intruder performance measure, 2) to evaluate the sensitivity of the groundwater performance objective (groundwater dose to a receptor) to alternative conceptual models, and 3) to quantify the impact of uncertain parameter values on the potential groundwater dose to a receptor. Because of the computational efficiency of the integrated system model, all relevant radionuclide and hazardous chemical constituents were included in the inventory. The results of the system model were used to confirm the observation made in the previous PAs, as well as the environmental impact statement, that the key radionuclides relevant to the groundwater pathway performance objective and groundwater resource evaluations were ^{99}Tc and ^{129}I.

4.1.2 NRC Evaluation of Assessment Context

The NRC staff reviewed various documents that described components of the assessment context. The NRC staff reviewed the assessment purpose, regulatory framework, assessment philosophy, dose modeling methodology, compliance boundaries, assessment end points, and assessment timeframes to determine if DOE adequately addressed the questions that need to be answered to give context to the PA: what is being assessed, why is it being assessed, and what is the scope of the assessment?

DOE adequately described the purpose of the assessment, which was to demonstrate that disposal of certain types of wastes at the IDF would meet the relevant regulatory criteria (e.g., protection of public health). DOE provided the regulatory criteria that were used and described the complex regulatory framework and requirements that apply. DOE used a combination of deterministic and probabilistic modeling in the PA to estimate potential dose impacts to members of the public. Compliance points differed depending on if institutional controls were in effect or had lapsed (assumed to occur at 100 years after closure).

The scope of the assessment was developed and described in the PA (RPP-RPT-59958, 2018). Based on historical operations, scientific studies, and other sources of information, DOE developed the representations of the systems as well as the uncertainties in FEPs as shown in Figure 4-2. DOE described the types and methods of model abstraction, as well as data and computational needs and the strategy for the PA. A combination of process models, abstractions, and system modeling was used for the assessment. The modeling was thoroughly described and well-documented. The dose modeling methodology that DOE utilized was described in the IDF PA. The NRC staff determined that the information DOE provided was complete and consistent.

The compliance time specified in DOE Order 435.1 for a PA evaluation is 1,000 years after closure. In the IDF PA, DOE included a 10,000-year analysis period based on the recommendations in NRC's guidance NUREG-1854 (NRC, 2007) and to provide information to decisionmakers about potential long-term doses. The 1,000 to 10,000-year timeframe was termed the post-compliance period and was used for sensitivity analyses. The NRC staff recommended for WMA C that for future WIR evaluations and assessments, DOE follow guidance within DOE Manual 435.1-1 and DOE Guide 435.1-1 on length of sensitivity-uncertainty calculations (i.e., model runs should include the maximum or peak dose regardless of the time at which the peak occurs). DOE performed additional analyses out to 500,000 years to estimate peak dose, however, those analyses were not generally used for sensitivity analyses. In the case of disposal of waste at IDF, the intruder doses are expected to be more limiting and those are largest in magnitude immediately after the institutional control period ends.

4.1.3 NRC Conclusions on Assessment Context

The NRC staff reviewed the assessment context and has the following conclusion:

- The NRC staff reviewed various documents that described components of the assessment context and determined that DOE adequately developed and described the context of the IDF PA.

The NRC staff has no new recommendations for this topic.

4.2 Scenarios and Conceptual Models

Scenarios are sets of FEPs that are a potential representation of the disposal system for which the expected likelihood of occurrence of the scenario can vary. The term "scenario" represents a description of a potential specific evolution of the disposal system from a given initial state. A scenario describes one possible future of the disposal system, corresponding to a combination of events and processes together with their characteristics and their chronological sequence. Characteristics, such as who the assumed receptor is, can change as well as assumed processes and events. Changes to processes and events can result in an alteration to the chronological sequence. The site conceptual model is the collection of features, characteristics and processes used to represent the disposal system and site. Scenarios are used to represent potential performance of the disposal system under expected and plausible conditions. DOE also analyzed unexpected and potentially hypothetical conditions. Scenarios are generally used to represent different receptors, disruptive events, and alternative evolutions of the disposal system resulting from uncertainty. The results of simulating the performance for different scenarios can be used to understand the range of potential future performance, identify key barriers, and identify important uncertainties. The following sections provide DOE's approach to scenario development, the resultant scenarios that were developed, conceptual models of the site and disposal system and the NRC staff's review of the information.

4.2.1 DOE's Scenarios and Conceptual Models

DOE developed scenarios to evaluate different receptors and to examine different states of the disposal system. A deterministic base case or expected scenario was developed to examine performance under anticipated conditions. DOE's base case the most plausible scenario in which disruptive events are considered implausible and a static environment with steady-state climatic conditions is assumed. The base case results were then supplemented with a variety of other analyses to examine performance under unanticipated conditions.

To establish the scope of a PA (e.g., conceptual models and scenarios), it is common that FEPs are used. DOE stated the process of using FEPs consisted of four steps: 1) identification of a comprehensive list of FEPs, 2) screening the comprehensive list to a manageable number, 3) describing the relationships between the FEPs, and 4) arranging them into calculational cases, or scenarios, for the PA. DOE used a hybrid approach combining a bottom-up FEP compilation and screening process with a top-down safety function examination.

DOE identified FEP lists applicable to deep geologic repositories and near-surface disposal facilities. DOE indicated that the International Atomic Energy Agency (IAEA) published a comprehensive set of FEPs for near-surface disposal based on the results of the Improvement of Safety Assessment Methodologies (ISAM) coordinated research program (IAEA-ISAM-1, 2004). DOE included the FEP list in Appendix A of the PA (RPP-RPT-59958, 2018). FEP lists specific to the Hanford Site have been developed but DOE indicated that those lists contain FEPs at a level of detail that was too fine to implement effectively in a PA.

DOE described a safety function as a feature of the system that provides a specific function that is relevant to the performance (or safety) of the facility. The set of safety functions presents a high-level summary of the strategy by which the performance of the disposal system is assured. The set of safety functions for the IDF and surrounding site were identified and described. FEPs and safety functions were identified using expert judgment. FEPs were identified that could impact the ability of a safety function to provide an assurance of safety in the future. An example given by DOE was the limitation of water flow by an engineered cover. That safety function could be impacted by different FEPs. DOE indicated that since each of these FEPs influence the system in a similar manner (i.e., changes in water flow through the cover), sensitivity analyses that vary this safety function represent an aggregated view of the potential negative effects of a suite of FEPs.

DOE provided a table of safety functions (Table A-1 of RPP-RPP-59958). Some of the safety functions were administrative or regulatory in nature, such as assumptions about how long control of the site would be maintained. Most of the safety functions were grouped into three types: hydrological, chemical, and mechanical/structural. A total of 29 safety functions were identified. The ISAM FEPs list contained 141 FEPs that were mapped to the 29 safety functions. An example of the integration of FEPs and safety functions is found in Figure 4-3, which provides a FEP associated with hydrological response to climate change (RPP-RPT-59958, 2018). The FEP (1.3.07) was described as relevant to the infiltration rate safety function. The FEP was then linked to safety functions I4 (infiltration below the root zone), CAP (design of the engineered cover), VZ1 (water flow in the vadose zone) and SZ1 (advective flow in groundwater leading to dilution).

In DOE's approach, sensitivity and uncertainty analyses were tools used in concert with the FEPs screening and safety function identification and analyses. Sensitivity and uncertainty analyses were designed to be complementary and to help decision makers judge the capability of the disposal system to protect public health and safety. DOE performed a variety of sensitivity and uncertainty analyses to examine the impacts of a FEP or group of FEPs on a safety function or safety functions. In this manner, DOE included the FEPs in the scope of the analyses, but the FEPs may not have been included in the base case or central scenario. The uncertainty analyses used the system model to explore the global impact of uncertainties in select parameters or inputs. The sensitivity analyses were used by DOE to examine changes to safety functions and the resultant impacts on the output metrics. In Table 6-19 of the IDF PA, DOE described 31 sensitivity analysis cases that were performed (RPP-RPT-59958, 2018). DOE examined changes to the inputs used to represent the various engineered and natural barriers, as well as other important system components such as the waste inventory. The largest sensitivity was observed for cases associated with the waste inventory. Groundwater pathway doses were projected to meet the performance objectives for the 1,000-year timeframe. Peak groundwater pathway doses for select inventory cases were projected to significantly exceed the 1,000-year standard but not until more than 2,500 years after site closure.

4.2.2 NRC Evaluation of DOE's Scenarios and Conceptual Models

DOE's approach to development of scenarios in the IDF PA was nearly identical to that used in the WMA C PA, which the NRC staff previously reviewed (NRC, 2020b). The NRC staff's review was completed after the IDF PA was completed, and therefore, DOE could not consider NRC staff's comments. A summary of the review of DOE's scenarios for WMA C is:

- The approach results in a plausible central (base case) scenario.
- The hybrid approach does not provide a clear mechanism for safety functions or changes in safety functions to produce alternative conceptual models.
- Interdependencies and interrelationships of FEPs are not consistently identified using the hybrid approach.
- The resulting one-at-a-time sensitivity analyses have limited value to examine global uncertainties and interrelationships. These analyses do provide value to understanding the resilience of a safety function to a stress or reduction in performance.

In an uncertain system with multiple reasonable interpretations of data and information and uncertainty in the data and information, the results from evaluation of a particular scenario does not provide more meaningful information than from evaluation of other plausible scenarios. A scenario describes one possible future of the disposal system, corresponding to a combination of events and processes together with their characteristics and their chronological sequence. The conservatism, optimism, or realism of a result of a particular deterministic simulation is primarily a reflection of the conservatism assigned by the analysts in the selection of parameter values and the model representation. Probabilistic modeling results and the associated statistics provide a more complete understanding of the range of plausible outcomes.

Hydrological/hydrogeological response to climate changes	1.3.07

Definition: FEPs related to changes in the hydrological and hydrogeological regime, e.g. recharge, sediment load and seasonality, in response to climate change in a region.

Comment: *The hydrology and hydrogeology of a region is closely coupled to climate. Climate controls the amount of precipitation and evaporation, seasonal ice cover and thus the soil water balance, extent of soil saturation, surface runoff and groundwater recharge. Vegetation and human actions may modify these responses.*

Key Concepts, examples, and related FEPs

Change in groundwater recharge	*Change in regional precipitation/infiltration/evaporation*	*Change in surface runoff*
Change in sediment load	*Change in seasonal ice cover*	*Increase in groundwater velocity*
Change in soil water balance		*Creation of local ponds*

Application to IDF: *Relevant in evaluating the infiltration rate.*

Potentially deleterious FEP*: This FEP has the potential to affect the cover infiltration safety function. Effects of climate change on infiltration are included in the range of rates derived from the paleo record on precipitation. Potential anthropogenic effects are within the range of past climates. See Section 3.*

Figure 4-3 Example of the Integration of FEPs and Safety Functions
[Page A-39 of the IDF PA (RPP-RPT-59958, 2018)]

Different methods have been suggested to identify interdependencies and interrelationships of FEPs (e.g., interaction matrix). These methods are not particularly amenable to the safety function approach because groups of FEPs are mapped to safety functions, however, FEPs themselves can interact. A safety function is generally what happens to the system (e.g., water flow to the waste) rather than what causes changes to the system(s) (e.g., freeze-thaw effects on an engineered cover). The evaluation of many combinations of safety function performance is essentially a probabilistic evaluation without the probabilities. The challenge is how to evaluate FEPs that influence numerous safety functions as well as FEPs that influence other FEPs. Multiple FEPs can experience changes concurrently that alter the performance of the disposal system. A FEP for hydrological response to climate change (as shown in Figure 4-3) would be expected to influence not only the infiltration rate but also groundwater velocities. The change in groundwater velocities could be the result of increased recharge (captured in the DOE approach), or due to changes in the dam operation and river stages (not reflected in changes to recharge rates). Changes to infiltration rates could also impact degradation of engineered barriers, such as a cementitious wasteform, through increased leaching of the components of the wasteform matrix. The goal of the process is a dynamic system model with the appropriate interactions and dependencies to represent the real-world system.

The use of one-off sensitivity evaluations to examine the change in the performance of a safety function is useful to develop an understanding of the importance of the safety function with respect to all the other safety functions held constant. However, one-off sensitivity evaluations of changes to the performance of a safety function can have more limited value if there are many uncertainties associated with multiple safety functions. This is because the one-off result reflects only a small portion of the uncertainty-response space. In addition, though a relative change of +/- a factor of 2 may be applied to a parameter, without the probability of those values of the parameter occurring, a key piece of the risk triplet[13] is not available to interpret the result in a risk-informed evaluation. If there is not a common approximate relative probability to the values assigned for the different changes to parameters, then the magnitude of the resultant effect cannot be placed in the proper context. One-off sensitivity evaluations are a good tool to develop understanding of first-order effects but provide an incomplete picture of plausible sequences of events or conditions that can result in alternate scenarios.

DOE indicated that the FEP lists developed for Hanford provided information at too fine of a level of detail that was not amenable to implementation in a PA. The reports referenced included:

- 2002 LLW Repository PCSC – FEP Consideration [Phifer, 2011].
- FCRD-USED-2011-000297, “Features, Events and Processes for the Disposal of Low-Level Radioactive Waste – FY 2011 Status Report”).
- BHI-01573, “Groundwater/Vadose Zone Integration Project – The Application of Feature, Event, and Process Methodology at the Hanford Site”.
- WMP-22922, “Prototype Hanford Features, Events, and Processes (HFEP) Graphical User Interface”.
- WCH-520, “Performance Assessment for the Environmental Restoration Disposal Facility, Hanford Site, Washington”.

[13] The risk triplet is the combination of what can happen, how likely is it, and what are the consequences.

As of 2004, 179 primary Hanford Features, Events, and Processes (HFEP) were identified in WMP-22922. A total of 2,474 HFEPs were identified as lower tier. The NRC staff examined the lower tier HFEPs to get an understanding of the level of detail and why the more detailed FEPs could not be used in the PA development process. Below is a subset of the list of lower-tier FEPs that would be included in a traditional FEP screening process.

Anthropogenic climate change (numerous effects)
Increased recharge from agricultural operation
Dams
Dams and reservoirs built and drained
Introducing complexing agents
Differential thermal expansion of barriers
Biological degradation of grout formulations
Phase separation (glass)
Glass recrystallization
Biological degradation of wasteforms
Chemical alteration of backfill
Colloid formation
Improper function of drains
Barrier and cover designs to withstand earthquakes
Disposal of pyrophoric waste
Subsidence (cover and/or backfill)
I-Cs migration to canister surface
Hyperalkaline carrier plume forms
Reaction of infiltrating water with engineered barriers
Electrochemical cracking in waste
Episodic, pulse release
Alpha recoil enhances dissolution
Galvanic coupling
Complexation by organics
Microbial assisted corrosion of concrete
Nonuniform heat output, coupled temperature effects
Radiolysis
Desertification/sand dune formation
Episodic infiltration enhances colloid infiltration
Condensate formation

The intention of this list is to emphasize that it is not clear to the NRC staff that the safety function approach combined with barrier sensitivity analyses and parameter uncertainty analyses provides the risk information associated with the likelihood and consequences of these FEPs impacting the disposal system.

DOE's approach provided a clear description of what was included in the analysis, but not a clear basis for what was not included. Most of these FEPs are plausible, and therefore, either a probability-based or consequence-based screening argument would be needed to eliminate them. FEP screening can be challenging because the analyst must assess the combined consequences of the FEPs that are being screened. Examination of a FEP by itself may provide a relative change with all other FEPs held constant, but not the global change of all FEPs that may have been examined (and eliminated) in this manner. Improper screening can result in an overly simplified conceptual model that does not include important phenomena.

DOE's safety function approach attempts to confirm the subjectively generated conceptual model rather than mapping the FEPs and their influences to develop a conceptual model. Given the resources to iterate and enough observational information, either approach (i.e., traditional bottom-up FEPs or top-down safety functions) can work. Regardless of the path taken, the conceptual model should include significant FEPs that can lead to materially different performance states of the system. The combined effect of included FEPs is represented in the dynamically responsive conceptual model.

The following example is used to illustrate the differences that may occur between the bottom-up FEP screening and safety function approaches to developing a conceptual model. At the closed commercial disposal facility in Beatty, NV, a large rainstorm resulted in a fire, explosion, and ejection of LLW barrels from the disposal trenches (NDPS, 2015)[14]. An investigation of the incident revealed that disposal of metallic sodium combined with subsidence of cover materials and episodic rainfall led to water contacting the sodium and the sodium catching fire. FEPs associated with this scenario would be:

Reactivity of the waste
Subsidence of the engineered cover
Episodic infiltration

The corresponding safety functions may be:

Release from the wasteform
Water flow through the engineered cover
Dimensional changes (subsidence) performance of the engineered cover

Disposal of reactive waste is prohibited using modern waste acceptance criteria, however, these disposals occurred prior to the introduction of modern waste acceptance criteria. If there were not criteria prohibiting the disposal of reactive waste, then this FEP would be included in the conceptual model. Likewise, subsidence of covers, especially older covers, has been observed to occur along with episodic infiltration that is driven by episodic precipitation. A bottom-up FEP approach would result in all of these FEPs being included in some form or another in the conceptual model or special scenarios.

The safety function approach would likely result in a different outcome if not applied cautiously. In a very arid environment such as Beatty, the evaporation rate (on an annual basis) greatly exceeds the precipitation rate. Under a properly functioning cover scenario, the expected infiltration rate would be very low, zero, or possibly even negative (upwards). A properly designed, implemented, and maintained cover would retain episodic precipitation and allow for evaporation. A one-off sensitivity analyses would conclude episodic infiltration has no effect. Cover subsidence, with low constant annual average precipitation, would likely be evaluated as having no impact because the moisture would still be simulated to evaporate and what little amount of water that makes it to the waste would not impact the release rates. Finally, reactive waste (metallic sodium) in the absence of water would have no impact. The safety function approach would likely eliminate the FEPs or conclude that the event sequence was impossible (or extremely low probability). The three FEPs used in this example are interrelated and have

[14] DOE has not proposed to dispose of reactive waste in the IDF; this example is only to illustrate the different methods to develop models.

conditional probabilities and complex responses such that the event sequence probability was not low and was the dominant performance scenario of the site. If performed over in the same manner, the same outcome would likely have occurred; there was not a special combination of rare conditions that produced the outcome. DOE may successfully use the safety function approach for simple sites with limited uncertainties but may be challenged to produce reliable site conceptual models for more complex sites with numerous uncertainties.

A bottom-up FEP approach is designed to ensure that all relevant FEPs are included in the development of the site conceptual model. The top-down safety function approach ensures that safety functions are identified and evaluated but does not ensure that the conceptual model is complete. The safety function approach ensures that changes to a conceptual model are considered. As stated in the review of WMA C, there is not a clear mechanism by which safety functions produce alternate conceptual models as implemented by DOE (NRC, 2020b). While the DOE central scenario or base case is plausible, other plausible scenarios may be generated. DOE evaluated many of these scenarios in sensitivity cases. The uncertainties associated with these other scenarios are equally relevant to the decision-making process as the deterministic results from the central scenario.

A simplified representation of DOE's site conceptual model is shown in Figure 4-1. The main components of the conceptual model for the groundwater pathway are the engineered cover, infiltration to the waste, release from the wasteforms, transport to the unsaturated geology, transport through the unsaturated geology, dilution and dispersion in the aquifer, and pumping of contaminated water for use by different receptors. The engineered cover was assumed to divert infiltration until degraded. Long-term infiltration rates were based on present-day naturalized conditions. Performance of stainless-steel canisters containing the glass waste was not credited. Release rates of radioactivity from the glass wasteforms were based on laboratory experiments of select glass compositions. Radioactivity that was released was transported by diffusion and advection through the engineered barriers to the surrounding geology. Some lateral dispersion was simulated during transport in the unsaturated zone; however, the vertical transport direction was vertically down driven by recharge (gravity). Because the flux rate of water was projected to be so much lower through the unsaturated zone than the underlying aquifer, significant dilution (mixing) of contaminants occurred as the flux of water through the unsaturated zone entered the saturated zone. Dispersion and dilution were the primary phenomena in the saturated zone. This conceptual representation of the groundwater pathway is reasonable and plausible, and it was supported by a variety of technical studies.

4.2.3 NRC Conclusions on Scenarios and Conceptual Models

The NRC staff's conclusions on scenario and conceptual model development for the IDF PA are consistent with the staff conclusions for the review of WMA C:

- Due to the overall safety margins in the results analyzed, including uncertainty and sensitivity analyses, the NRC staff finds that DOE has adequately developed appropriate conceptual models and scenarios for disposal of certain wastes at the IDF.
- Scenario and conceptual model development are a significant source of uncertainty. DOE's safety function methodology is not able to identify all significant interdependencies and interrelationships between FEPs that could result in plausible alternative future scenarios or alternative conceptual models. Uncertainty and sensitivity analyses, including one-at-a-time

sensitivity cases, do not identify risk-significant interdependencies and interrelationships between features and phenomena.

The NRC staff has no new recommendations on this topic.

4.3 Climate and Ecology

Climatic conditions and the ecology in the environment in and around the disposal system are primary factors that influence fluxes (e.g., water, air, heat, biota) used to estimate releases of radioactivity through various pathways. Climate and ecology are closely interrelated, and both contribute to projecting the future performance of the disposal system. Climate states can influence the ecology. The ecology can impact the amount of water present, the rates of water movement, disruption by flora and fauna, and receptor exposure pathways. Some disruptive processes may be a function of the ecology.

Climate can impact the durability of an engineered cover and other engineered barriers. The temporal and spatial patterns of precipitation and evapotranspiration are a primary determinant of infiltration and recharge rates. Infiltration, or the water that enters the subsurface, is the driving mechanism of transport of radioactivity through water pathways. Recharge (the infiltration that penetrates below the root zone and eventually reaches the water table) is used to estimate travel times of contaminants to the underlying aquifer and the amount of dilution when contaminants enter the water table. Direction and magnitude of winds is a primary determinant of the concentration of radioactivity in the atmosphere that results from fluxes of airborne contaminants that may be emitted from a disposal facility.

4.3.1 DOE's Analyses of Climate and Ecology

Climate can be evaluated in terms of different temporal regimes: past, present, and future. Evaluation of past and present climate can be used as a basis for projecting future climate states. The current climate of the Pasco Basin, where the Hanford Site is located, can be classified as either mid-latitude semi-arid or mid-latitude desert, depending on which climatological classification system is being used.

Climatological data for the Hanford Site has been collected and processed at monitoring sites since the early 1980s (PNNL-15160, 2005; PNNL-6415, 2007). Prior to the 1980's, information was collected at nearby towns. Based on data collected from 1946 through 2004, the average monthly temperatures ranged from a low of -0.2°C (32°F) in December to a high of 24.6°C (76°F) in July. The maximum temperature recorded was 45 °C (113 °F) while the minimum temperature recorded was -30.6 °C (-23.1 °F). Summers are hot and dry while winters are cold and relatively more wet. There are large diurnal temperature variations. Normal annual average relative humidity is 54% (76% winter, 36% summer).

Based on observations, average annual precipitation at the Hanford Site was 17 cm (6.7 in). The driest season was the summer of 1973, when only 0.1 cm (0.04 in.) of precipitation was measured, while the wettest season on record was the winter of 1996-1997 with 14.1 cm (5.6 in) of precipitation. Most precipitation occurred during the late autumn and winter with approximately 50 percent of total precipitation occurring from November through February. Snowfall accounted for about third of that amount. The seasonal record snowfall was 142.5 cm (56.1 in).

DOE evaluated analog information for past climate data to estimate the range in variation of temperature and precipitation and to infer potential ranges in future climate states. In the report PNNL-13033, DOE represented future climate conditions by scaling the current temperature and precipitation data to match paleoclimate observations derived from pollen data (PNNL-13033, 1999). A 125,000-year paleoclimate record was constructed from the pollen record in cores taken from Carp Lake, near Goldendale, Washington. Carp Lake is located ~175 km (~109 mi) southwest of the Hanford Site. The Carp Lake location was used as a proxy for paleoclimate information for the Hanford Site. For the Holocene (i.e., the last 10,000 years), DOE indicated that the data suggested that annual temperatures and precipitation ranged from 0 to 2.8 °C (0 to 5 °F) warmer and 0 to 50% drier compared to the modern climate. During the glacial period prior to the Holocene, annual temperatures ranged from 0.2 °C (0.36 °F) warmer to 2.5 °C (4.5 °F) cooler and precipitation ranged from 75 to 128% of modern levels.

Based on the paleoclimate data, DOE stated that the range of precipitation and temperature conditions over the last 100,000 years were expected to bracket the future range in precipitation and temperatures. The change in precipitation and temperature translated into a range of possible future net infiltration conditions (see Section 4.4). DOE did not expect the groundwater flow directions and water table levels to be significantly impacted by the range of expected future recharge conditions due to the high transmissivity of the Hanford formation.

Uncertainty in the future net infiltration and recharge conditions was addressed in sensitivity and uncertainty analyses. Probability distributions for the range in net infiltration for future climate states was not changed from the range representing natural variability. To propagate uncertainty in post-closure recharge rates following degradation of the surface cover, DOE used a triangular probability distribution with a minimum, maximum, and mode of 0.5 mm/yr (0.02 in/yr), 5.2 mm/yr (0.2 in/yr), and 1.9 mm/yr (0.07 in/yr), respectively. DOE indicated the uncertainty range was the same as that chosen for the background infiltration rates. The prescribed uncertainty covered the natural variability observed in annual precipitation in the climate record for the last 100,000 years. Changes to the site ecology, disruptive processes or events and other phenomena resulting from changes to climate were not implemented in the performance analyses. DOE made the explicit assumption (indicated in Table 2-13 of the IDF PA) that the observed present-day conditions were applicable for the next 1,000 years and longer.

DOE provided a summary of the site ecology (primarily taken from PNNL-6415). In the report PNNL-6415, both terrestrial and aquatic ecology are discussed. The IDF is in an area of the site that does not have nearby aquatic resources; therefore, emphasis was placed on terrestrial ecology. DOE indicated that most of the site is comprised of undeveloped land (about 6 percent of the area is currently developed). The site was used in the past for agricultural purposes with some tillage and grazing. No farming has occurred at the site since the government took control in 1943.

The site is a shrub-steppe ecosystem that has adapted to the mid-latitude, semi-arid climate of the region. The ecosystem is dominated by a shrub overstory with a grass understory. In the early 1800s, dominant plants in the area were big sagebrush with an understory of grass species. Agricultural production contributed to the colonization of the site by non-native vegetation species that currently dominate portions of the landscape. The dominant, non-native species, cheat grass, is an aggressive colonizer and has become well-established across the site. Wildfires have occurred and have altered the ecological environment. Non-native species supplant the native species after a fire event. DOE stated that the primary impact on the disposal system by flora and fauna would be roots penetrating and animals burrowing through surface barriers into the disposal facility. In addition, DOE indicated that the types of plants and animals and their density can affect net recharge to groundwater. Most of the waste disposal and storage sites are covered by non-native vegetation or are kept in a vegetation-free condition by the controlled application of approved herbicides. Wildlife use of actively managed areas of the site is limited, however, a variety of animals such as badgers, coyotes, pocket mouse, and mule deer have been recorded in surveys.

4.3.2 NRC Evaluation of DOE's Analyses of Climate and Ecology

The NRC staff reviewed the information provided in the IDF PA and supporting documents. Staff determined that information on climate and ecology was complete and accurate for its intended use. DOE's use of paleoclimate data to develop ranges of precipitation and temperatures was an appropriate approach. Precipitation and temperatures were not used directly in the PA model but rather were used to develop ranges for net infiltration (see Section 4.4) under future climatic conditions. DOE's exclusion of the biotic pathway as a release mechanism was appropriate. The NRC staff reviewed various documents on biotic pathways.

DOE made the explicit assumption that the observed present-day conditions are applicable for the next 1,000 years and longer. The NRC staff does not support the technical accuracy of this assumption, however, does not view the impacts as being risk-significant. The magnitude and timing of anthropogenic climate change is a subject of considerable debate and uncertainty. Historical (paleoclimate) data does not reflect this new change to climate. However, the risk-significance of this FEP to the disposal system is not likely to be large if the range of infiltration rates projected from paleoclimate data is sufficiently large and an engineered cover with robust erosion protection is utilized. The IDF is a deep engineered trench in an area of limited relief. The more deeply waste is placed below the land surface, the less likely releases of radioactivity from the waste will be impacted by surface phenomenon. This does not diminish the magnitude of the impacts that can occur during climate transitions. Transitions from one climate state to another can result in large changes to the near-surface environment, as was inferred from scientific investigations of the transition to the current interglacial period (i.e., the Missoula Floods) for the location of the present Hanford Site. From a radiological risk perspective, the impacts of such extreme events are likely to be insignificant because material released would be mixed in an enormous volume of water and sediment.

Biota (e.g., animals and plants) can impact a waste disposal facility by providing a direct pathway for the release of radioactivity or by disturbing engineered barriers. DOE did not identify any plants and animals likely to reach the depth of the disposed waste. The barrier most likely to be impacted by biota is the engineered cover. Deeply rooted plants could penetrate layers of an engineered cover and animals could burrow into the cover. At the Hanford Site, herbicides are applied to waste disposal areas to eliminate vegetation that could uptake radioactivity. Herbicide application was not assumed to continue after closure of the disposal facility. Studies performed at the Hanford barrier experiment did not identify significant impacts to the moisture balance from biotic disturbance of the Hanford barrier (PNNL-14143, 2002; PNNL-18845, 2011). The observations cover many years, but they are still short in duration relative to the desired period of performance.

DOE also considered changes to the biotic system. DOE indicated that vegetation loss due to fires and firefighting activities exposed the soil at the Hanford Site to erosion by subsequent wind and rain and can enhance recharge by removing vegetation from evapotranspiration barriers. Current invasive species at the Hanford Site were discussed in the IDF PA (RPP-RPT-59958, 2018). Vegetation can play a key role in minimizing soil loss due to erosion and removing water by transpiration prior to water infiltrating below the root zone to the waste. Range or brush fires are common. Native and non-native vegetation can have different tolerances to fires. DOE completed a forced burn of vegetation on the Hanford barrier, and while changes to the soil structure were observed (due to the intensity of the heat generated) significant changes with respect to infiltration of water was not observed (PNNL-18934, 2009). If rock or a rock mulch is used to provide long-term erosion protection and the area is expected to be subjected to repetitive fires, then the rock should be tested to ensure it can maintain its dimensions and properties over the course of repetitive fires over long timeframes.

The NRC staff evaluated the information DOE provided associated with biotic pathways. DOE stated that the engineered cover will have a minimum depth of 5 m (16 ft) from the cover surface to the waste. Based on design drawings, that depth will likely be considerably larger for much of the plan view area, on the order of 10 m (33 ft) or more. DOE provided a summary of site-specific and generic penetration depths for biota at the Hanford Site and similar semi-arid conditions in report WMP-20570 (WMP-20570, 2006). Penetration depths were summarized in Table 4-39 of the PA (RPP-RPT-59958, 2018). The maximum depths a plant or animal was expected to cause disturbance was approximately 3 m (10 ft). The thickness of the cover would be sufficient to protect the waste from disturbance. Without significant erosion of the engineered cover, it is unlikely that the biotic pathway will significantly impact the performance of the disposal facility. In addition, the engineered cover will have a gravel layer designed to inhibit biointrusion by burrowing animals.

Features, events, and processes associated with climate and ecology also influence the biosphere especially the types of crops produced, the quantity of locally produced crops, and the amount and source of irrigation. Climate and ecology effects on the biosphere are discussed in Section 4.10.

4.3.3 NRC Conclusions on Climate and Ecology

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on climate and ecology. The NRC staff has the following conclusions with respect to climate and ecology:

- The information provided on climate and ecology was clear and complete.
- The influences of climate and ecology were properly integrated into the modeling and technical analyses.
- Because of the use of engineered barriers and relatively deep waste burial, the impacts of climate and ecology on releases from the disposal facility are significantly reduced.
- Extension of the present-day climate for future climate conditions is a reasonable proxy, given the risk context.

The NRC staff has no new recommendations associated with climate and ecology.

4.4 Infiltration

Infiltration, or the water that enters the subsurface, is discussed along with recharge (the infiltration that penetrates below the root zone and eventually reaches the water table) in the following two sections. The amount of infiltration is a function of the climate, or how much water is produced, and the resistance that water encounters to entering the subsurface. Flat surfaces and permeable materials promote infiltration. Climatic conditions are a primary factor in estimating releases of radioactivity through various pathways. The climate can impact the durability of an engineered cover and other engineered barriers and the temporal and spatial patterns of precipitation and evapotranspiration are a primary determinant of infiltration and recharge rates. Direction and magnitude of winds is a primary determinant of the concentration of radioactivity in the atmosphere that results from fluxes of airborne contaminants that may be emitted from a disposal facility.

The NRC staff is making a distinction between infiltration and recharge in this review. These values are expressed in a volume per unit area per unit time (e.g., units of mm/yr). Infiltration is the initial value for determining what may produce recharge. Infiltration may be diverted or focused by the presence of engineered barriers. Recharge may have a different temporal and spatial pattern than the infiltration. Recharge is the amount of moisture, including its temporal and spatial patterns, that reaches the unsaturated zone underlying the disposal facility. When no engineered barriers are present, or their performance has degraded, and evapotranspiration is insignificant, recharge is essentially equal to infiltration. The reason for making this distinction is the NRC staff has already reviewed DOE's information associated with infiltration at WMA C (NRC, 2020b) and DOE used essentially the same information for disposal of VLAW at the IDF.

4.4.1 DOE's Analyses of Infiltration

DOE used the terms "infiltration", "net infiltration", "natural recharge", and "recharge" interchangeably. The term "infiltration" was mostly used to describe water that falls as precipitation and is not lost by evaporation or evapotranspiration after entering surface soils. "Recharge" was generally used to refer to deeper flows that can eventually reach the underlying aquifer. The term "rate" may also be used to indicate the difference between the value (e.g., 5 cm/yr) from the process (e.g., transfer of a material over time). The natural features of the site (e.g., geology, plant community, slopes, soil properties) determine the amount of infiltration that results from precipitation or anthropogenic sources of water. DOE included a multi-layer engineered cover system in the design of the IDF to limit the amount of water that will result in infiltration and recharge. The engineered cover system is discussed in Section 4.5.

DOE has studied infiltration at the Hanford Site for multiple decades. DOE used the same information for infiltration for VLAW and WMA C (NRC, 2020b), as both WIR Evaluations cover waste located the Hanford Site. DOE referenced a variety of reports completed over the last 25 years as basis for their prescribed infiltration values (Gee et al., 1992; Fayer et al., 1996). Values for infiltration ranged from near 0 to over 100 mm/yr (4 in/yr), the latter value being representative of bare gravel. Highest infiltration rates were associated with coarse-grained surface materials with little vegetation, whereas the lowest values were associated with fine-grained surface materials with relatively larger amounts of vegetation. Infiltration was observed to be highly variable both spatially and temporally. Rapid snowmelt and other similar processes were observed to contribute to temporal variability. DOE stated that for the broader Hanford Site, the magnitude of recharge at a particular location is influenced by five main factors:

climate, soils, vegetation, topography, and springs and streams. At the location of the IDF, natural topography is relatively flat and no springs or streams are present. DOE indicated that events also affect recharge rates, such as the fire that burned vegetation from a large portion of the Hanford Site during the summer of 2000.

DOE stated that the base case value for infiltration at the IDF, after the closure cover was assumed to no longer be performing effectively, was pessimistically assumed to be 3.5 mm/yr (0.14 in/yr). The assumed value of 3.5 mm/yr (0.14 in/yr) was higher than the expected long-term average value of 1.9 mm/yr (0.07 in/yr). DOE indicated that there was uncertainty in the estimate of the long-term average net infiltration rate due to uncertainty in the possible long-term changes in soil conditions (due to slow-moving dunes) and vegetation (due to non-native plant species). DOE did not include these uncertainties in the estimates for long-term average net infiltration rates.

DOE performed analyses to evaluate the significance of uncertainty in infiltration and to provide relevant risk-related information. Table 4-2 provides a summary of the results. The magnitude of doses in the timeframe prior to 1,000 years (i.e., DOE's compliance period under Order 435.1) were all very low but highly sensitive to the infiltration rate. This is because only long-lived mobile radionuclides can reach the water table and they have not fully achieved breakthrough at the water withdrawal point of the receptor. The magnitude of doses after the 1,000-year timeframe were less sensitive to infiltration rates. With an infiltration rate of 5.0 mm/yr (0.2 in/yr), the peak dose was less than $2x10^{-2}$ mSv/yr (2 mrem/yr). DOE concluded that the information available was sufficient to support the infiltration values assigned and that the risk context of the information was properly understood.

Table 4-2 Sensitivity of Doses to Infiltration Rates

Infiltration Rate* (mm/yr)	Maximum Dose within 1,000 Years (mrem/yr)	Peak Dose after 1,000 Years (mrem/yr)	Time of Peak Dose (yr)
3.5	$2.15x10^{-4}$	1.65	6,415
1.7	0.00	0.99	5,854
5.0	$2.85x10^{-2}$	1.93	4,574

* long-term infiltration rate after the closure cover has degraded

4.4.2 NRC Evaluation of DOE's Analyses of Infiltration

The NRC staff performed a risk-informed review of DOE's infiltration analyses. The NRC staff reviewed DOE's information associated with infiltration at WMA C (NRC, 2020b) and DOE used essentially the same information for disposal of VLAW at the IDF. The NRC staff review of infiltration focused on differences. The waste disposed at the IDF will contain many different radionuclides ranging from short- to long-lived and from mobile to immobile. Because the Hanford Site has a thick vadose zone (approximately 80 m (260 ft) distance to the water table), short-lived contaminants decay during transport and do not reach the water table except under very high infiltration rates (bare gravel conditions with high water usage or application rates). Infiltration is significant only for those radionuclides with low affinity for sorption to geologic materials. In the up to 10,000-year timeframe, a high fraction of the projected groundwater doses comes from ^{99}Tc and to a lesser extent ^{129}I, both long-lived radionuclides with high mobility (low sorption). The risk-significance of infiltration is tied to the engineered barriers and wasteforms that are used and how much of the infiltration eventually contacts the waste.

Infiltration varies both temporally and spatially. DOE described three main components as drivers of infiltration at the Hanford Site: climate, surficial soil types/structures, and plants that are present. All three of these components are dynamic and can be interrelated. DOE appropriately described the climate as semi-arid with low annual precipitation rates and high annual rates of evapotranspiration. The combination of low precipitation and high evapotranspiration results in limited amounts of water that penetrates the root zone resulting in recharge. Staff review of the past, present, and future climate states is found in Section 4.3.2.

Over the long-term, surficial processes can alter present-day conditions. Infiltration at the Hanford Site has been shown to be strongly sensitive to the amount of fine-grained soils and types of plant species that are present. Fine-grained soils hold moisture more strongly than coarse-grained soils. The moisture held by fine-grained soils can be evaporated or evapotranspired. Wind is the dominant process that has resulted in redistribution of surficial soils since the great Missoula floods. Winds operate in concert with brush fires that destroy native vegetation allowing soils to become less cohesive and subject to aeolian redistribution. Stabilized sand dunes cover much of the Pasco Basin, but there are areas, such as along the Hanford Reach National Monument, where active sand dunes remain. Sand dune formation and redistribution is likely to continue to occur well after man no longer has any waste management activities at the site. After fires, non-native species may be more aggressive in establishing a biologic footprint, which can permanently change the water balance that may have been established based on historical and present-day observations. These uncertainties have the potential to affect long-term infiltration.

To evaluate these uncertainties, DOE performed simulations with infiltration rates that were about ten times higher than the long-term recharge rate assumed in the base case. They resulted in peak ^{99}Tc releases to the saturated zone that were four to five times higher than the base case peak result. The peak also occurred earlier (within the first 1,000 years after closure) but did not result in a dose exceeding the performance objective. The NRC staff agrees that a sensitivity analysis was an appropriate tool to use to examine this type of uncertainty and the results of the analyses were consistent with staff's understanding of the significance of infiltration. Because the uncertainties associated with long-term infiltration rates have the potential to have a compounding influence with other uncertainties that are discussed in this TER, it is recommended that DOE continue to invest in research such as long-term field studies of isotopic migration to quantify long-term infiltration rates that may apply to areas of disturbance and coarse-grained soils (e.g., dunes) (Recommendation #2). In the report PNNL-23711, it was noted that the southernmost 200 m of the IDF site is covered by a stabilized sand dune that is up to 8.0 m (26 ft) thick (PNNL-23711, 2015).

4.4.3 NRC Conclusions on Infiltration

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on infiltration. NRC had previously reviewed DOE's information on infiltration with respect to WMA C. The NRC staff has the following conclusions with respect to infiltration:

- The information provided on infiltration was clear and complete. Significant research has been completed to quantify infiltration values.

- Results of the impacts from the groundwater pathway can be sensitive to the long-term values applied for infiltration (after barrier failure).
- DOE provided adequate technical basis for present-day infiltration rates.
- Sensitivity analyses demonstrated that large changes to infiltration are unlikely to result in the performance objectives being exceeded.
- Uncertainty in the long-term infiltration values after barrier failure is moderate to high.

The NRC staff has the following recommendation associated with infiltration:

- DOE should continue to invest in research such as long-term field studies of isotopic migration to quantify long-term infiltration rates that may apply to areas of disturbance and coarse-grained soils (e.g., dunes) (Recommendation #2).

4.5 Near-field Hydrology

Near-field hydrology is used to describe what happens to moisture in the subsurface after it passes the root zone and until it enters the unsaturated (vadose) zone below the disposal facility. The engineered materials and barriers considered include the engineered cover system, wasteforms, disposal containers, engineered liner, sump system, and the natural soils immediately surrounding those barriers. Recharge at a specific location is determined by the soil, plant, and weather conditions that control the water balance at that location, as well as the flow through and around the engineered materials that are placed in the system. Near-field hydrology is the flow of moisture around and through the wasteforms, especially if they are cracked.

4.5.1 DOE's Analyses of Near-field Hydrology

DOE's representation of near-field hydrology consisted of the flow of water through the engineered features comprising the IDF, namely the surface barrier, backfill, and liner system. The liner is designed to collect potential releases during operations and for some period afterwards. The cover system is yet to be built and its performance has not yet been demonstrated as no waste has been placed within the disposal facility. DOE referenced studies on the Hanford Barrier Prototype to provide technical basis for the expected performance of the closure barrier (PNNL-17176, 2007). The closure cover is a thick, multi-layer barrier with approximately a 5 percent slope from the apex to the start of the side slopes along with much steeper side slopes. The closure cover will use a variety of different materials including structural fill, a gravel filter layer, sandy soil, an asphalt layer and base, and a drainage layer comprised of gravel. The thickness of the surface barrier will range from about 11 m (36 ft) at the intersection of the side slope and the top to about 16 m (53 ft) at the apex. Figure 4-4 is a cross-section of the proposed surface barrier.

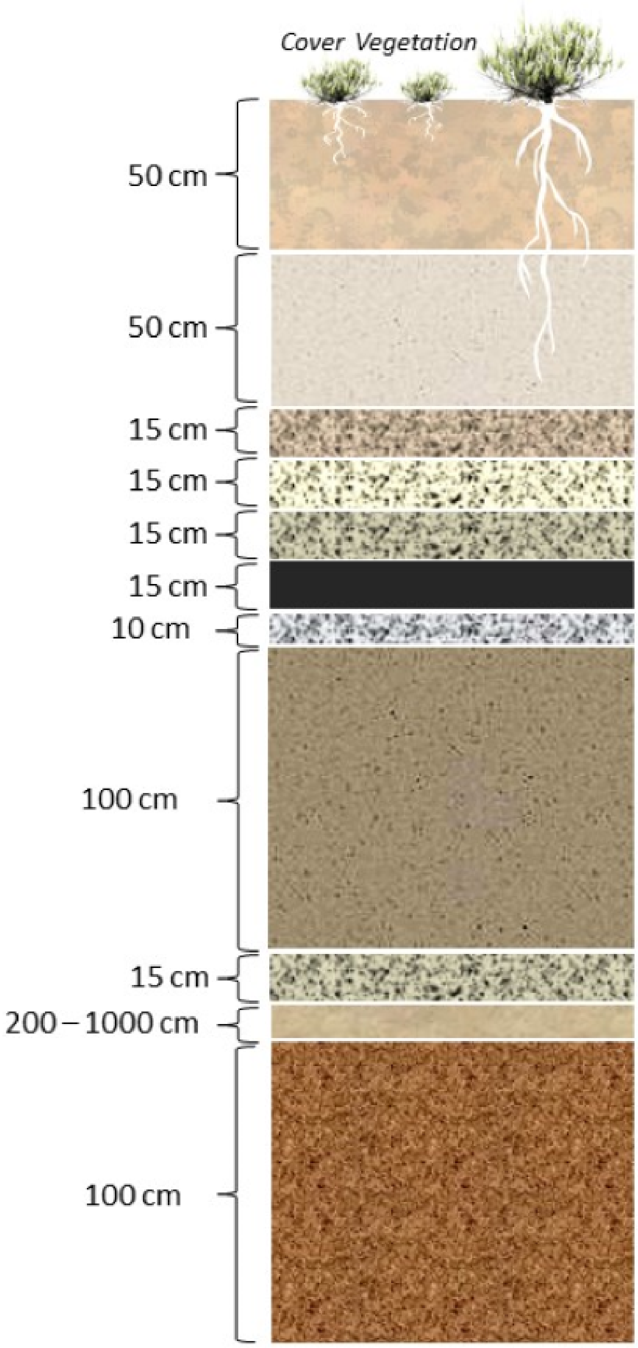
DOE documented computational modeling of near-field hydrology in report RPP-CALC-61029 (RPP-CALC-61029, 2017). In this calculation package, DOE described the steps taken to complete the modeling as well as the different cases that were analyzed. DOE focused on the base case scenario, where the cover and liner system degraded after 500 years. The computed flow distributions through the IDF were simulated for different surface infiltration rates selected to cover the possible range of infiltration rates (from 0.0 to 33 mm/yr [0.0 to 1.3 in/yr]) based on the data package reports (PNNL-14744, 2004).

The near-field hydrology model was a flow-only model; it did not simulate contaminant transport through the near-field. Near-field hydrology was analyzed using a two-dimensional vertical cross-section of the facility with the unsaturated flow capabilities of STOMP to evaluate water flow through the surface barrier, within the facility, and through the liner system. The near-field hydrology modeling provided boundary conditions for wasteform release modeling. Figure 4-5 provides an example of a two-dimensional cross-section used by DOE with additional detail provided for the cover and liner components. The color bar at the top of the figure reflects the different engineered materials that are present. The vertical exaggeration is a factor of 5 (z dimension). The x and y dimensions are large compared to the z (elevation) dimension.

DOE indicated that the flow rate through the engineered surface barrier is dependent on the assumed flow properties of the components of the engineered barrier, as well as the soil and vegetation at the surface that control the net infiltration into the barrier. The surface barrier was assumed to have a design life of 500 years. Prior to the design life being exceeded, the net infiltration was assigned a value of 0.9 mm/yr [0.04 in/yr]. During operations, infiltration was assumed to be captured by the liner and sump system of the IDF such that the recharge was 0.0 mm/yr [0.0 in/yr] even though small amounts of infiltration would flow through the system above the liner and sump system. After the design life of the liner is exceeded (i.e., 500 years), the net infiltration through the surface barrier returned to the ambient net infiltration in the native soils at the facility. A nominal value of 3.5 mm/yr [0.14 in/yr] was prescribed. Other infiltration and recharge values were considered in sensitivity and uncertainty analyses cases. Various scenarios were analyzed to examine the effects of deterioration in performance of different components of the system. These included the surface boundary condition (infiltration), surface barrier properties, liner system properties (admix layer, geomembrane), erosion of top layers of the surface barrier, and accounting for different times for degradation of surface barrier (earlier) compared to the liner system (later).

Uncertainties in the assumed characteristics of the surficial soils and other materials of the engineered surface barrier that affect the net infiltration into the IDF identified by DOE included: 1) vegetation and soil properties of the surficial soils that control evapotranspiration from the surface, 2) ability of the engineered barrier to maintain the slope of the capillary break that causes the lateral diversion of infiltrating water, and 3) properties of the layers that create the capillary break. It was assumed that the vegetation that reestablishes at the site after the surface barrier is installed is analogous to the current shrub-steppe vegetation. DOE stated that PA maintenance activities conducted during the time of institutional controls can be designed to test the as-built characteristics of the surface barrier, but that it is difficult to design a test to evaluate the larger-scale net infiltration rates over the range of expected meteorological conditions the surface barrier will be exposed to during the lifetime of the facility.

DOE’s near-field hydrology modeling provided the hydrologic boundary conditions for the release and transport modeling. Though the liner and sump portions of the facility are constructed, the facility has not yet accepted waste and backfill and closure covers are not installed; therefore, numerical modeling must be used to project future performance. As shown in Figure 4-5, a two-dimensional cross-section of the facility was simulated. The different engineered components of the facility have unknown, but estimated, service lives. For example, a buried geosynthetic composite layer (GCL) may have performance benefits well beyond that assumed in the base case analyses (i.e., 500 years). There are no direct natural analogs to these man-made materials and, therefore, a pessimistic approach to estimating future



Layer Number	Thickness In inch	Layer Description
1	20″	Silt/loam w/pea Gravel admix
2	20″	Compacted topsoil
3	6″	Sand filter
4	6″	Gravel filter
5	6″	Aggregate
6	6″	Asphaltic concrete
7	4″	Asphalt
8	40″	Sandy soil
9	6″	Gravel filter
10	80″ - 400″	Structural Fill
11	40″	Backfill operations layer - 1 meter above top of canisters

Figure 4-4 Cross-Section of the Proposed Surface Barrier Showing Key Materials
[Figure 3-106 of the IDF PA (RPP-RPT-59958, 2018)]

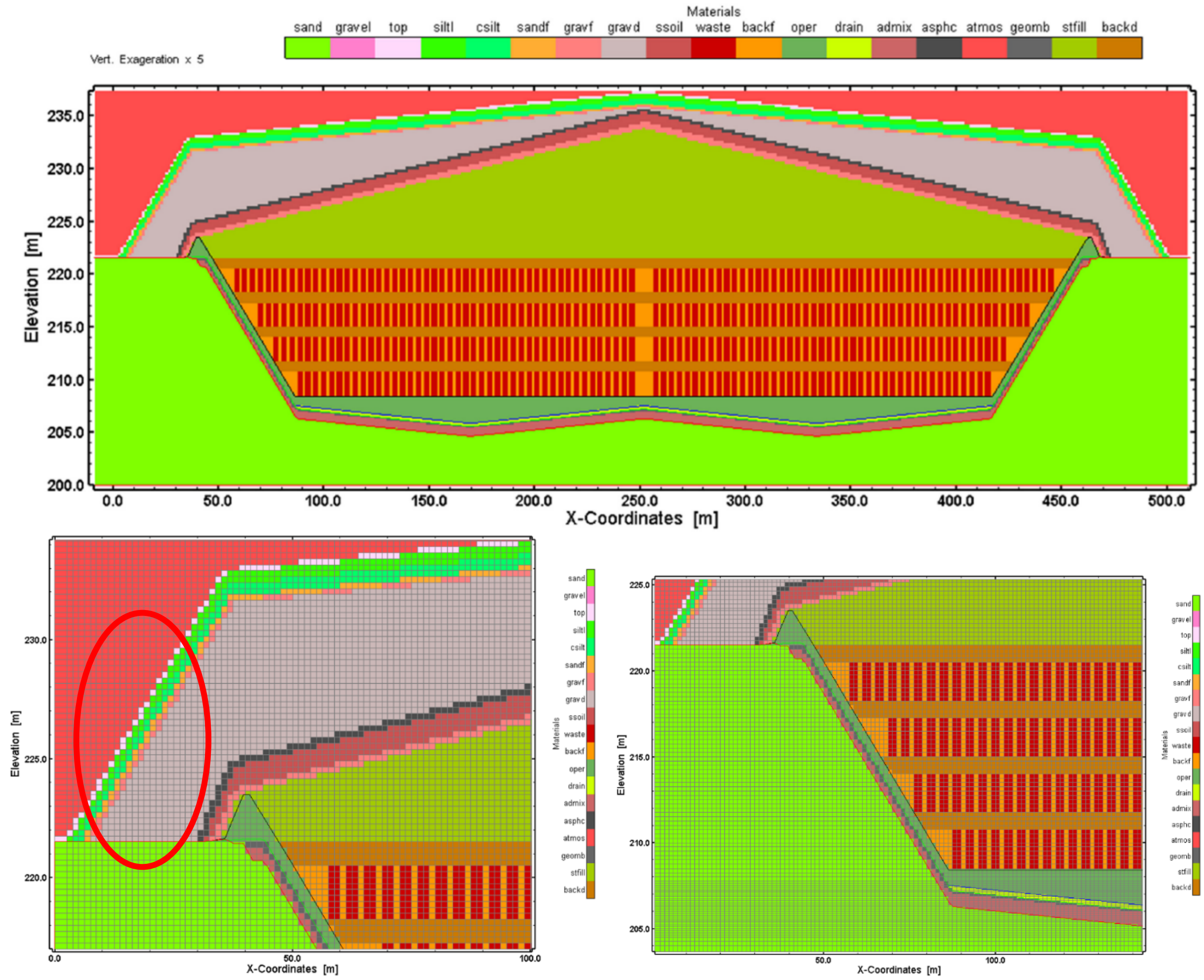


Figure 4-5 Vertical Cross Sections of Engineered Features Represented in Performance Modeling of the IDF
[Figure 2-3 of (RPP-CALC-61029, 2017)]

performance was warranted. Instead of trying to project the service life of the different materials, DOE elected to define time periods at which engineered barriers stopped performing their intended functions and identify the impacts to performance objectives. Because of the variety of different materials that were used in the design and uncertainty in the performance lifetime of those materials, different performance cases were examined to identify risk-significant performance characteristics.

For the near-field hydrology modeling, DOE used a traditional approach of assigning moisture characteristic curves to each media and uniform properties within a layer or material type. One outcome of this approach is there can be very large discontinuities in material properties at some interfaces that produce numerical results that may not be physical. For example, on page 5-4 of the IDF PA, DOE indicated that in some cases, the hydraulic conductivity of some layers was assumed to be different than provided in data packages because of numerical problems encountered with drastically different material types (RPP-RPT-59958, 2018). The problem domain that DOE needed to simulate was quite large (thousands of cubic meters), whereas some of the material layers are designed to be quite thin (scale of centimeters). As shown in Figure 4-5, the discretization of some of the layers in the numerical model was not continuous (shown outlined in red oval).

Water that exits the bottom layer of the cover system can reach the (future) disposed waste. Modeling was used to estimate how much of the water flows through the waste and how much flows around the waste. DOE used STOMP for modeling the wasteforms and near-field environment. The wasteform and near-field environment were described in terms of governing-conservation equations and constitutive functions. The governing coupled flow equations were partial differential equations for the conservation of water mass, air mass, and thermal energy. Constitutive functions related primary variables to secondary variables. The governing equations for thermal and hydrogeological flow processes were solved simultaneously using Newton-Raphson iteration. The governing conservation equations were discretized following the integrated finite difference, which is locally and globally conserving. This transformation required that the physical domain be spatially discretized into an orthogonal computational domain which comprises non-overlapping volumes (nodes). Intrinsic properties were assumed to be uniform over the volume domain. Intrinsic properties were defined for a node point at the geometric center of the volume. Flux quantities were defined at the geometric center of the surfaces between node volumes and along a direction parallel to the surface normal. Fluxes across node surfaces between neighboring inactive nodes and/or adjacent to the domain boundary were controlled through boundary conditions. Solution of the governing conservation equations in time required discretization of the time domain.

DOE used a nominal grid spacing of 15 cm (5.9 in) for wasteform release modeling and evaluated the impact of using a finer grid with a nominal grid spacing of 2 cm (0.8 in). The simulation with the finer grid spacing took approximately 6 weeks to run to completion. The calculated fractional release rate was 51 percent higher for the case examined. DOE concluded the differences seen in the results were not significant considering the extremely long run times. DOE needed to complete many simulations to examine sensitivities, which made use of the finer grid impractical.

4.5.2 NRC Evaluation of DOE’s Analyses of Near-field Hydrology

The NRC staff performed a risk-informed review of DOE’s analyses of near-field hydrology. The risk-significance of near-field hydrology for the IDF varies based on the timing and specific engineered components of the system. For instance, the liner and sump system are expected to be functional during operations. During this timeframe, the flow of water in components prior to reaching the liners and sumps is not significant. However, after the operational time period, the flow of water in these components can be significant. The most risk-significant aspect of near-field hydrology appears to be the amount of water that reaches the wasteforms and then flows through them. Though DOE considered the amount of water flowing through the waste under waste release modeling, the NRC staff elected to consider the amount of water flowing through the waste in this section because the same technical considerations are applied, and waste release modeling has a unique set of technical considerations that are not applicable to near-field hydrology.

The numerical model STOMP, which has been widely used at the Hanford Site, was used by DOE to simulate near-field hydrology. The NRC staff found that STOMP was an appropriate choice of a numerical model, proper QA procedures were applied to its use, and that STOMP had the necessary capabilities to simulate near-field hydrology. Numerous verification tests were applied for STOMP prior to its use. DOE presented documentation that qualified software was used. The staff using the software were qualified and the NRC staff verified that DOE completed independent checking of the results. DOE documents this information in Environmental Model Calculation Files (EMCF)s.

The data DOE used for material properties was clearly described in technical reports. Physical, hydraulic, and transport properties of sediments and engineered materials were provided by DOE (PNNL-23711, 2015). Properties and parameters that were estimated included particle size distribution, particle density, bulk density, porosity, water retention characteristics, saturated hydraulic conductivity, unsaturated hydraulic conductivity, dispersivity, and diffusion coefficients. DOE provided input parameters for the IDF cross-section models of the near-field in Table 4-1 of the report RPP-CALC-61029 (RPP-CALC-61029, 2017). Parameters provided included bulk density, porosity, permeability, and alpha α, n, and S_{rl} of the van Genuchten saturation function for 17 different materials. Some properties were based on limited measurements, which correlates with higher uncertainty. Permeabilities varied by up to 8 orders of magnitude, while the alpha parameter (related to the inverse of the air entry suction) varied by up to six orders of magnitude between different material types. The presence of these widely different materials (in proximity to one another) can result in challenges for successful execution of numerical models. As noted in Section 4.5.1, DOE experienced numerical problems and had to adjust some of the material properties assigned. While this is sometimes necessary in numerical modeling, especially for unsaturated flow simulations, it can indicate that the numerical model results may not be accurate. In DOE's approach, a given material type is assigned to an element in the numerical model and the numerical value of the material properties were then uniform for all elements assigned that material type. In other words, there was no intra-material variability. Effects such as capillary barrier phenomenon may be enhanced in the numerical model results when uniform properties are assigned (Ho, 1998). It is recommended that DOE complete barrier flow analyses with stochastically (geostatistically) generated material properties at finer scales including spatial variability for a given material type (Recommendation #3).

The scale of the simulation is large with dimensions of 100s of meters in each direction. The scale of some of the barrier layers is on the order of centimeters or even smaller for the case of fractures in glass. The NRC staff asked questions about DOE's discretization in the numerical model in its RAI (NRC, 2020a). In their response to NRC, DOE summarized the model discretization that was used (DOE, 2021c). For near-field modeling away from the wasteform, the vertical thickness of each node in the near-field flow model was predominantly 0.25 m (9.8 in). Near the apex of the asphalt layer of the surface barrier, the node thickness was reduced to 0.15 m (5.9 in) for three node layers; the thicknesses of the nodes containing the liner system were 0.125 m (4.9 in); and the top of the vadose zone below two 0.25-m (9.8 in) thick grid cells at the bottom of the liner system had a node thickness of 0.5 m (20.0 in). Some layers in the design have thicknesses that are less than the node thickness. Given the slopes in the barrier design and the vertical thickness of these nodes (elements), DOE did not have continuous layers of some materials in their numerical model. As shown below in Figure 4-6, the model discretization introduced numerical artifacts in the patterns of liquid flow and saturation within the near-field.

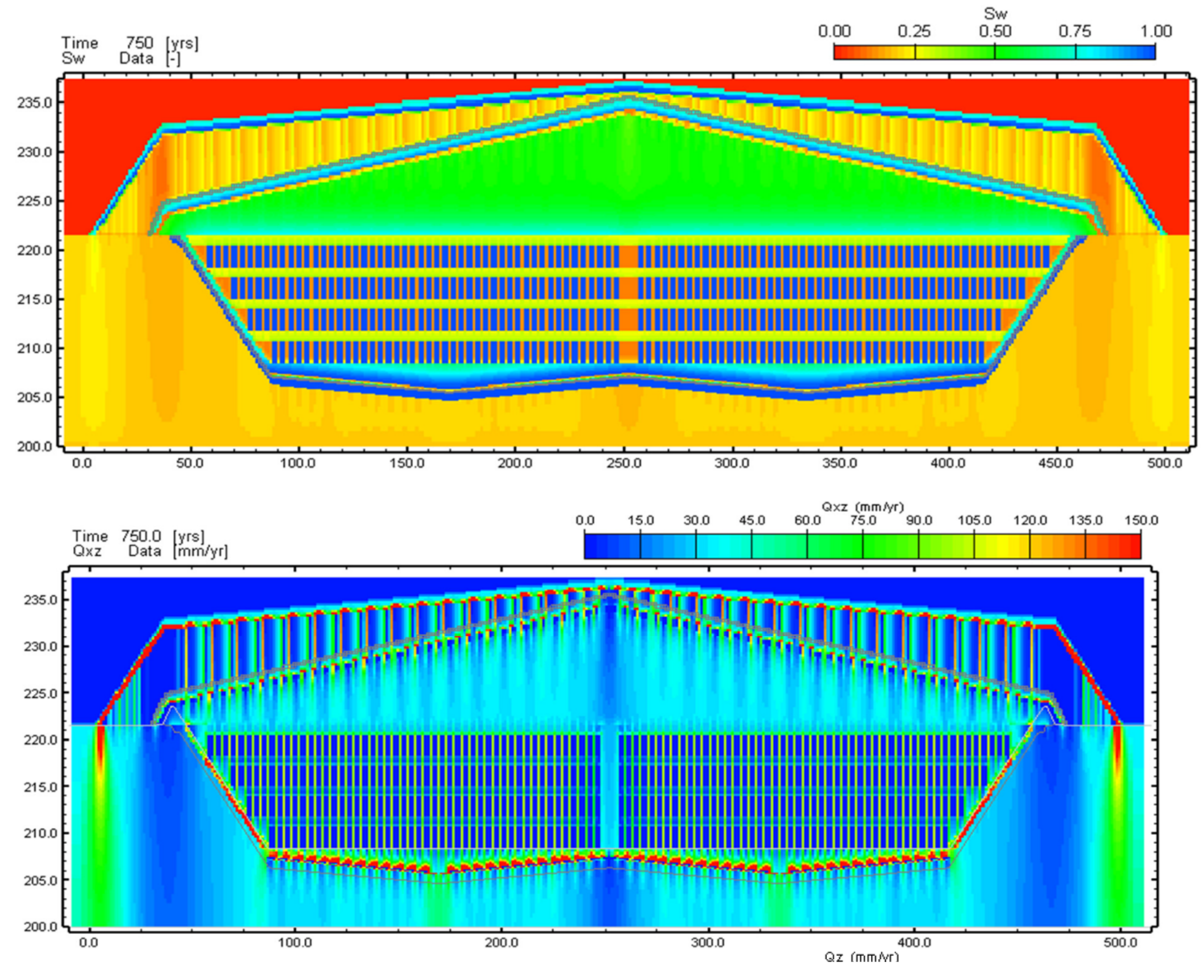


Figure 4-6 Simulated Near-Field Liquid Saturation and Fluxes
[Figure 5-11 of (RPP-RPT-59958, 2018)]

DOE acknowledged the discontinuities in the layers in the model but stated that the near-field hydrology modeling was used to define boundary conditions for the wasteform release and transport modeling. The near-field flow model was used to develop estimates of saturations in the backfill and wasteforms in the absence of long-term data and to evaluate the distribution of moisture movement into the vadose zone in the presence of an intact and a degraded liner system. The NRC staff believes it is unlikely that the artifacts DOE introduced by the discretization of the surface barrier significantly impacted the boundary conditions for the wasteform release modeling. The distance from the modeling artifacts in moisture distributions and flow to the waste is generally many meters, allowing time/space for the artifacts to be reduced. The amount of water reaching the glass wasteform and the amount of water flowing through the cracked glass wasteform were substantially different. If the magnitude of the diversion of water was sensitive to the boundary conditions, then the discontinuities in the overlying layers could be significant. It is recommended that DOE evaluate if the numerical grid used was sufficient and eliminate numerical modeling artifacts to the extent possible in future evaluations (Recommendation #4).

The real system is three-dimensional, but the modeling performed by DOE was two-dimensional. As currently envisioned, the slope of the top of the engineered cap is relatively flat with much steeper side slopes. The top will likely have a soil/gravel mixture where vegetation will be present to promote evapotranspiration. The side slopes will likely need to be armored with larger rock to provide erosion protection. Therefore, the side slopes are likely to experience much higher infiltration. Because a geosynthetic liner is used in the design, additional moisture resulting from higher infiltration at the boundaries is unlikely to reach the waste while the liner system is functioning. The simplification from three to two dimensions was necessary to reduce execution time to more manageable values. Because the overall areal extent of the facility is large in comparison to the area of the side slopes, the influence of the dimensionality reduction is likely to be minimal and only will impact simulated moisture distributions at the corners. The NRC staff concludes the impacts are not risk-significant.

DOE performed flow and transport modeling for releases from the wasteform (RPP-CALC-61031, 2017). This modeling used an approach similar to that described above for the rest of the engineered features in the system. Uniform properties were assigned to a given material type within the numerical model. DOE provided rates of water flow through the wasteform for different recharge rates (Table 5-1 of RPP-CALC-61031). Flow rates (cm^3/yr) per unit area through the cracked glass wasteform were on average a factor of 11 less than (range of 10 to 12) that which would be expected based on the recharge values. Most of the water that reached the wasteform was modeled to flow around the wasteform. Water can be more strongly held in the fine-grained material when fine-grained material is next to coarse-grained material (capillary barrier effect). Capillary effects can result in diversion of water. The size distribution of the aperture width of fractures in glass is likely to be broad, ranging from microns to possibly centimeters. Fine-grained material would be expected to progressively fill some of the fractures after corrosion of the disposal canister. The comparatively coarse numerical grid used in the modeling is unlikely to accurately represent the flow at the interface between the fine-grained material and the cracked glass. In addition, limited information is available to specify moisture characteristic curves for fractured glass. Capillary barrier effects can be sensitive to parameters such as air entry pressures.

In its RAI, the NRC staff asked if DOE had information to support the magnitude of flow diversion that was shown in the STOMP model results (NRC, 2020a). DOE stated they did not have any additional information. DOE indicated that they would collect data from field-scale lysimeter experiments that may provide information for model support. Those experiments will be completed over decades and collecting information on flow rates when the flow rates are small may be difficult. The magnitude of flow diversion simulated in the model may be accurate, however, the NRC staff recommends that DOE complete laboratory measurements of flow through cracked glass surrounded by porous material to provide model support because an order of magnitude reduction in flow is likely to be risk-significant (Recommendation #5). DOE has produced large-scale glass specimens during testing and development activities. These specimens could be used to verify the simulated flow behavior at the glass-porous media interface. Without adequate experimental data or other model support, the correctness of the numerical modeling results cannot be fully ascertained. DOE used information from the Sisson and Lu experimental site (RPP-RPT-59344, 2016) to verify numerical modeling with STOMP and indicated that future model support will be developed under the PA maintenance program. Some hydrologic systems can be slow to respond when infiltration is small. Collection of large-scale observations may be difficult under these conditions. A combination of field observations and additional finer-scale numerical experiments may be sufficient to address the remaining uncertainties.

To account for uncertainties associated with the timing of degradation of different system components, DOE used conservative assumptions to overestimate the impacts on release, and therefore, risk. The infiltration through the engineered cover was assumed to revert to natural values at 500 years after closure. The GCL was assumed to fail over the full areal extent at 100 years. The stainless-steel waste canisters were assumed to provide no performance benefit with respect to hydrology. The drain and sump system were assumed to no longer be maintained after 100 years (sumps were assumed to no longer be pumped). Because the failure or degradation times are not known, DOE examined different cases where the timing of system component degradation was different from the base case. The cases examined by DOE were appropriate. However, given the different materials used for different systems, the number of different permutations of performance cases could be large.

In a model of a complex system, it can be difficult to identify states of the system that are the most conservative because performance impacts can be non-linear and complex. It may be useful for DOE to allow the various system components' performance to be represented probabilistically to identify combinations that result in lower performance and then determine if the combination of those component states is credible. The NRC staff reviewed the assumed degradation or failure times and determined that the assumed values are generally pessimistic, given current understanding. Buried stainless steel is likely to provide a barrier to moisture contacting the waste or release of radioactivity from the waste for hundreds to thousands of years or longer. The buried asphalt layer of the engineered cover may provide performance longer than assumed by DOE. Buried GCL liners have been estimated to provide performance much longer than 100 years; however, direct validation of the estimates is not possible. Given the uncertainties, the approach taken by DOE was reasonable.

4.5.3 NRC Conclusions on Near-field Hydrology

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on near-field hydrology. Near-field hydrology described the flow of water inside the disposal system. The NRC staff has the following conclusions with respect to the review of near-field hydrology:

- Near-field hydrology is important with respect to the performance of the disposal system.
- The NRC staff found that STOMP was an appropriate choice of a numerical model, proper QA procedures were applied to its use, and that STOMP had the necessary capabilities to simulate near-field hydrology. The staff using the software were appropriately qualified.
- The data used in the analyses was clearly described.
- As a result of coarse discretization, considerable modeling artifacts were observed in the results. Staff conclude it is unlikely that the artifacts DOE introduced by the discretization of the surface barrier significantly impacted the boundary conditions for the wasteform release modeling.
- Limited verification and model support data exists for the flow of water through the wasteforms.

The NRC staff has the following recommendations associated with near-field hydrology:

- DOE should complete barrier flow analyses with stochastically (geostatistically) generated material properties at finer scales including spatial variability for a given material type (Recommendation #3).
- DOE should evaluate if the numerical grid used was sufficient and eliminate artifacts associated with coarse discretization of numerical models in future analyses (Recommendation #4).
- DOE should complete laboratory measurements of flow through cracked glass surrounded by porous material to provide model support (Recommendation #5).

4.6 Glass Wasteform Performance

Wasteform performance is a risk-significant component of the PA. The primary wasteform to be disposed in the IDF is a borosilicate glass[15]. Secondary wasteforms are generally cementitious in nature though some may be encapsulated wastes of different compositions. The wasteform provides a stable media to limit void formation after closure, limit the dispersibility of the waste upon disturbance, slow the release of radioactivity to the environment, and provide self-shielding. The wasteform can also buffer or control the chemical environment that may be beneficial (or detrimental) to the performance of other engineered barriers. A large amount of diverse research was completed over multiple decades to develop data and numerical models to represent wasteform performance in the IDF PA (RPP-RPT-59958, 2018).

4.6.1 DOE's Analyses of Glass Wasteform Performance

The glass wasteform release models were used by DOE to evaluate the movement of water, radionuclides, and chemicals released from the waste containers in the near surface of the IDF to the top of the vadose zone. The simulated release of constituents of potential concern (COPCs) were used as input to the vadose zone/saturated zone flow and transport model.

System barriers affecting the release of radioactivity are the wasteforms, containers, backfill around the containers, filler, and overlying cover. DOE collectively referred to the wasteform, container, and filler together as a "waste package". DOE did not prescribe credit to the waste containers in the analyses. The filler material was assumed to have the same physical, hydraulic, chemical, and mineralogical properties as the backfill. DOE plans to place waste containers in the IDF in layers or lifts as many as four high. Backfill would be used between containers. DOE estimated a nominal vertical spacing of 1 m (3.3 ft) and a nominal horizontal spacing of 0.1 m (0.33 ft).

DOE established the performance of the glass wasteforms based on multiple decades of multidisciplinary research and development. Borosilicate and other glasses have been used in a variety of radioactive waste management programs to immobilize radioactivity to achieve long-term disposal objectives. Figure 4-7 is DOE's conceptual representation of glass release (RPP-

[15] The production of glass wasteforms will result in failed and spent melters. These melters will contain residual amounts of glass that will harden on surfaces. The performance of the glass on the surfaces of the melters was assumed to be equivalent to the performance of the glass wasteforms. The volume of glass in melters is about 5% of the total.

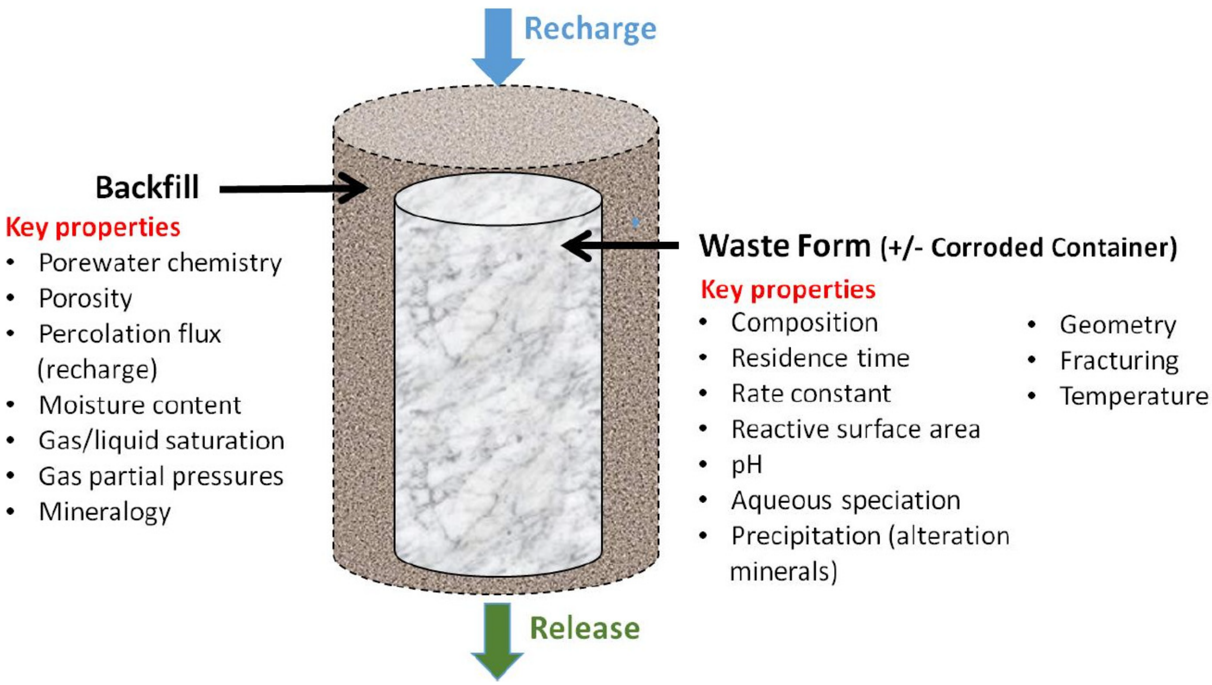


Figure 4-7 DOE Representation of Key Properties for Glass Release
[Figure 4-8 of the IDF PA (RPP-RPT-59958, 2018)]

RPT-59958, 2018). Key properties of the surrounding environment include the porewater chemistry, porosity, recharge, moisture content, gas and liquid saturation, gas partial pressures, and minerology. The properties of the surrounding environment establish the rate of transfer of moisture and chemical species to the glass surface. Key properties of the glass wasteform include the composition, rate constants, reactive surface area, pH, aqueous speciation, alteration minerals, geometry, fracturing, and temperature. DOE envisions tailoring the composition of the glass to accommodate the diverse secondary waste streams that will be generated from processing the waste contained within the tank farms.

DOE indicated that the dissolution rate of the glass matrix may depend on the recharge rate and chemical composition of the water that contacts the glass matrix. The composition of this water will depend on the initial composition of rainfall/snow melt entering the top of the IDF, chemical reactions that occur as the water percolates and infiltrates through the various cap and backfill materials, and interactions with the stainless-steel container and the dissolving waste glass. The initial water composition used in the PA analyses by DOE did not account for reactions with these materials but did account for changes once the water contacted the glass.

In the DOE conceptual model, pore water flowed by advection vertically downward through partially saturated backfill beneath the cover and through fractures in the glass. The residence time of pore water in contact with the glass was controlled by the velocity, which in turn was a function of the rate of water flow and the amount of moisture present (liquid saturation). The glass was assumed to dissolve at a rate that is compatible with Transition-State Theory (TST) and with experimentally determined kinetic parameters. In the model, if evolved solution compositions reached saturation with one or more secondary minerals those minerals were assumed to precipitate under local equilibrium or kinetic control in the backfill or on glass surfaces. Aqueous concentrations and the total contaminant inventories in the glass wasteforms were used to calculate normalized fluxes, or fractional release rates, across a transverse plane located at or slightly below the bottom of the lowest wasteform within the IDF.

The parameters for the expression describing release from the wasteform were determined from experiments on different glass compositions (PNNL-24615, 2015). The parameters were thought to be dependent on the glass and waste composition. The Hanford tanks contain many different types of waste (e.g., high-Na, low-Na, high sulfate). DOE derived parameters for glass compositions designated as LAWA44, LAWC22, and LAWB45. DOE indicated that additional parameters will likely be derived for future glass compositions and that parameters may be revised in response to operational experience.

DOE indicated that dissolution and ion-exchange reactions control the overall alteration behavior of borosilicate glasses. DOE stated borosilicate glass alteration will progress in stages, as shown in Figure 4-8. Fresh glass is more reactive and has higher alteration rates (Stage I). As the surface layers are depleted of some species, species build up in solution such as dissolved silicon, and as species from solution diffuse through an alteration layer, the rate of alteration slows (Stage II). Some glasses have been observed to reach a third stage (Stage III) where the rate of alteration increases. The understanding of Stage III behavior is developing. It is currently believed to be caused by precipitation of certain secondary minerals (zeolites) and is thought to occur mainly at temperatures above ambient. Minerals that precipitate were termed the “secondary mineral reaction network” or SMRN. DOE indicated in the IDF PA that they did not believe conditions would occur within the IDF to promote Stage III behavior (RPP-RPT-59958, 2018). Stage III behavior had been predominantly observed under closed conditions and at higher than ambient temperatures (PNNL-24615, 2015).

Key technical considerations for simulated glass performance include but are not limited to glass cracking, the secondary mineral reaction network and the potential for Stage III behavior, retention of volatile species in glass, the recycle of volatile species, glass and waste constituent compositions, and derivation of parameters from empirical data.

4.6.1.1 Glass Cracking

The amount of cracking present in glass is important to the PA because release of radioactivity from the glass (or other) wasteform is proportional to the specific surface area exposed. The degree of cracking in nuclear waste glass is largely driven by the cooling profile through the annealing range and the properties of the container in contact with the cooling glass. The significance of cracks in glass is also influenced by how accessible the cracks are to the ingress and egress of species. DOE indicated that the contribution of cracks to release is governed by factors such as the location of the cracks in the monolith, the geometric properties of the cracks, and the rate of corrosion of the glass and corrosion product diffusion. Cracks that are completely internal to the wasteform are not accessible to water infiltration. In cracks that are accessible to water infiltration, corrosion of the glass may form alteration products that can fill the cracks and result in self-sealing (Verney-Carron et. al, 2008). DOE explained that the relative rates of processes can be important to determine the significance of cracking in glass. If the rate of glass dissolution is rapid compared with the diffusive or advective movement of the dissolved glass components through a crack, then the accumulation of glass components in the pore fluid will slow the dissolution reactions. The significance of cracks to glass dissolution and

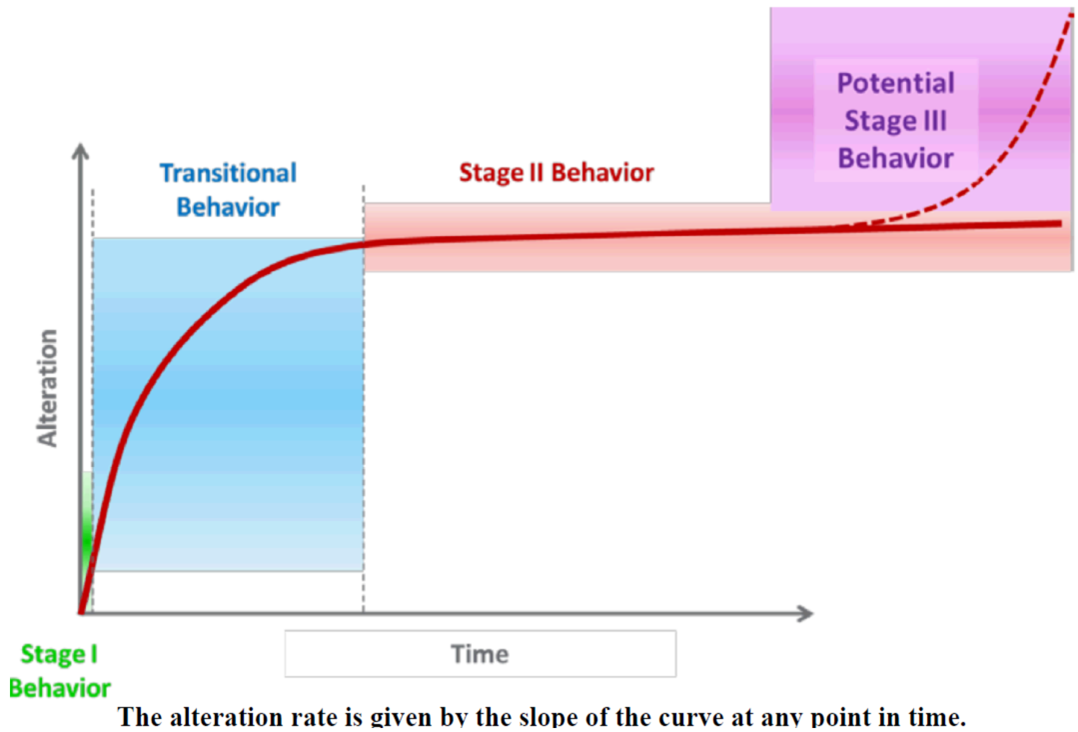


Figure 4-8 DOE Conceptual Representation of Glass Alteration Stages
[Figure 4-10 of the IDF PA (RPP-RPT-59958, 2018)]

release must be evaluated considering surface area as well as mass transfer and other processes. In the IDF PA, DOE applied a factor of 10 increase to the release of radionuclides from the glass wasteform to account for cracking of the glass. The basis for the factor of 10 was the report PNNL-13369, which states, "Fracturing is expected to increase the glass surface area a maximum of 10X over its geometric surface area" (PNNL-13369, 2001).

The reactive surface area increase of 10 times (10X) (to account for cracking) was introduced in the first ILAW PA in 2001, where it is stated that fracturing is expected to increase the glass surface area a maximum of 10X over its geometric surface area based on two references (Peters and Slate, 1981; Farnsworth et al., 1985). DOE stated that there is expected to be sparse fracturing based on prior experience with HLW glass. DOE examined the impact of uncertainty in the amount of fracturing with a one-off sensitivity case (RSA – Reactive Surface Area) where the reactive surface area was increased an additional factor of 10 over the base case. For two of the three glasses, the increase in fractional release rates were close to directly proportional to the increase in fracturing (specific surface area).

In its RAI, the NRC staff questioned the basis for the factor of 10. The staff cited the report PNNL-5947, which has numerous estimates of the increase in glass surface area, some of which were as high as 65 (PNNL-5947, 1986). The NRC staff indicated that DOE had not adequately described their operational plans and designs for cooling that could influence the amount of fracturing that may be expected. In response to the RAI, DOE discussed the rationale for the factor of 10 and described factors that could influence glass fracturing. Most importantly, the fraction of the crack surface area that is available to water for dissolution (β) must be considered. DOE described four different estimates for β that ranged from 0.13 to 0.4 (DOE, 2021c). In Table 2-8-3 of the RAI response, DOE indicated that the combined product of the factors influencing the importance of cracking in the glass was a maximum of 30. Because the peak groundwater concentration for the limiting radionuclide was below the EPA maximum contaminant level (MCL) by a factor of 10 and estimated doses were well below 0.04 mSv/yr (4 mrem/yr), an overall factor of 3 increase (30 divided by 10) would not be significant to the conclusions made by DOE about the PA modeling results.

4.6.1.2 Secondary Mineral Reaction Network (Stage III Behavior)

The SMRN can be important to estimate rates of release of radionuclides from glass wasteforms. As the glass reacts, chemical constituents may increase in concentration in solution, which can lead to saturation with subsequent precipitation of solid phases that form at the surface of the glass. The precipitated phases are termed the SMRN. The processes can be important because they can determine if conditions will be achieved that lead to Stage III (accelerated release) behavior. Stage III behavior in glass corrosion or degradation is believed to occur late in the reaction sequence by the formation of zeolites and other phases that deplete silicon and other species in solution faster than the rate at which those species are added to solution. The phenomenon is not well-understood but has been associated with higher temperatures and closed systems. DOE stated that at the low temperature anticipated for the IDF (15°C) and based on similar low-temperature weathering of natural glasses, these precipitated phases may be poorly crystalline or amorphous. It is also possible that contaminants could become incorporated into precipitated solids and be solubility limited.

In the base case, DOE selected a SMRN that was believed to be consistent with solution chemistry data from Product Consistency Tests at 90°C (PNNL-20781, 2011). Scanning electron microscopy and X-ray diffraction data indicated that many other minerals that were not included in this network can form during glass corrosion (PNNL-20781, 2011; PNNL-24615, 2015). DOE stated that these minerals, which typically include a variety of clay minerals and zeolites (also seen as weathering products of natural glasses (Dibble and Tiller, 1981)), often have uncertain compositions and crystallinity. In some cases, the minerals and zeolites have unknown thermodynamic properties such as standard Gibbs free energies and enthalpies of formation, standard entropies, and isobaric heat capacities. DOE noted an additional difficulty in representing clay minerals and zeolites in reaction networks for VLAW was that they commonly exist as solid solutions rather than phases having a fixed stoichiometry. Reliable models and data that account for solid-solution behavior as a function of changes in aqueous-solution chemistry may not be available for secondary minerals resulting from glass corrosion (PNNL-24615, 2015).

Though DOE did not expect Stage III behavior to occur within the IDF, they examined sensitivity cases (termed SMRN) to examine the impact of the formation of alternative secondary minerals on fractional release rates. These calculations were performed for each of the three glass compositions. DOE used Geochemist’s Workbench to first identify the mineral phase at the end of the reference case simulation for each glass composition that was most super-saturated. This mineral phase was then added to the reaction network and the process repeated. The process was continued until no minerals were super-saturated. This evaluation was limited in scope to only those minerals observed in laboratory testing and for which thermodynamic data was available in the database used. DOE indicated that minerals that could conceivably generate Stage III glass corrosion were not evaluated because such behavior is considered unlikely under IDF-relevant conditions (PNNL-24615, 2015). For the three glass compositions examined, DOE observed fractional release rates that increased less than an order of magnitude. DOE concluded that the preliminary results indicated that fractional release rates are unlikely to be significantly impacted.

In its RAI, the NRC staff asked for additional information associated with the potential to form conditions amenable to Stage III behavior (NRC, 2020a). Staff stated that the presence of an intact GCL layer and liner may result in conditions that are more representative of a closed system. In addition, the composition of the water interacting with the glass will have been modified by the layers of material it flowed through prior to reaching the glass. Because the evaluation of the impacts of SMRN uncertainty completed by DOE eliminated minerals that could conceivably generate Stage III behavior, the probability of occurrence and the impacts were not adequately assessed. In response, DOE indicated that they expect the disposal system to be open but acknowledged the uncertainty (DOE, 2021c). DOE stated that the potential for Stage III behavior would continue to be investigated as part of PA maintenance activities. Those activities would likely include laboratory tests and a field lysimeter test. Information is expected to be generated by the field lysimeter test to support the modeling completed in the IDF PA analyses. Figure 4-9 provides the images of DOE's field lysimeter test that will be used to investigate and verify field-scale performance. The facility has a series of field lysimeter tubes (most extend to a depth of 3 m (10 ft)) that can be filled with various wasteforms, including glass and cementitious materials, and back filled with soils. Water infiltration rate is controlled within each lysimeter tube. DOE stated that water samples will be collected at various depths within the lysimeters over the next several years, along with measurements of the in-situ distribution of moisture. Modeling has been completed to estimate wasteform performance prior to the tests and will be compared to information generated by the tests.

4.6.1.3 Retention and Recycle of Volatile Species

The retention of volatile species in the glass wasteform is a key technical issue for waste disposal at the IDF. The waste contains some species (e.g., ^{99}Tc, ^{129}I, ^{137}Cs) that volatize under the high temperatures used to produce glass. In the IDF PA, DOE examined different inventory cases to account for how much of the volatile species would be retained in glass compared to disposition in other (primarily cementitious) wasteforms (RPP-RPT-59958, 2018). The quantity of volatile species retained in different wasteforms was found to be the most significant uncertainty.

The treatment process developed by DOE will employ tank side ion exchange to remove a large fraction of the ^{137}Cs. The volatilization of cesium is not anticipated to be a concern. The species ^{99}Tc and ^{129}I will not be removed by ion exchange and will be subject to volatilization during glass production. These radionuclides are also key contributors to risk for offsite receptors. Glass is produced in a melter using a high temperature process. Glass frit, other additives and components, and waste are combined in the melter. Molten glass exits the bottom of the melter and is poured into stainless-steel canisters. The stainless-steel canisters cool and are welded shut.

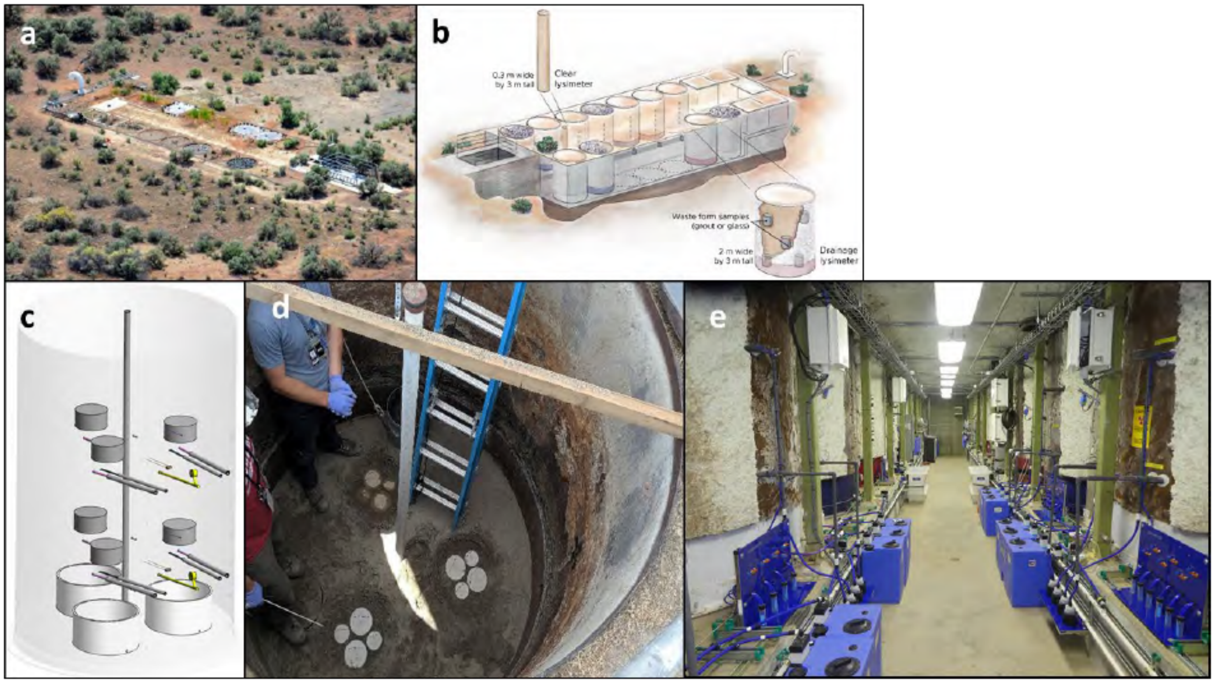


Figure 4-9 DOE's Field Lysimeter Test to Evaluate Wasteform Performance
[Figure 2-7-8 from (DOE, 2021c)]

DOE used measurements in the laboratory setting as well as from testing of engineered systems with differing scales to estimate the amount of ^{99}Tc and ^{129}I that would be retained in glass. DOE observed early in the development process that single-pass retention of volatile species could be significantly lower than if recycling of the off-gas was used. Initial single-pass retention values for ^{99}Tc and ^{129}I were generally less than 50% (Pegg, 2015). Through testing and research, DOE was able to improve these single-pass retention values. Formation of a cold cap on the glass melt was observed to improve capture of ^{99}Tc in the glass. DOE previously found that the addition of reductants to make Tc less volatile could also improve retention (UCRL-53440, 1983).

In response to the NRC staff's RAI, DOE indicated that during vitrification of the LAW, some radionuclides, including ^{99}Tc and ^{129}I, will volatize (DOE, 2021c). DOE stated that the LAW Vitrification Facility will, by design, maximize the capture of the volatized ^{99}Tc and ^{129}I into the VLAW. DOE explained that since the completion of the IDF PA, the latest flowsheet modeling shows that approximately 98 percent of the ^{99}Tc and approximately 96 percent of the ^{129}I would be captured into the VLAW, and approximately 99 percent of all radioactivity in the pretreated LAW will be incorporated into the VLAW (DOE, 2021c).

DOE also indicated, in response to the NRC staff's RAI, that the scaled melter tests with simulants are indicative of the efficiency and effectiveness that could be expected from the full-scale production system (DOE, 2021c). Those studies provided valuable insight into specific aspects of volatile constituent behavior and performance of the off-gas system and its components. With proper consideration of the test system's limitations and realistic interpretation of the results, the data support the expectation that the full-scale production system will function as designed and will be capable of achieving very high recycle efficiency and incorporation of volatile constituents into the glass.

DOE estimated that a temperature gradient can persist from the centerline to the walls of the canister as the canister cools. DOE assumed that volatile species would be distributed uniformly throughout the cooled glass. In its RAI, the NRC staff asked for additional information associated with different aspects for the retention of volatile species in glass and how volatile species would be distributed within the glass and canister (NRC, 2020a). The staff cited experimental observations that technetium may deposit on the walls of the canister during cooling, thereby impacting the calculated fractional release rates that assume the contaminants will be uniformly distributed within the wasteform (Kim, 2018). The NRC staff cited a DOE report that noted the concentration of technetium in a sulfate layer that formed on top of the glass melt was up to 50 times higher than expected (RPP-54130, 2012). In response to the RAI, DOE provided information developed in laboratory tests comparing poured glass samples to dip samples (VSL-11R2260-1, 2011). DOE cited the results of dip measurements that were taken in DM10 and DM100 (these are different scale representations of the production-scale system) experiments that showed only limited variation. With respect to potential deposition on canister walls, DOE indicated that the container will be maintained under active ventilation during pouring and for several hours after filling. DOE also stressed the importance of the formation of a cold cap and the use of reductants to reduce the volatilization of certain species.

4.6.1.4 Glass and Waste Compositions

DOE's approach to developing glass with sufficient performance characteristics was to empirically determine glass compositions that were tailored to specific waste compositions. DOE stated that a few key constituents in the waste streams can affect processing and performance factors for VLAW glasses even when present in relatively minor concentrations (VSL-17R4330-1, 2017). Sulfur, chlorine, and fluorine, for example, can have significant impacts on glass performance. Although several VLAW glass formulations have been proposed for use in the IDF, for purposes of the IDF PA, DOE focused on three glasses (designated as LAWA44, LAWB45 and LAWC22) for which complete sets of laboratory test data existed. The compositions of the glass formulations were provided (PNNL-24615, 2015). DOE stressed that the glass formulations assessed were not final and research was ongoing.

Over the operational history at Hanford, a diverse set of waste streams have been stored within the tank farm systems. Many of these waste streams have been intermixed and have experienced complex physical, chemical, and thermal processes during the prolonged storage period. Numerous diverse chemicals were used during operations. The operations conducted at Hanford included, but are not limited to, bismuth phosphate production, uranium recovery process, reduction-oxidation, waste fractionation, plutonium-uranium extraction (PUREX), and processes conducted at the Plutonium Finishing Plant. Over 1,800 chemicals have been used including many organic compounds. The bulk phases in the tank waste are a variety of salts, oxides, and other materials distributed in solid and liquid phases. For regulatory compliance, emphasis was placed on radionuclides and chemical species such as mercury, chromium, and nitrate, however, wasteform performance is likely to be influenced more by the mineral and chemical compounds present as well as, in the case of glass, minor species that are present in the waste. Recycling of off-gas is used to retain volatile species such as iodine and technetium. Species that are present in the off-gas stream that are not removed by one of the unit operations will build up in concentration over time and may impact performance. These species can include but are not limited to F^-, Cl^-, and SO_4^{2-}.

Because of the diverse operations that have been used over a long timeframe (~ 80 years) at Hanford, the waste composition has significant variability. To reduce the impact of waste variability on glass or non-glass wasteform performance, DOE proposed to use tanks AP-107 and AP-106 to receive and pretreat waste. After waste is transferred to Tank AP-107, it is sent to the TSCR facility to remove ^{137}Cs. Pretreated waste will then be stored in Tank AP-106 (capacity of 4.2 million L (1.1 million gallons)) before being transferred in batches with a volume of approximately 34,000 L (9,000 gallons) to the vitrification facility. DOE identified areas of uncertainty associated with the inventory to include waste phase partitioning fractions, distribution of the key radionuclides and chemicals, operational decisions, and overall flowsheet configuration.

DOE's approach for ensuring the performance of the glass wasteform relies on laboratory testing and process controls to ensure the engineered systems are operated within established ranges. DOE has no plans to verify the performance of the production-scale glass. DOE stated it is not practical or necessary to verify the performance characteristics of VLAW glass made from real waste at the production scale through testing of actual VLAW glass samples. Glass durability as determined via methods such as the Product Consistency Test (ASTM C1285) and Vapor-phase Hydration Test (ASTM C1663) has been correlated with glass composition to demonstrate compliance with Waste Acceptance Product Specifications for HLW and WTP contract specifications for VLAW glass (PNNL-22631, 2013; PNNL-25835, 2016). The detailed strategies for demonstrating compliance with glass durability requirements for DOE vitrification facilities are described in the wasteform compliance plans for those facilities. The data and test results demonstrating compliance are found in the wasteform qualification reports for those facilities. DOE stated that all rely on laboratory-scale testing of glasses made from simulated waste to cover a range of compositions of glasses to be made from real waste. The composition of nuclear waste glass during actual production operations is then controlled within the pre-qualified glass composition region and the performance of glass produced can be predicted from models that correlate glass performance with glass composition, after accounting for uncertainties in composition and performance (PNNL-22631, 2013; PNNL-25835, 2016).

DOE indicated that data have been collected demonstrating the relationships between glass composition and glass properties and performance for nuclear waste borosilicate glasses. There are numerous reports of tests and demonstrations of VLAW glass production with simulated waste at scales ranging from crucible scale (hundreds of grams) to a 1/3 scale pilot melter system producing greater than 5 metric tons (11,000 lbs) of glass per day. Properties of glass made from real waste at the crucible scale were found to be in excellent agreement with VLAW glass made from simulated waste in melter runs at different scales (PNNL-13372, 2000).

4.6.1.5 Modeling of Glass Performance

DOE stated it is important to demonstrate a sufficient understanding of the factors that influence releases from glass wasteforms to support a reasonable expectation that release rates will not result in exceeding the established performance objectives. DOE used a reactive-chemical transport-modeling framework. The fluid chemistry interacting with the waste was coupled with kinetic rate equations that describe the response of the glass dissolution rate to changes in liquid composition in the disposal facility. These kinetic rate equations assume that the driving force for transforming unstable silicate materials into stable ones is governed principally by the magnitude of displacement from thermodynamic equilibrium. DOE used laboratory testing to provide the key input data required to assess the long-term performance of VLAW glasses. Four principal experimental methods have been used to identify glass corrosion rates.

Glass-water interactions entail complex sets of coupled physicochemical mechanisms and a single equation or model that fully addresses the complexities and reflects observed behaviors has not yet been developed. A mathematical model that describes glass reactivity is needed to predict the long-term fate of glass in the subsurface over the period of regulatory concern. The DOE modeling of glass release was based on the Transition-State Theory (TST) of chemical kinetics (PNNL-11834, 1998). The reaction rate is governed by the slowest elementary reaction. Waste glasses are a complex system, but the assumption made was that similar processes define their degradation. Other models of glass degradation continue to be developed (Pierce et al., 2014). DOE stated that the TST rate law best described the experimental data collected over 35 years of glass/water reaction studies (PNNL-11834, 2000). Therefore, the technical approach implemented was a robust strategy to model the key processes that control long-term VLAW glass corrosion.

Laboratory testing was used to assess the temperature and pH dependency of the rate-controlling parameters (PNNL-11834, 1998). The assessment was completed on representative VLAW glasses based on projections of glass compositions. Glasses were formulated for each of the original A, B, and C operating envelopes. The dissolution rate equation is given below. The dissolution rate, r_{diss}, is a function of the surface area (A expressed in m^2) and a coefficient k expressed in mol/m^2-s.

$$r_{diss} = kA$$

The coefficient k was represented by:

$$k = \vec{k} a_{H^+}^{-\eta} exp\left(\frac{-E_a}{RT}\right)\left[1 - \left(\frac{Q}{K_g}\right)^{\sigma}\right]$$

The dissolution rate parameters were derived from tests completed on different glass formulations and were given as:

k_o = dilute rate coefficient
η = pH power law coefficient
E_a = dilute rate dissolution activation energy
K_g = pseudo-equilibrium constant
r_{iex} = ion exchange rate
σ = Temkin coefficient (assumed to be 1)

DOE derived the parameters for the rate expression from the experimental data. The experimental data were summarized in technical reports. Figure 4-10 is an image of the relevant data provided (PNNL-24615, 2015). DOE provided uncertainty estimates for some of the parameters while others were fixed at constant values for a given glass formulation. The dissolution rate expression was implemented in the PA GoldSim model to translate dependent

	Parameters							
	$\bar{k}_0$		K_g [(a)]	η	E_a	σ	r_{IEX}	
Glass	Reported Forward Rate Constant (g/[m² d])	Converted[(b)] Forward Rate Constant (mol/[m² 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/[m² s])	Reference
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 al. (1997)
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}	McGrail et al. (2001a)
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 al. (2004)
LAWB45	1.6×10^4 ($R^2 = 0.90$)	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 al. (2004)
LAWC22	1.0×10^5 ($R^2 = 0.98$)	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 al. (2004)

Figure 4-10 DOE's Derived Parameters for the Glass Dissolution Expression
[Data from (PNNL-24615, 2015)]

variables in the PA to a fractional release rate from the glass. For the approach to be valid, the suite of tests used, the developed parameters, and the rate model must be applicable for the disposal environment. To estimate the fractional release rate that would result from different disposal conditions (e.g., moisture flow, geochemical conditions), DOE performed modeling with the process models STOMP and Geochemists Workbench®. Both codes were approved, managed, and used in compliance with DOE requirements. DOE provided a detailed summary of the parameters used in the glass release modeling completed with STOMP and Geochemists Workbench® in Table 4.8 (4 pages) of the IDF PA (RPP-RPT-59958, 2018). The modeling was described in more detail in the report RPP-CALC-61031 (RPP-CALC-61031, 2017). DOE applied the concept of abstraction in the application of glass release modeling. Numerical process models with detailed multi-physics were used to develop computationally efficient expressions for fractional release rates that were then implemented in the PA model. Modeling was completed by properly trained and qualified analysts. Verified software was used and the results were checked by independent reviewers. Model input and outputs were archived to maintain transparency and traceability. Test cases were used to perform code verification.

The modeling used a variable grid spacing of 10 by 17 cm (4 by 7 in) at the smallest to 18 by 20 cm (7 by 8 in) at the largest, however, DOE examined grid spacing as small as 2 by 2 cm (0.8 by 0.8 in). Execution times exceeded 6 weeks for the finer grid and were viewed as being impractical. Estimated release rates were 51 percent higher for the refined grid. NRC staff asked for additional information on the sufficiency of grid spacing (NRC, 2020a). During discussion on this topic after DOE's response to the NRC staff's RAI, DOE staff indicated that recent simulations had been completed with a 1 cm (0.4 in) grid spacing and the calculated fractional release rates were an additional 20 percent higher.

4.6.2 NRC Evaluation of DOE's Analyses of Glass Wasteform Performance

Engineered wasteforms are one of the most risk-significant barriers used to mitigate potential radiological doses from radioactive waste disposal. Properly designed and implemented wasteforms can result in a risk reduction of many orders of magnitude compared to not using a wasteform. Because the wasteforms are designed (man-made), uncertainties can be better controlled in comparison to the natural system (e.g., retardation, dispersion, dilution). In the IDF, wasteforms provide a stable media to limit void formation after closure. The wasteforms will likely have acceptable compressive strength and mechanical properties to support waste emplacement and limit future subsidence. The wasteforms will also have mechanical properties that limit the dispersibility of the waste upon potential disturbance. The following sections describe the NRC staff's evaluation of wasteform performance with an emphasis on retention of radioactivity and potential release to the environment.

The NRC staff focused much of the review effort on the glass wasteform because DOE anticipated that as much as 90 percent of the waste (by volume), and potentially more by activity, would be in the glass wasteform. The NRC staff reviewed numerous technical reports in formulating its conclusions on DOE's analyses of glass wasteform performance.

The conceptual approach used by DOE accounted for relevant FEPs that the NRC staff expects would determine glass wasteform performance. The experimental basis used to develop understanding of the alteration processes and to parameterize the constitutive relationships was appropriate. Many different types of tests and characterization methods were used. The variety of tests and methods used can help identify potential biases with individual techniques or measurements.

4.6.2.1 NRC Evaluation of Glass Cracking

The NRC staff reviewed the information DOE provided associated with the impacts of cracking of the glass wasteform. In the IDF PA, DOE applied a factor of 10 increase to the release of radionuclides from the glass wasteform to account for cracking of the glass. The NRC staff verified in the PA model that the factor was appropriately applied to the specific surface area. The technical basis for the assumed amount of cracking was not apparent to the NRC staff and thus the staff asked RAI 2-8 (NRC, 2020a). In response to the RAI, DOE stated the basis for the factor of 10 was from the report PNNL-13369 (PNNL-13369, 2001). DOE stated that fracturing was expected to increase the glass surface area a maximum of 10X over its geometric surface area. DOE summarized the results of experiments and analyses associated with glass cracking (DOE, 2020a). Some experiments showed increases in surface area that were larger than a factor of 10, however, not all the increase in surface area will contribute to release. Because of physical and chemical limitations, some of the surface area does not contribute to release. DOE described two parameters, α and β, to describe the increase in area and the decrease in contribution of the area associated with fractures in glass.

The NRC staff concurs with DOE's conclusions above with the following caveats. First, the reduction in the contribution of cracking to release (β) may be sensitive to the moisture content of the system when the release occurs. Cracks will have a distribution of apertures and lengths and, under unsaturated conditions, capillarity may influence how much the cracks contribute to release. Experiments that examined this factor should be carefully analyzed to ensure that the conditions would be comparable to projected future disposal conditions. Second, the amount of increase in area (α) may be sensitive to glass composition and is likely sensitive to the projected cooling profile and canister properties. Though the cited observations provide a good summary, there is uncertainty as to whether the amount of cracking for VLAW canisters at Hanford will fall within the observed range. Rapid cooling is good for glass quality but can lead to cracking. Slow cooling can lead to less cracking but may result in crystallization of the glass. It is recommended that the amount of cracking for VLAW glass be determined from analyses of samples of appropriate scale and composition using the cooling profile specification that will be used in the design. (Recommendation #6).

4.6.2.2 NRC Evaluation of Secondary Mineral Reaction Network (Stage III Behavior)

The SMRN can be important to estimated release rates of radionuclides from glass wasteforms. The NRC staff reviewed the information provided by DOE in the IDF PA, supporting reports, and in response to the NRC staff's RAI. Depending on complex processes and different factors that are currently not well-understood, a glass can transition from a slower release state (termed Stage II) to a faster release state (termed Stage III). Stage III behavior in glass corrosion is believed to occur late in the reaction sequence from the formation of zeolites and other phases that deplete silicon and other species in solution faster than adding them to solution.

In the base case PA, DOE did not include adjustments to release rates to account for Stage III behavior because the disposal system is expected to be "open" from a gas and liquid flow and geochemical standpoint and temperatures are anticipated to be around ambient below ground values of 15º C (59º F). Stage III behavior has been associated with closed systems at higher temperatures. In the NRC staff's RAI, the staff indicated that while the temperature of the system should be close to ambient in the absence of degrading organic matter or heat-generating waste, the openness of the system is more uncertain (NRC, 2020a). The impermeable asphalt layer in the engineered cover over the waste combined with the GCL liner under the facility can result in very low flow rates and conditions that could approximate a closed system. In addition, in the simulations DOE performed, the fluid that reacts with the glass was not a fluid that had reacted with the overlying engineered barriers but was a Hanford groundwater composition.

DOE used chalcedony as a kinetic control in the modeling of glass degradation 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 had not been detected as an alteration product of glass corrosion (PNNL-24615, 2015). Chalcedony and the assumed kinetic controls were calibration parameters used by DOE to fit the empirical data. The NRC staff stated that the information DOE provided in the IDF PA and supporting documents did not justify the assumption of the absence of Stage III behavior. DOE completed an assessment of different secondary mineral formation, but DOE did not consider minerals that could conceivably generate Stage III glass corrosion because such behavior was considered unlikely under IDF-relevant conditions (PNNL-24615, 2015).

The NRC staff reviewed the information provided by DOE in response to its RAI (DOE, 2021c). DOE maintained that conditions that would initiate Stage III behavior are not anticipated but recognized the uncertainty. DOE summarized a variety of tasks that are being or will be completed to reduce the uncertainty (DOE, 2021c). Laboratory work is being completed to investigate two questions:

- If the IDF has conditions closer to a closed system rather than an open system, will Stage III occur (e.g., zeolite formation)?
- If Stage III were to occur, would the increase in dissolution rates be high enough to cause release of contaminants to exceed the IDF performance objectives?

DOE summarized the results of long-term tests (up to ten years) that have been completed. Stage III rate acceleration was observed at temperatures at or above 90°C for samples with high surface-area-to-liquid volume ratios. Limited testing had been completed at lower temperatures that would be representative of IDF conditions. Data from 40°C showed less alteration and has not shown an acceleration of dissolution rates that would be observed with Stage III behavior. At the higher temperature, results for ten Hanford VLAW glasses (and a reference glass ANL-LRM-2) incubated in water for ten years clearly demonstrated several of the glasses had accelerated dissolution rates indicating Stage III behavior.

The triggers for Stage III behavior and the propensity of various compositions to exhibit such behavior are poorly understood, particularly in the relatively low temperature conditions expected in the IDF. Additional research is necessary to establish criteria to inform whether a particular composition/conditions set is likely to lead to Stage III behavior. Experimental work proposed by DOE will help to address the uncertainty associated with determining the likelihood and impacts of Stage III behavior. While the current data indicate that Stage III behavior is less likely at lower temperatures, DOE should use caution in interpreting the data. The likelihood of occurrence could be the same, but the kinetics are slowed at the lower temperatures. In other words, if given enough time, the samples at lower temperature could experience Stage III behavior at the same frequency as the samples at higher temperature. The wasteform glasses must perform for 1,000 years and longer. A long-term experiment of ten years is a small fraction of the desired period of performance. However, DOE should be commended for committing to long-term experiments because they can be difficult to financially support and technically justify if “changes” are not being observed. The experiments will likely yield valuable insights with respect to the potential long-term performance of the wasteforms.

DOE summarized the results (some preliminary) of zeolite seeding tests to address the second question (what happens if Stage III occurs) (DOE, 2021c). In these tests, mineral phases that are believed to lead to Stage III behavior were purposely introduced. DOE also summarized the plans for future (or ongoing) tests designed to cover the full range of anticipated glass compositions over a range of temperatures. DOE will examine the data to assess the SMRN that was assumed in the IDF PA. The ratio of the preliminary predicted dissolution rates for seeded to unseeded tests (i.e., [seeded Stage III rate]/[rate prior to seeding]) at 15°C ranged from 0.9 to 470 with a geometric mean of 5.7 (PNNL-28898, 2019). DOE stated that the IDF PA results indicate that a glass dissolution rate increase of less than approximately 9 times over the nominal Stage II glass dissolution rate would not result in contaminant releases exceeding regulatory limits at the point of compliance (RPP-CALC-63176, 2020). Eighteen of the 24 glasses tested had ratios at, or below, nine times and only one glass had a ratio greater than 11 (PNNL-28898, 2019).

The NRC staff strongly supports more work on the zeolite seeding experiments. These experiments provide valuable information to address the third component of the risk triplet – “what are the consequences?”. DOE appropriately understood the uncertainties associated with Stage III behavior. However, the results of the DOE analyses may not be as useful without the second component of the risk triplet – “how likely is it?”, which is a more difficult question to answer given kinetics, reaction paths, and thermodynamic data uncertainty. The experimental results completed to date show that the expected increase (5.7) would be less than the magnitude needed (9) to exceed the performance objectives. The results showed that there can be substantially different relative performance depending on glass composition (range of 0.9 to 470). The overall performance impact may be sensitive to how much of the glass that is produced performs at a poor level (if any) and how that wasteform is distributed within the IDF.

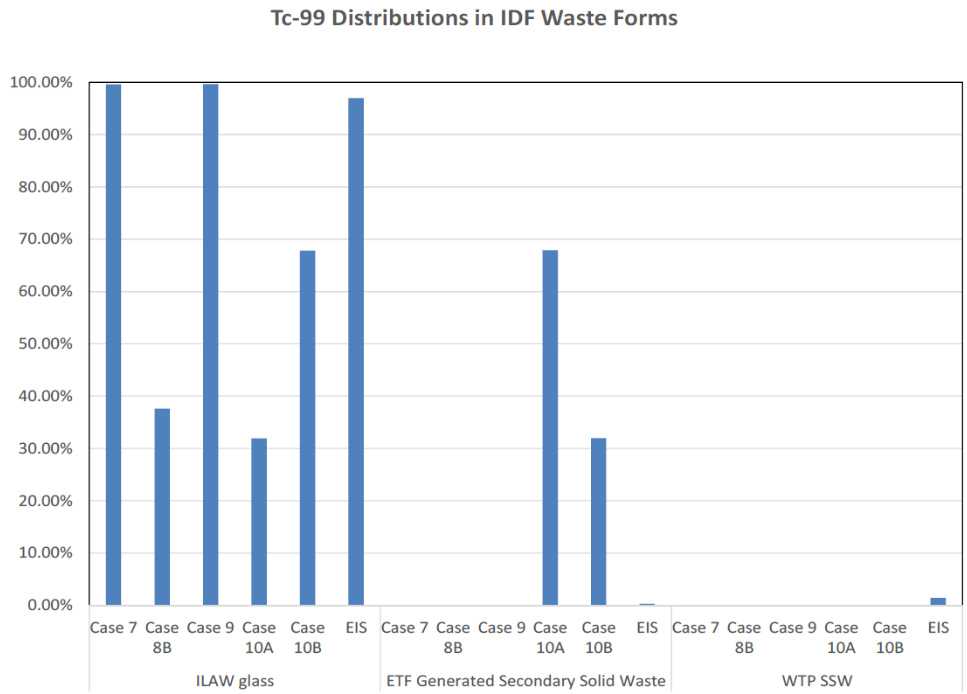
The NRC staff does not agree that a value of nine can be used to assess the threshold for significance of Stage III behavior. That factor is dependent on all other things remaining the same and no other uncertainties being addressed in a similar manner. As discussed in Section 4.6.1.5, DOE examined the impact of discretization in glass release modeling and concluded that increases of 40-50% would not be significant from a performance standpoint. The discretization effect represents a bias rather than an uncertainty. Application of the increase to the factor of 9 would decrease it to around 5. DOE should consider all uncertainties and biases that are addressed using risk arguments when drawing conclusions about their significance. If numerous uncertainties are addressed in this manner, a probabilistic evaluation should be used. The likelihood of combinations of conditions or uncertainties can be determined (e.g., not all good or bad conditions are likely to occur at the same time).

4.6.2.3 NRC Evaluation of Retention and Recycle of Volatile Species

DOE identified the amount of volatile species that partition between different wasteforms (i.e., glass or non-glass) as a key uncertainty. The NRC staff agrees that the partitioning of volatile species between wasteform types is a key uncertainty. In the IDF PA, DOE evaluated different inventory cases. Figure 4-11 shows the disposition of two volatile radionuclides (^{99}Tc and ^{129}I) for the different inventory cases. DOE stated that the inventory cases produced small doses in the 1,000-year compliance period (well below the performance objective). However, peak doses from options 10a and 10b were much higher than other alternatives and exceeded the performance objective in the sensitivity and uncertainty analysis period. Doses from Case 7b were also higher than the remaining cases and were close to the 0.25 mSv/yr (25 mrem/yr) protection of the public performance objective in the sensitivity and uncertainty analysis period.

DOE's evaluation of timeframes under DOE Order 435.1 includes a compliance period of 1,000 years. The 0.25 mSv/yr (25 mrem/yr) dose limit does not apply after this period and is only referenced for comparison purposes. Doses are low in the compliance period and higher in the uncertainty and sensitivity analyses period because of slow transport through the unsaturated zone. The main differences between the wasteform partitioning cases were the use of recycling of the off-gas and the values assumed for single-pass retention of volatile species.

During vitrification of the LAW, some radionuclides, including ^{99}Tc and ^{129}I, will volatize. DOE indicated that the LAW Vitrification Facility will, by design, maximize the capture of the volatized ^{99}Tc and ^{129}I into VLAW. NRC asked for additional information to support the percentage of ^{99}Tc and ^{129}I incorporated into glass. In response to the NRC RAI, DOE explained that updated flowsheet analyses showed that approximately 98 percent of the ^{99}Tc and approximately 96 percent of the ^{129}I will be captured into the VLAW, and approximately 99 percent of all radioactivity in the pretreated LAW will be incorporated into the VLAW (DOE, 2021c). Most of the radionuclides in the LAW, including ^{99}Tc and ^{129}I, will be captured in the VLAW and will not be entrained in secondary waste. The NRC staff reviewed the information provided by DOE and requested various additional references (RPP-RPT-57991, 2019; MR-50461-00, 2019; 24590-WTP-ES-PE-19-001, 2019). The transparency and traceability of this key information was lacking. The increased partitioning to glass was based on assumed increases in the single-pass retention rates, but staff was unable to verify the basis or how the steady-state retention values were derived. The assumed increases in single-pass retention rates are very uncertain.



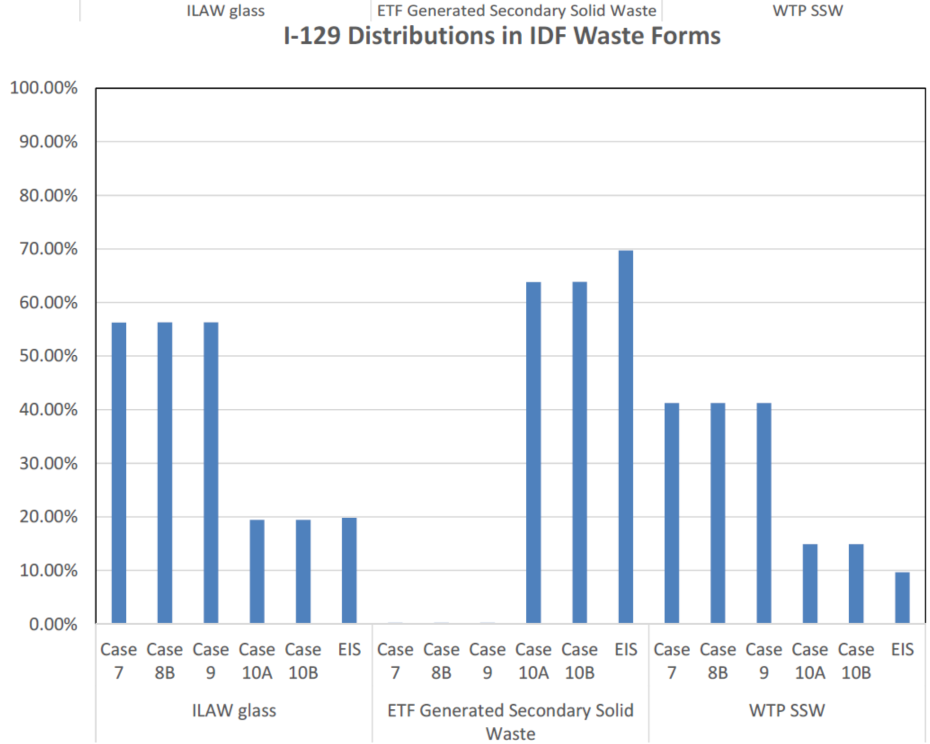


Figure 4-11 Partitioning of Select Volatile Radionuclides Among Wasteforms for Different Inventory Cases
[Figure 3-119 and 3-121 of (RPP-RPT-59958, 2018)]
Note: DOE uses the terms "ILAW Glass" and "VLAW" interchangeably

In the NRC staff's RAI, staff summarized some experimental results on the retention of volatile radionuclides using recycle (NRC, 2020a). Challenges included achieving mass balance and holdup of volatile species in engineered systems. If holdup is significant, it can be managed by proper flushing of components. At the end of operation of the system, flushing cannot be used effectively because there will be no more glass being produced to flush the off-gas system into. In response to the RAI, DOE indicated that work is continuing to improve the mass balance in the experimental measurements (DOE, 2021c).

A reference provided in response to the NRC staff's RAI for the increased retention of ^{129}I (24590-WTP-ES-PE-19-001, 2019) has updated information. The key decontamination factors (DF)s in the DOE approach are associated with the submerged bed scrubber (SBS) and the wet electrostatic precipitator (WESP). DOE performed an analysis of the available data and applied various screening criteria to ensure the information was applicable. The result was that the information to support the retention of ^{129}I was sparse. Of the 64 tests considered, only three were deemed to be applicable to provide information for the WESP and SBS DFs. Many uncertainties apply to the information and analysis including:

- DOE used a power law scaling relationship to convert the data for application to the VLAW system. Since data from only one sized system (the DM1200) was used for the WESP and SBS decontamination factors, this scaling relationship may not be valid.
- DOE assumed that DFs of components are independent and can be multiplied. For some systems and over some concentration ranges, the DFs may be independent but for others it may not be. The information is too limited to support the assumption of independence.
- The results show strong sensitivity to feed type (Envelope A, B, C). The three envelopes tested do not encompass the full range of feed variability associated with volatile species and their retention.

DOE used a variety of different engineered systems to perform tests and design the off-gas system. The testing completed and research performed was of high quality. However, most of these systems differed in the scale and components present from what will be used to produce LAW glass. As stated above, the data from only one system (the DM1200) were used to supply key decontamination factors for ^{129}I. This system was not operated in the exact way the off-gas system for VLAW will be operated nor were some of the components scaled in the exact ratios. There is uncertainty associated with the representativeness of the limited data that has been collected in addition to the uncertainties listed above. If the glass wasteform from producing actual waste is not verified, DOE may not know if the assumed volatile species retention values are being achieved.

The NRC staff also reviewed a reference from the Savannah River Site (SRS) where the authors provided an independent assessment of recycle of volatile species in off-gas during glass production at Hanford (SRNL-L3300-2020-00019, 2020)[16]. That report raised technical

[16] Many references cited in the SRS report and in the Hanford Site reports given above were not available to the NRC staff. DOE worked with the NRC staff to provide additional references when requested, however, that process to request and receive references cleared for release is not efficient especially when the source reference may be at the end of a long, linked collection of references. NRC and DOE should work together to identify possible improvements to the process for providing references.

issues with the information collected to date and the cumulative DFs put forth at the Hanford Site and in response to the NRC staff's RAI. When aggregated, those technical issues could result in markedly different DFs for ^{129}I retention. The SRS reference generated a cumulative DF of 4.4 compared to 43.2 for Bechtel National, Inc.

The NRC staff recommends that DOE improve the basis for the retention of volatile species in VLAW glass. Base case analyses should use lower (more conservative) values for retention of volatile species in glass until additional information is developed to support higher retention values (Recommendation #7). Staff acknowledges the research that DOE has done on use of a cold cap and reductants to improve the retention of volatile species. There are many uncertainties, the representative testing is limited, and there are no plans to verify what will be achieved on the production scale. Based on information to date, DOE has not demonstrated the high retention values of 98 percent for ^{99}Tc and 96 percent for ^{129}I. It is recommended that DOE use more conservative values in the base case evaluation until additional information is developed to support higher retention values.

The distribution of volatile species within cooled glass wasteforms and canisters is a risk-significant uncertainty. The NRC staff did not identify information where DOE investigated the potential for deposition of volatile species on cooled surfaces. DOE cited the results of dip samples that demonstrated that the quantity of volatile species in molten glass was similar to that observed in hardened glass. While that is an important observation, it does not fully address the issue of where the volatized species deposit during the cooling process and if the concentrations in the product are uniform (as assumed in the IDF PA). If the annual fractional release rate from the glass product is 1×10^{-7} to 1×10^{-8}, then the deposition of only fractions of a percent of technetium or iodine at the periphery of the wasteform or within the headspace of the canister (or within the confines of the building/cell housing the melter) could be risk-significant.

4.6.2.4 NRC Evaluation of Glass and Waste Compositions

The NRC staff reviewed DOE's approach for ensuring performance of the glass wasteform and the basis for reliance on empirical data. The glass compositions considered were clearly described. The use of the large storage tanks to mix waste and then submit batches to the glass production facility should help to reduce variability in waste compositions. There is a large volume of waste that must be processed and wastes of many different types such that variability is inevitable. It is likely that unforeseen production issues may arise from this variability. DOE should have clear procedures in place to identify when operational changes are necessary to avoid compromising the durability of the glass.

DOE has no plan to verify the performance of the production-scale glass. Though staff recognizes the substantial body of work that DOE has developed on glass performance, staff does not agree with DOE that it is unnecessary to characterize the production-scale glass. The glass production for VLAW at the Hanford Site is unique. DOE experience at other sites is valuable, however, aspects of the Hanford Site waste and glass processing may be sufficiently different and may result in performance issues. DOE needs to address the following issues in production-scale glass:

- Performance of glass when using extensive recycle
- The variability in major and minor chemical species in the waste

- The distribution of volatile species within the waste container
- The amount of cracking

To increase the retention of volatile species in glass, DOE plans to use extensive recycling of the off-gas. Though the knowledge of the glass scientists at the Hanford Site is extensive, the combinations of the waste variability with hundreds to thousands of different compounds, glass formulations, and processing steps and unit operations (recycle) is a high order of complexity. It is beyond scientific capability to reliably predict the cause and effects that will occur in such systems.

DOE's approach is dependent on empirical information. For that approach to be successful, the data must be representative and account for uncertainties. DOE cited the consistency of empirical data that has been collected with systems of different scales (DOE, 2021c). The NRC staff reviewed that information. It appears that very little of the existing empirical information (leaching data) developed to establish glass performance used glass produced with the complete set of the unit operations (recycle) that will be used in the production-scale system. Recycling of off-gases will concentrate minor species if they are not removed in unit operations downstream of the melter before the off-gas is returned to the melter. For example, sulfate salt phases were observed on the melt pool surface after two tests using scaled melter systems in the laboratory (RPP-54130, 2012). The sulfate salt phase showed an approximate fifty-fold enrichment in technetium over the bulk glass. In some experiments, volatile species have been observed to not be uniformly distributed between the glass and disposal container/containment for the experiment.

On an annual basis, fractional release rates from the glass are on the order of 1×10^{-7} or $1/10^{th}$ of a part per million. If only 0.01% of the volatile species were deposited on the canister walls, headspace, or within phases located at the geometric boundary of the wasteform, that amount of material available for rapid release would be 1,000 times larger than considered in the current release modeling. It has been observed in different wasteforms that there is a rapid release fraction (e.g., some cements, spent nuclear fuel). The PA results will likely be sensitive to a rapid release fraction from the glass. At the SRS, in line samplers were used to collect samples of actual production-scale glass for the development of glass as a wasteform for HLW disposal. The technology exists and could be deployed at the Hanford Site. However, those samples were not directed at measuring the distribution of volatile species in the disposal container or the impacts on glass durability produced with significant recycling of the off-gas.

DOE considered variability in major chemical components of the waste when performing durability testing of the glass and developing glass formulations. However, the NRC staff note that the range of different waste types is broader than DOE considered in the testing, with many more minor species. DOE stated that glass formulation development is ongoing. The NRC staff recommends that DOE compare the range of waste compositions tested to establish glass performance to the wastes present in the tank farms and the wastes expected to exit the pre-treatment tank (AP-106), evaluate whether the previous tests adequately considered the range of compositions expected in the waste stream, and perform additional glass durability tests as needed. It is also recommended that DOE assess the buildup of minor species resulting from recycling and include the resultant phases during glass durability testing (Recommendation #8).

The amount of cracking that will occur for the actual cooling profiles used and the range of glass compositions produced is unknown. DOE has assumed that the empirical data collected for other systems and glasses will be representative for VLAW glass. To proceed to hot commissioning, DOE will produce cold samples. It is recommended that those samples be non-destructively or destructively examined to determine the amount of cracking (Recommendation #9). Though not a perfect substitute, ruthenium could be added as a surrogate for ^{99}Tc and non-radioactive iodine could be added to determine the distribution of volatile species within the cooled disposal container resulting from the cold testing of the production-scale with recycle.

4.6.2.5 NRC Evaluation of Modeling of Glass Performance

The NRC staff performed a risk-informed review of DOE's modeling of glass performance. Because glass comprises a high fraction of the wasteforms that will be generated, and it is anticipated to provide a significant barrier to release of radioactivity to the environment, the staff focused considerable effort on reviewing the data, assumptions, modeling, and results. The NRC staff did not limit the review to top-level references. The DOE approach to modeling glass release and performance can be summarized as one of experimentation, process modeling, abstraction, and system-level modeling with the development of risk insights. The NRC staff concludes that the approach was sound and justified. DOE demonstrated a sufficient understanding of the factors that influence releases from glass wasteforms. The factors that influence the release from glass wasteforms include the composition of the glass, the composition of the fluids that interact with the glass, the temperature of the system, the active surface area, secondary minerals that form, and the kinetics and equilibrium (reaction paths) of the processes.

DOE used reactive-chemical transport modeling to simulate the interaction of fluids with the glass. STOMP and The Geochemist's Workbench® (GWB) were the computational models DOE used to estimate fractional release rates from the glass for a given set of conditions and glass composition. DOE appropriately described the QA processes and procedures that were used to implement the computational models. The NRC staff reviewed the information provided and found that QA was applied and documentation justifying the appropriate use of QA was provided. DOE had a good practice of archiving model input and outputs to maintain transparency and traceability. A key step in the software QA process is to perform verification tests on the software prior to use. DOE performed a set of verification tests to address different capabilities of STOMP, such as flow under unsaturated conditions. A challenge occurs when the code will be used in a multi-physics calculation where many processes occur together. Verification tests that are representative of the applied scenario can be difficult to identify. For these complex cases there are not analytical or other solutions available to compare against. Use of two different computational models, as DOE did in the assessment, was a reasonable approach to help address this challenge. Given similar inputs, the two computational models produced similar results, which helps to address verification. DOE provided evidence of independent review, also a key part of QA applied to modeling.

DOE applied the concept of abstraction in the application of glass release modeling. The NRC staff examined the abstractions developed by DOE and found them to be appropriately representative of the process modeling results. The staff endorses the concept of abstraction, especially when used in a large PA model that may experience slow execution times if burdened with integrating complex process models. Process models with detailed multi-physics were used to develop computationally efficient expressions for fractional release rates that were then implemented in the PA model. Modeling was completed by properly trained and qualified analysts. Verified software was used and the results were checked by independent reviewers. Model input and outputs were archived to maintain transparency and traceability. Test cases were used to perform code verification.

The NRC staff agrees with DOE that glass-water interactions entail complex sets of coupled physicochemical mechanisms and a single equation or model that fully addresses the complexities and reflects observed behaviors has not yet been developed. The DOE modeling of glass release was based on the TST of chemical kinetics (PNNL-11834, 1998). DOE stated that the TST rate law best described the experimental data collected over 35 years of glass/water reaction studies (PNNL-11834, 2000). The NRC staff reviewed the information provided and the basis for DOE's conclusion. Based on currently available information, the TST model appears to be sufficient for the intended purpose of estimating the future releases from glass wasteforms. That is not meant to imply that future research in this area should not be completed or that the science has been finalized; rather that the modeling yields results that are consistent with the experimental results. As DOE indicated, other models of glass degradation continue to be developed (Pierce et al., 2014). Because the systems are very complex, it would be prudent to remain apprised of future research and if necessary, modify the approach.

DOE used laboratory testing to provide the key input data required to assess the long-term performance of VLAW glasses. Four principal experimental methods were used to identify glass corrosion rates (PNNL-11834, 1998; PNNL-23503, 2014). Those methods provide data that is useful to perform the glass release modeling. Laboratory testing was used to assess the temperature and pH dependency of the rate-controlling parameters (PNNL-11834, 1998) for different glass compositions (A, B, C). Parameterization of the glass release expression required interpretation and analyses of many different test results. DOE clearly described how parameters were developed, what reports were used, and the NRC staff was able to trace the flow of information (e.g., from underlying references to PNNL-24615 to the PA).

In the NRC staff's RAI, staff asked questions about the development of parameters for the release rate expressions from the experimental data (NRC, 2020a). The NRC staff indicated that the derived parameters did not reflect some potential sources of uncertainty. In response, DOE described different activities that are ongoing or future work to improve the representation of uncertainty in glass dissolution rates (DOE, 2021c). DOE described principal component analyses that was completed to examine potential correlation in dilute rate parameters. A new technique, Stirred Reactor Coupon Analysis (SRCA), will be used that will allow the analyses of a greater number of samples. The NRC staff identified some important parameters that were fixed as constants in the DOE rate expression (NRC, 2020a). DOE is currently investigating additional methods to determine if some of those parameters should no longer be fixed as constants or if they should be fixed at different values. DOE cited work ongoing since FY 2017 to gather information on the time, temperature, and pH dependence of alkali ion exchange in VLAW glasses to better represent the ion-exchange process in the rate model (PNNL-26594, 2017). The current and future work DOE described should provide appropriate information to address uncertainties in the glass release expressions. DOE has properly accounted for these uncertainties in the PA maintenance program.

DOE described the discretization that was used for modeling glass release. The discretization was coarse in comparison to the phenomena that may occur at the surface of the glass during glass degradation and release. When DOE refined the numerical grid to be less coarse, there was an estimated 51 percent increase in the fractional release rate. The more refined numerical grid took much longer to simulate. DOE indicated that a 51 percent increase was insignificant in the context of a PA that has many uncertainties. The NRC staff does not concur with DOE's conclusion. This type of result, increasingly larger estimated releases with smaller grid spacing, represents a source of numerical bias rather than a sensitivity. Sensitivities are many times balanced such that the mean or median result are not significantly impacted. If the PA modeling has many parameters or phenomenon that are uncertain, then sensitivity analyses performed from a biased starting point may not lead to effective decision-making. A grid spacing of 1 cm (0.4 in) is very large in relation to the phenomenon driving glass release which occurs in surface layers that may be microns thick. The NRC staff recommends DOE perform additional analyses to investigate the potential bias in results from using a coarse numerical grid (Recommendation #10).

DOE estimated that 373 m^3 (13,200 ft^3) of waste would be associated with 18 melters that are projected to be used (spent) or fail during the vitrification of LAW. These melters will likely be filled with grout to eliminate void space and be placed in large carbon steel overpacks for transportation and disposal. The filled overpacks will each weigh 336,000 kg (740,000 lb). A melter may fail at any time during processing, leaving some partially or unmolten glass as well as waste that may not have been completely stabilized. The high pH of grout used to fill the void space may also impact release rates. Deposition of volatile radionuclide on spent/failed melter surfaces may also occur. These processes may contribute to releases from spent or failed melters being significantly higher than glass wasteforms. However, the volume of glass associated with the spent or failed melters is only slightly more than 0.1 percent of the total volume. DOE modeled the release from spent or failed melters as being the same as the glass wasteform. For completeness, the release from spent or failed melters should be modeled with different parameters from the glass wasteform.

4.6.3 NRC Conclusions on Glass Wasteform Performance

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on wasteform performance. The NRC staff has the following conclusions with respect to glass wasteform performance:

- The glass wasteform will provide a significant barrier to the release of radioactivity contained in the glass. The glass wasteform will help limit the impacts from disposing of radioactive waste in the IDF.
- Significant research has been completed to provide technical basis and understanding of the performance of the glass wasteform.
- Though not part of the glass wasteform, DOE elected to not take credit for the stainless-steel disposal containers. In the subsurface environment at the Hanford Site, stainless steel is likely to provide a substantial and temporally durable barrier to release.
- DOE has not demonstrated the high retention of volatile species in glass (98% for ^{99}Tc and 96% for ^{129}I).
- The DOE analyses of glass performance were clearly described, transparent, and traceable.
- The use of the TST model for glass release was appropriate.

- DOE has developed plans for future research to help address uncertainties associated with glass performance (e.g., field-scale lysimeter test, additional glass release experiments).
- It is not appropriate to make conclusions about the significance of uncertainties in glass release without considering other uncertainties that are also addressed using a relative change in system performance argument.

The NRC staff has the following recommendations with respect to VLAW glass wasteform performance:

- The amount of cracking for VLAW glass should be determined using samples of appropriate scale and composition using the cooling profile specification that will be used in the design. (Recommendation #6)
- The basis for the retention of volatile species in VLAW glass should be improved and base case analyses should use more conservative values for retention of volatile species in glass until additional information is developed to support higher retention values. (Recommendation #7)
- DOE should compare the range of waste compositions included in glass durability testing to the wastes present in the tank farms and the wastes expected to exit the pre-treatment tank (AP-106), evaluate whether the previous tests adequately considered the range of compositions expected in the waste stream, and perform additional glass durability tests as needed. This testing should also assess the buildup of minor species resulting from recycling and include the resultant phases during glass durability testing. (Recommendation #8)
- It is recommended that cold glass production-scale samples from be non-destructively or destructively examined to determine the amount of cracking. (Recommendation #9)
- DOE should perform additional analyses to investigate the potential bias in results of glass release modeling as a result of using a coarse numerical grid. (Recommendation #10)

4.7 Non-Glass (Cementitious) Wasteform Performance

As part of the vitrification process of LAW in the WTP, secondary waste is expected to be produced that will subsequently be disposed at the IDF. Additionally, liquid waste streams are expected to be generated at the WTP during this process that would be transferred to the ETF for further treatment, and a concentrated portion of this liquid waste stream would be solidified and stabilized into a cementitious wasteform for disposal at the IDF[17]. Although DOE does not consider the secondary waste streams generated during the vitrification process of DFLAW to be within the scope of the Draft WIR evaluation for VLAW, as described in Section 1.4, the NRC does consider these waste streams to be within the scope of the WIR determination.

[17] DOE referred to SSW generated by WTP operations as “SSW” and SSW generated by ETF operations as “ETF SSW”. DOE also refers to the ETF-SSW waste stream originating from WTP operations as WTP LSW, ETF-treated LSW, or ETF-LSW. The NRC staff uses the term “SSW” to refer to all secondary wastes generated by waste processing as a result of producing the VLAW wasteforms.

In the IDF PA (RPP-RPT-59958, 2018)], DOE stated that it intends to solidify or encapsulate the secondary waste streams from the WTP and the ETF into cementitious wasteforms prior to their disposal at the IDF. DOE indicated that the use of cementitious waste matrices has been studied over the past few decades at Hanford. DOE noted that the cementitious materials offer beneficial physical and geochemical properties (e.g., low hydraulic conductivity, mechanical strength, sorption) and the wasteforms can be augmented to enhance physical or chemical properties (e.g., enhance the sorption of key radionuclides).

In the following section, the NRC staff provides an overview of DOE's modeling of release from the cementitious materials and the NRC's review of the modeled performance of these wasteforms. The expected contribution of the cementitious wasteforms to the overall dose from the vitrification and disposal of the VLAW waste is summarized.

4.7.1 DOE's Analyses of Non-Glass (Cementitious) Wasteform Performance

Cementitious release models were used by DOE to evaluate contaminant release rates due to advection and diffusion from the solid secondary waste containers and solidified liquid secondary waste containers. The performance characteristics of cementitious wasteforms include low hydraulic conductivity to limit advection, low effective diffusion coefficients to limit diffusion, and high mechanical strength to provide dimensional stability and to limit dispersion. DOE used experimental measurements and analog information to develop the parameter ranges prescribed for the cementitious wasteforms. DOE simulated contaminant release from cementitious wasteforms and transport to the base of the facility (above the liner). Three-dimensional models of the different cementitious wasteform configurations (encapsulated or solidified, boxes or drums, mortar or paste or hydrated lime grout) were simulated using the unsaturated flow and transport capabilities of STOMP. DOE also developed a system model with a simplified release from the cementitious wasteforms to perform sensitivity and uncertainty analyses using GoldSim.

4.7.1.1 Secondary Waste Streams Modeled in PA

Secondary waste is expected to be generated by WTP and ETF operations as a result of the production of VLAW glass. Forms of secondary waste DOE expects to generate from WTP operations include: melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, HEPA filters, and other process-related waste. DOE stated the WTP SSW waste streams explicitly addressed in the SSW data package (SRNL-STI-2016-00175, 2016) included the GAC (carbon adsorber beds), ion-exchange (IX) resin, HEPA filters, and silver mordenite (AgM). DOE indicated that these waste streams were explicitly included because they are expected to contain higher radionuclide inventories than the other WTP SSW. DOE addressed the cementitious wasteform that would be generated from the solidification and stabilization of liquid waste from the ETF in PNNL-25194/RPT-SWCS-006 (PNNL-25194, 2016).

DOE stated that for the purposes of the IDF PA, SSW is generally classified into two categories: debris waste (waste with a particle size > 60 mm (2.4 in)) and non-debris waste (waste with a particle size ≤ 60 mm (2.4 in)). DOE described these wastes as being contained in a cementitious matrix through solidification or encapsulation. DOE plans to use solidification for non-debris wastes such as resins or GAC that could be homogeneously blended into a grout or other stabilization matrix. DOE plans to use encapsulation for wastes with major voids (e.g., HEPA filters) that will be filled by an encapsulating matrix and will have an envelope (thickness) of encapsulating matrix around the waste. In the modeling DOE assumed the encapsulating material would be 10 cm (4 in) thick.

In Section 3.3.3.2 of the IDF PA (RPP-RPT-59958, 2018), DOE noted that the final specification of the solidification and encapsulation matrices for SSW from the WTP are currently under development. DOE said that they intend to select the matrices based on several performance factors, including adequate mechanical strength for handling, transportation, and emplacement, compatibility with other engineered barriers and wasteforms in the IDF, and limited rates of release of contaminants of potential concern into the IDF. In the responses to the NRC staff's RAI (DOE, 2021c), DOE clarified that no decision has been made to develop an alternate grout mix to the grout mix assumed in the PA, Hanford Grout Mix 5, so this grout mix remains the expected path forward for disposal of cementitious wasteforms at the IDF. In Section 3.3.3.3 of the IDF PA, DOE noted that two different cementitious wasteforms were being considered for the ETF-generated LSW grout: a fly ash-based formulation and a hydrated lime formulation. Section 5.1.3.2.6 of the PA indicates that the current expectation is that the hydrated lime formulation will be used.

DOE indicated that they expect the LLW disposed at the IDF to be placed in containers similar to those currently being used for disposal at the Hanford Solid Waste Burial Grounds (i.e., 55 gallon [0.25 m^3] drums, B-25 rectangular container, and the waste burial box). These containers are typically made of carbon steel. DOE did not take credit for the containers in its waste release models.

In Section 3.3.5 of the IDF PA, DOE described the assumed inventories of radionuclides in the different waste streams in the different inventory cases. DOE notes that in Inventory Case 7 (i.e., the nominal inventory case for the IDF PA), the AgM and GAC waste streams are major contributors of the iodine inventory expected to be disposed at the IDF. According to Table 3-27 in the IDF PA, the expected curies of ^{129}I from the SSW is only slightly less than the inventory from the glass (i.e., 12.1 Ci for SSW and 16.52 Ci for the ILAW glass). DOE also noted that the ion-exchange resins and the HEPA filters are the largest source of ^{99}Tc in the SSW. Per Table 3-27 in the IDF PA, the ^{99}Tc in the SSW represents a smaller fraction of the total ^{99}Tc compared to the ILAW glass. The predicted inventory of ^{99}Tc and ^{129}I in the ETF-LSW waste in Inventory Case 7 is small. However, as described in more detail in Section 3.2.1.1 of this TER, DOE evaluated alternate inventory cases, and in some of these cases (e.g., Case 10A and 10B) a majority of the ^{129}I and a significant fraction of the ^{99}Tc was estimated to be in the ETF-LSW.

In RPP-CALC-61030 (RPP-CALC-61030, 2017), DOE identified a number of uncertainties associated with the waste loading scenarios, including: the spacing between waste containers for different wasteform types, the loading of different wasteforms into the IDF (i.e., the number of containers per lift and the number of lifts), the potential for mixing of different wasteforms vertically or spatially, the trench space taken by each wasteform, and waste container sizes used for each wasteform. DOE indicated that their STOMP models simulated release from one lift and assumed that the release would be additive over four lifts. DOE assumed that the concentrations of the radionuclides in the backfill surrounding the containers are low enough that the releases from upper lifts would not significantly affect the modeled releases from lower lifts.

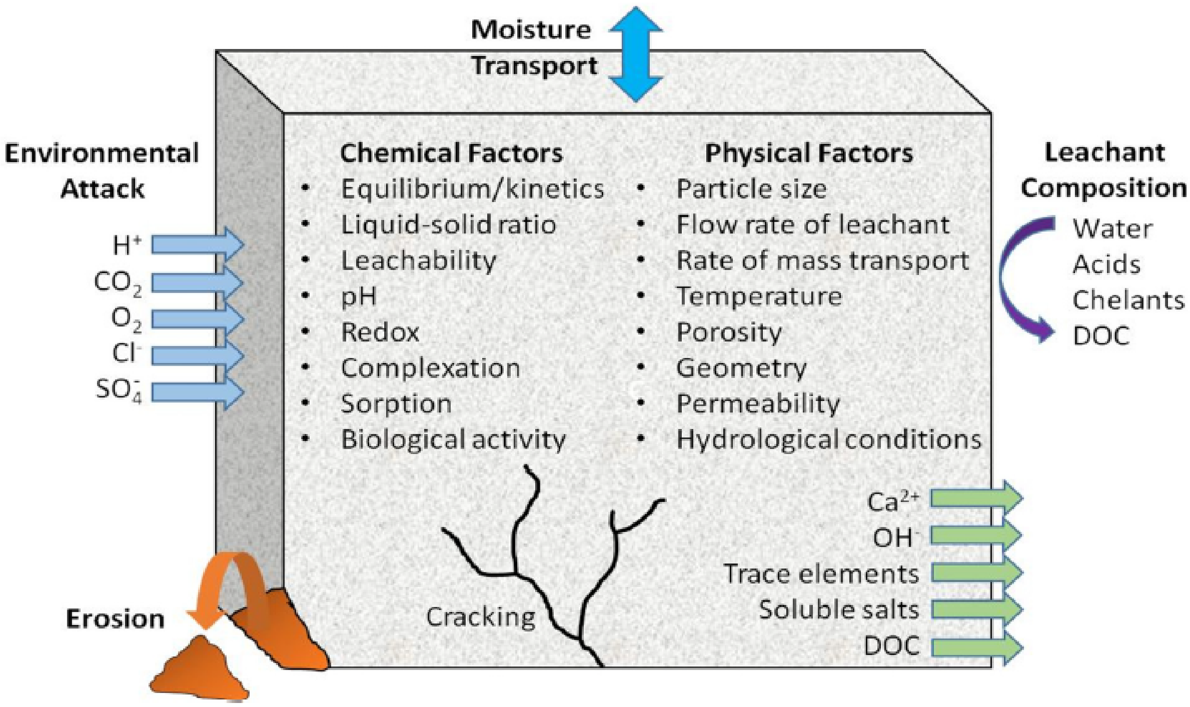
4.7.1.2 Models Used to Calculate Release from Cementitious Wasteforms

To develop an understanding of potential performance of cementitious wasteforms, DOE completed modeling with STOMP and GoldSim. DOE describes these calculations in the IDF PA and in other supporting documents (RPP-ENV-58738, 2015, RPP-RPT-59342, 2016, and RPP-CALC-61030, 2017). The cementitious wasteforms are one component of the broader performance assessment model for disposal of LAW at the IDF and thus the modeling of cementitious wasteforms needed to be integrated into the performance assessment modeling. In RPP-RPT-53942, DOE considered three separate but related numerical models: (a) a two-phase flow model, (b) an advective-diffusive release model, and (c) a diffusion-limited release model (RPP-RPT-53942, 2016). The two-phase flow model was used to evaluate the spatial/temporal distribution of moisture/air/water vapor in the backfill in and around the wasteforms. This model was also used to evaluate the near-field saturation and flow conditions for advective-diffusive release models. Two release models were developed to evaluate the release of radionuclides by diffusive and advective transport to the base of the IDF where they became a source term for the vadose zone fate and transport model. The release models differed primarily in the significance of advective release from the cementitious wasteforms. For high-quality cementitious wasteforms releases are generally dominated by diffusion whereas for lower performing wasteforms advection may be significant if enough recharge is present.

DOE indicated the key components of the PA that the cementitious release modeling must consider included:

- The post-closure inventory,
- Engineered barrier degradation (chemical [e.g., redox] and physical [e.g., aging]),
- Infiltration and recharge,
- The hydrogeological setting,
- The model domain and boundary conditions, and
- Vadose zone flow and transport.

DOE stated that the relevant processes and factors affecting radionuclide release and transport from the cementitious wasteforms included two-phase flow, advective-diffusive transport, solubility, and retardation/sorption. DOE initially considered these processes and factors in detailed process models. For eventual implementation in the system model many of these processes and factors were simplified. Processes associated with the aging of the wasteform were identified and included mechanical and chemical attack affecting the multi-phase flow- and advective-diffusive transport processes (Figure 4-12).



DOC = dissolved organic carbon
Source: RPP-RPT-59342, "*Integrated Disposal Facility Model Package Report: Non-Glass Release*," Figure 3-1.

Figure 4-12 Processes Potentially Affecting Constituent of Potential Concern Release from Cementitious Wasteforms
[Figure 4-11 in the IDF PA (RPP-RPT-59958, 2018)]

DOE indicated that the conceptual model of radionuclide release and transport depended on the nature of the waste stream incorporated in the cementitious wasteform. For example, redox-sensitive radionuclides could be precipitated in the pores of the cementitious wasteform due to low solubility of their reduced species. Radionuclides could also be distributed in pores of the cementitious wasteform that is encapsulated within a layer of clean grout. The partitioning of the redox-sensitive radionuclides between the cementitious material and the water in the pores varies depending on the redox conditions. The common characteristic of each of these starting conditions was the assumption that the radionuclides were distributed in the pore space of the cementitious wasteform or they could be released into the pore space. Release of radionuclides could occur by either diffusion (due to the concentration gradient) or advection (if there is sufficient flow of water through the wasteform, either by a capillary gradient or gravity). Though the wasteforms will be disposed in containers, DOE did not credit the steel walls of the container as a barrier to flow of gas or water.

DOE stated that the cementitious wasteforms are expected to evolve and degrade over time as a result of various processes. Physical degradation (e.g., desiccation, shrinkage, cracking) tends to lead to changes in physical properties (notably porosity and permeability) which may in turn affect the flow of water through the wasteform. Chemical degradation processes considered included the effects of carbonation, sulfate attack, and oxidation that may alter the mobility of radionuclides in the cementitious wasteform and lead to physical degradation. DOE indicated that while quantitatively predicting the temporal evolution of the physical-chemical characteristics of the cementitious wasteforms was conceptually possible it was not implemented in the PA. DOE had limited data associated with the controlling processes. For example, DOE stated that although mechanical degradation of the wasteform is possible, there are limited data available with which to correlate the degree of mechanical degradation to changes in effective diffusion in the expected partially saturated environment of the IDF. Similarly, although changes in the wasteform oxidation state may be anticipated due to the diffusion of oxygen into partially saturated fractures and pores in the wasteform, there are limited data available with which to correlate the degree of oxygen ingress to changes in sorptive capacity or solubility of radionuclides. In both examples, rather than explicitly attempt to model these processes, DOE used a range of parameter values derived in supporting data packages (PNNL-25194, 2016 and SRNL-STI-2016-00175, 2016) to evaluate the effects of changes in physical/chemical characteristics by external coupled processes. DOE stated that more detailed predictive approaches may be explored in the PA maintenance program if it is determined that the uncertainty in these coupled process interactions significantly affects the calculated performance of the facility.

The geometry of the wasteform is a key factor in determining transport distances and area available for release. Primary parameters of the diffusion-limited release model were dimensions and tortuosity of the wasteform, and the diffusion coefficient, distribution coefficient, and initial inventory for the radionuclides. The next section describes the parameters used by DOE. DOE also included two different waste container configurations (i.e., the B-25 container and 55-gallon drums) in their advective-diffusive transport models.

Two controlled-use software packages were used to simulate flow and transport to calculate near-field hydrologic conditions and source-term releases. These software packages were STOMP and GoldSim Pro, which were qualified for controlled use at the Hanford Site in accordance with their respective software management and testing plans. DOE described the different model configurations and associated aims, model purpose, key processes, and model dimensions. Both two- and three-dimensional modeling was used depending on the scenario and processes being evaluated (RPP-RPT-53942, 2016). Numerical models were constructed of representative sections of the IDF facility accounting for the geometry of the wasteform and surrounding backfill for different wasteform configurations. The geometry of the waste was represented in a refined mesh to track the saturation fronts and transport of radionuclides from the wasteform by advection and diffusion (both radially and vertically). Grid cell spacing ranged from 1 to 15 cm (0.4 to 6 in).

DOE's modeling of release from cementitious wasteforms in the GoldSim system model used more simplified representations than that used for the process modeling with STOMP. The wasteforms were generally represented with five cells to simulate diffusion within the waste and the liquid saturation in the wasteforms was prescribed rather than simulated. The liquid saturation was prescribed a value of either 0.9 or 0.95 (for either encapsulating or solidification) up until year 550 of the simulation and then a value of 0.99 until the end of the simulation. The high values were thought to be conservative, and the modeled results were not very sensitive to saturation values in these ranges. The wasteform cells were connected to cells representing surrounding backfill. The backfill cells were then connected to cells representing the unsaturated zone. Advection could occur within the backfill but only diffusion was simulated withing the cement wasteforms.

In summary, the following list provides a summary of DOE's key assumptions and modeling approach for releases from cementitious SSW:

- SSW, Fast Flux Test Facility (FFTF), onsite non-CERCLA non-tank and solid wastes are collocated in the IDF. These wastes comprise a small fraction (about 3%) of the IDF volume and footprint.
- Different SSW streams have different concentrations of radionuclides and different wasteform characteristics.
- SSW streams may be either encapsulated (debris SSW) or solidified (non-debris SSW).
- Encapsulated cementitious wasteforms have paste-like physical properties, while solidified cementitious wasteforms have mortar-like physical properties.
- Sorptive properties of waste streams vary; for example, ^{129}I is highly sorbed on granular activated carbon (GAC) and Ag mordenite, while high-efficiency particulate air (HEPA) filters are assumed to have no sorption of mobile radionuclides like ^{129}I and ^{99}Tc.
- Because the wasteforms have different physical-chemical properties, the releases from the wasteforms containing different waste streams are modeled individually.
- Degradation of cementitious wasteforms does not significantly affect physical properties.
- Diffusion is the release mechanism for dissolved radionuclides.
- The assigned diffusivity of radionuclides within the compacted HEPA filters does not include the possibility of grout seeping into the compacted HEPA filters.
- Retardation characteristics for redox-sensitive radionuclides are dependent on the saturation of the wasteforms and the composition (i.e., presence of reductants like blast furnace slag).

4.7.1.3 Parameters Assumed in Cementitious Wasteform Release Calculations

DOE described the parameter values used for the calculated releases from cementitious wasteforms in Section 4.4.1.3 of the IDF PA as well as in RPP-CALC-61030 (RPP-CALC-61030, 2017). The basis for these values was further described in two data package documents (PNNL-25194, 2016; SRNL-STI-2016-00175, 2016). DOE indicated that the SSW data packages relied on information from existing studies from cementitious materials that are representative of mixes that may be used for the stabilization or encapsulation of the SSW. Information on the inventory and volumes of the different waste streams was provided in RPP-ENV-58562 (RPP-ENV-58562, 2016). Physical, hydraulic, and transport properties and parameters for near-field and far-field engineered materials associated with the IDF in the 200 East Area of the Hanford Central Plateau were compiled in PNNL-23711 (PNNL-23711, 2015).

In the IDF PA, DOE identified the following as inputs to the non-glass release model:

- Water flow rate through the backfill above and along the sides of the waste containers developed using the near-field hydrology model,
- Waste type, including debris or non-debris,
- Waste package geometry,
- Wasteform type (solidified or encapsulated),
- Wasteform initial conditions, including initial saturation and redox-state,

- Chemical properties of the water and air in contact with the cement-based wasteforms, and
- Hydraulic and diffusive properties of representative grouts, including mortars and pastes.

The key data sets for calculating the COPC release rates from the cementitious wasteforms included hydraulic and related physical property characteristics, unsaturated characteristics as defined by the van Genuchten parameter values, effective diffusion coefficients, and sorption coefficients (e.g., K_d). Because the wasteform characteristics are different for the SSW and ETF-LSW, different property values were developed for each parameter. In addition, there are different SSW waste streams which have different treatment options (i.e., solidification versus encapsulation), which lead to different physical characteristics of the wasteforms.

DOE performed calculations using different assumptions and parameter values to evaluate the sensitivity of the conceptual models to uncertainties in the various inputs. These calculation cases included a series of references cases (referred to as ACM1), alternate conceptual model cases (ACM2), and parameter uncertainty cases (PUs). The ACM1 cases assume that there are oxidizing conditions in the cementitious wasteform, while the ACM2 cases assume reducing conditions. The ACM1 cases were calculated for both B-25 containers and 55-gallon drums and for two different infiltration rates. The parameter uncertainty cases included alternate assumptions on parameters such as the saturated hydraulic conductivity of the wasteforms, the initial water saturation of the wasteforms, the van Genuchten parameter α, the residual water saturation of the backfill, the infiltration rate, the effective diffusion coefficients, and the sorption coefficients (K_d). For the ETF-SSW, the calculations were performed using both the Case 7 and Case 10 inventories.

DOE summarized the data used in Appendix B of RPP-CALC-61030 (RPP-CALC-61030, 2017). Hydraulic property data of the near-field materials, mainly the low- and high-density backfill, were summarized in Table B-2. The backfill hydraulic properties were similar to those of the Hanford Formation underlying the IDF. The initial water saturations of the low- and high-density backfill were 0.09 and 0.2, respectively. The corresponding parameters for the different cementitious SSW formulation were summarized in Tables B-4 and B-5 including summary statistics for the hydraulic properties (i.e., saturated hydraulic conductivities, dry-and solid densities, porosities) and van Genuchten parameters for different cement compositions representing the potential uncertainty in the range of hydraulic properties. For example, the mean saturated hydraulic conductivity for SSW from 98 measurements was reported as 8.5×10^{-9} cm/s (2.8×10^{-10} ft/s). Dry bulk density averaged 1.59 g/cm^3 (99.3 lb/ft^3) and porosity was 0.33.

The effective diffusion coefficient is a key parameter for estimating releases from a wasteform when releases are predominantly from diffusion. In Table B-7 of RPP-CALC-61030 DOE identified best estimate effective diffusion coefficients for grout, paste, and mortar as 3×10^{-8} cm^2/s, 2.9×10^{-8} cm^2/s, and 5.4×10^{-8} cm^2/s, respectively. When diffusion is measured in the laboratory or field, different processes can be included in the results. In addition, the technical literature is not always consistent when reporting values and using terminology. DOE discussed the relationship between effective diffusion, molecular diffusion, tortuosity, porosity, and saturation. DOE explained how laboratory data was analyzed to ensure the correct value was being used in the simulations. The reported diffusion coefficients in SRNL-STI-2016-00175 were material specific, whereas in a STOMP simulation the diffusion coefficient is contaminant specific. To account for different diffusion coefficients for the encapsulated waste and surrounding clean grout, the D_e of the encapsulated waste was scaled by a tortuosity value to yield the corresponding D_e values for the encapsulating grout. For encapsulated waste, such as B-25 box containing compacted HEPA filters, a high value of $D_e = 5.6 \times 10^{-6}$ cm^2/sec was recommended, because of uncertainty in the form and effective properties of the debris wastes that may be disposed. This effectively resulted in no performance credit being taken for resistance to diffusive transport within the waste zone. Only the encapsulating grout provided resistance to diffusive transport, for which a mean value of $D_e = 2.9 \times 10^{-8}$ cm^2/sec was used.

An additional key parameter that can significantly affect the modeled retention of the radionuclides in the cementitious wasteforms is the assumed K_d value, or sorption coefficient. Both ^{129}I and ^{99}Tc exhibit large differences in sorption under different redox conditions with ^{99}Tc having low sorption (K_d) under oxidizing conditions and high sorption under reducing conditions and ^{129}I having low sorption under reducing conditions and high sorption under oxidizing conditions. DOE indicated that although the difference in the retention of these radionuclides under different redox conditions could be due to solubility limits, they are modeling this as a sorption phenomenon in the IDF PA. The K_d values assumed in the PA for the cementitious wasteforms were primarily developed using literature values. For the solidified wasteforms (i.e., GAC, AgM, and IX-Resin), the overall K_d value of the wasteform was developed by volume averaging the K_d values of the individual components (e.g., volume averaging the K_d for GAC and cement to determine the K_d for the solidified GAC).

The NRC staff asked for additional information associated with the parameters assigned to solidified LSW (NRC, 2021). Staff stated that the effective diffusion coefficient values used were lower than those used for other wasteforms and the diffusive path length was longer. In response to the RAI, DOE indicated that the wasteform developed for LSW was tailored to the waste stream and therefore would be expected to have different performance characteristics. DOE indicated that additional grout formulations are being evaluated and will be incorporated using the change control process. DOE summarized new information that indicated that the release of ^{129}I from ETF-LSW may have been underestimated by approximately an order of magnitude. Because ETF-LSW only contributed a small amount to groundwater doses, the increase in dose would not be significant.

DOE described different processes that could contribute to degradation (aging) of non-glass wasteforms (see Section 4.7.1.2). DOE stated that aging of the wasteform has been correlated with the amount of water that interacts with the wasteform. Because limited water is expected to contact the wasteforms in the IDF, DOE indicated that degradation from chemical attack would be expected to be minimal. DOE stated that physical degradation of the wasteform due to deformation cracking may be significant, but the adverse effect of cracks is expected to be minimal with respect to moisture and solute transport due to the low saturation in the surrounding backfill material. The associated enhanced migration of oxygen into the wasteform was accounted for by assuming oxidizing conditions for redox-sensitive contaminants. Oxidizing conditions were modeled by specifying the sorption coefficients that are applicable under oxidizing conditions. As a result, the potential effects of degradation of the wasteform were not included in the process modeling. However, DOE performed sensitivity analyses to evaluate the effect of aging by increasing the effective diffusion coefficient of the cementitious wasteforms after 500 years.

DOE continues to complete research to better understand non-glass wasteform performance and to develop parameter values. In PNNL-28545, Rev. 1 DOE evaluated properties for three SSW streams included in the IDF PA. Ongoing research will be used to verify the assumed values for SSW in the IDF PA and to fill limitations in available data by gathering site-specific data for the Hanford SSW disposal. These may include (1) the sorption/desorption behavior of key contaminants (technetium, iodide, iodate, mercury) expected to be found in Hanford SSW (GAC, AgM, HEPA) in simulated grout pore water conditions; (2) the leaching behavior of iodide from stabilized/blended GAC/AgM in oxidized and reduced grout; (3) the ability of two down-selected grout mixes to stabilize GAC/AgM upon curing; and (4) providing additional solid characterization data on candidate grout mixes to immobilize SSW. In the PA maintenance plan, DOE identified activities to continually evaluate national and international research on grout properties and release models to see if useful insights can be learned and used to compare with conceptual and numerical model assumptions and parameter values used in the IDF PA. DOE illustrated this data collection effort to protect key assumptions in the IDF PA.

4.7.1.4 Calculated Releases from Cementitious Wasteforms

In Section 5.1.3 of the IDF PA, DOE provided the results of the calculated releases from the cementitious wasteforms from the bottom of the IDF to the top of the vadose zone. Releases were calculated for each waste stream and DOE focused on the release of ^{99}Tc and ^{129}I since they have historically been the radionuclides that are of most concern at the IDF. DOE provided the results in terms of graphical time histories of the fractional release rates and cumulative fraction released for a period of 10,000 years as well as presenting the fractional release rates averaged over periods of 1,000 years and 10,000 years. DOE used an initial infiltration rate of 0.5 mm/yr (0.02 in/yr) after 100 years followed by a long-term infiltration rate of 3.5 mm/yr (0.14 in/yr) after 500 years in the base case simulations (RPP-RPT-59958, 2018). DOE simulated releases from B-25 boxes and 55-gallon drums (releases from a B-25 box are lower than a drum keeping all other parameters the same).

Figure 4-13 provides an example of the simulated releases. The results provided are for releases of ^{99}Tc from HEPA filters stabilized as a cement wasteform. These results are representative of the simulated results from most secondary wastes for radionuclides with limited sorption. As DOE explained, the ^{99}Tc fractional release curves showed an early increase after about 100 years, followed by a rapid increase at 500 years to a maximum rate of 1.0×10^{-3} 1/yr followed by a rapid decline to about 3.9×10^{-4} 1/yr and a subsequent slower decline to about 4.6×10^{-6} 1/yr after 10,000 years. The cumulative fraction of the initial ^{99}Tc inventory released from the wasteform increased to about 0.26 after 1,000 years, and to 0.98 after 10,000 years, indicating that most of the SSW-HEPA inventory was released by the end of the 10,000-year simulation period. The results for ^{129}I indicated that encapsulated SSW-HEPA was the main contributor to the total ^{129}I released.

Though a significant fraction is released from the wasteform in the first 1,000 years, the impacts to a potential user of groundwater are delayed until well after the 1,000-year compliance period even for a very mobile radionuclide (^{99}Tc). The spike in the release curves corresponds to the assumed failure of the engineered cover and the increase in infiltration rates around year 500. The spike in release rates from the wasteform does not directly correspond to a similar response in dose to a receptor because the transport through the engineered barriers of the facility and the natural system results in significant dispersion (smoothing) of the release rates (curves). The slope of the cumulative release fraction curves, with a time delay from transport,

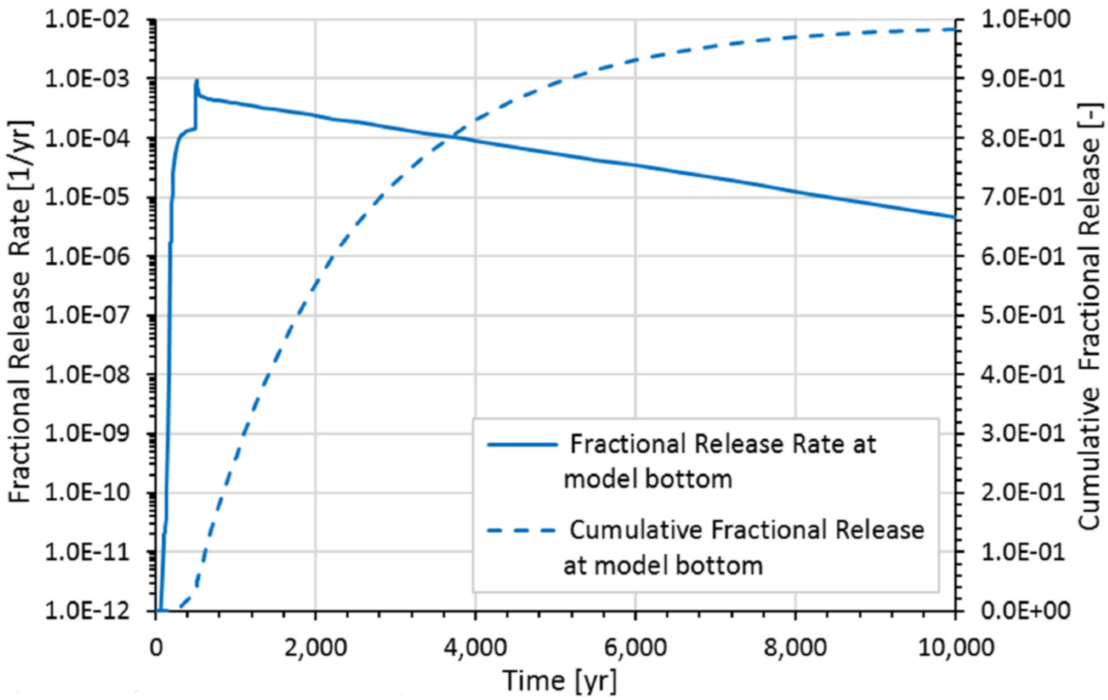


Figure 4-13 Fractional and Cumulative ^{99}Tc Release Rates from HEPA Filter Cement Wasteforms

[Modified from Figure 5-51 in the IDF PA (RPP-RPT-59958, 2018)]

is a better indicator of the eventual dose to a receptor. The SSW waste streams have different release rates, concentrations of radionuclides, and volumes which then translate into different risks. Overall, the SSW was the dominant contributor to the overall groundwater pathway doses for the base case.

DOE supplemented the base case evaluation with numerous sensitivity cases to examine changes in parameter values or other assumptions. For example, in the base case the cement was assumed to be oxidizing. An alternate case examined the introduction of blast furnace slag or other materials to produce a reducing wasteform. Use of reducing cement reduced the release of ^{99}Tc by many orders of magnitude but can increase the release of other radionuclides. Table 4-3 lists sensitivity cases examined by DOE for the Grout Release Safety Function for cementitious wasteforms. DOE compiled the results of sensitivity cases in tables (e.g., Table 7-19 of RPP-CALC-61030). Outside of the reducing conditions case, the fractional cumulative release of ^{99}Tc in the first 1,000 years ranged from 0.058 to 0.669 (base was 0.26).

The results demonstrated that there is uncertainty associated with the estimated release rates from cementitious SSW. DOE supplemented the previously discussed analyses with evaluations focused on specific safety functions.

Table 4-3 Sensitivity Cases for Cementitious Wasteforms

Case Identification	Description
Reducing K_d	Reducing K_d for ^{99}Tc and ^{129}I throughout the time period from 0 to 10,000 years post-closure.
Vary_ K_d	Same as Starting Case except vary grouted wasteform K_d values gradually from reducing to oxidizing for ^{129}I and ^{99}Tc from outer to inner portions of the wasteform.
Vary_ K_d _Deff	Same as Starting Case except vary grouted wasteforms K_d values gradually from reducing to oxidizing from the outer portions of the wasteform to the inner portions of the wasteform for ^{129}I and ^{99}Tc. Increase the D_{eff} by a factor of 60× at the same spatial and temporal rate as the change in K_d.
EIS_K_d_Deff	Same as Starting Case except change K_d values for ^{129}I and ^{99}Tc and 60× increase D_{eff} instantaneously 500 years after closure.
5cm_Encap	Same as Starting Case except reduce the thickness of the encapsulating grout used for HEPA, FFTF, non-CERCLA, non-tank and solid waste and other debris from 10 cm to 5 cm.
15cm_Encap	Same as Starting Case except increase the thickness of the encapsulating grout used for HEPA, FFTF, non-CERCLA, non-tank and solid waste and other debris from 10 cm to 15 cm.

Different safety function analyses associated with cement wasteforms were:

- **Grout Deff** - Use best estimate effective diffusion coefficients for SSW mortar and paste based on SRNL-STI-2016-00175 (Table 7-4). Use geometric average effective diffusion coefficient for sodium for hydrated lime-based ETF-LSW based on laboratory data summarized in PNNL-25194, Table 3.1.
- **Grout K_d** - Use best estimate K_d value for cementitious solids under oxidizing conditions for Stage I cement degradation (Table 8-4 of SRNL-STI-2016-00175) or use waste stream-specific K_d values for those specified in SRNL-STI-2016-00175.
- **Grout Diffusion Lengths** - Use nominal diffusion lengths between wasteform and backfill and nominal waste container sizes and encapsulation thicknesses (for example 10 cm of encapsulating grout used for compacted HEPA filters and other debris waste).
- **Backfill Diffusion Lengths** – The properties of the backfill can impact the estimated release rates from the cementitious wasteform as a result of the calculated concentration gradients.

The results of these analyses were similar to the previously described results in terms of cumulative release fractions with time.

DOE examined the impact of parameter uncertainty on the releases from cementitious SSW using the system model. DOE provided comparisons of results of similar cases from the process modeling and system modeling to demonstrate that the insights from the system modeling would be consistent with that expected from using the process model. DOE also

provided comparisons of the process model results from the current evaluation with previous results obtained for the EIS. DOE identified the parameter uncertainty cases as:
• PU1: Ksat (saturated hydraulic conductivity of the wasteform)
• PU2: Sw (initial water saturation of the wasteform)
• PU3: VG-α (van Genuchten parameter)
• PU4: Srw (residual water saturation of the backfill)
• PU5: Inf (infiltration rate)
• PU6: Deff (effective diffusion coefficient)
• PU7: K_d (sorption coefficient).

DOE evaluated parameter uncertainty with both one-at-a-time sensitivity analyses as well as global probabilistic analyses. Both normalized and raw data were compared to facilitate understanding of relative and global significance. Figure 4-14 provides the results of probabilistic analysis of the performance of cementitious secondary solid wastes. Both timing and magnitude of potential doses were influenced by the uncertainties examined. The maximum groundwater pathway dose was approximately 0.1 mSv/yr (10 mrem/yr). DOE estimated that it is very unlikely that groundwater impacts will occur within the 1,000-year compliance period. Not included in Figure 4-14 was the impact of uncertainty in the partitioning of volatile radionuclides between glass and secondary wastes.

DOE stated that the base case, sensitivity cases, and uncertainty analyses were used to develop an understanding of system performance. The analysis of the suite of cases was used to identify the parameters that have the most impact on simulated releases so that a greater emphasis can be placed on reducing the uncertainty in these parameter estimates provided the models indicate that additional protection from these wasteforms is necessary.

4.7.2 NRC's Evaluation of DOE's Analyses of Non-Glass Wasteform Performance

Non-glass wasteforms are anticipated to be generated by the treatment of low-activity waste at Hanford. Engineered wasteforms are one of the most risk-significant barriers used to mitigate potential radiological doses from radioactive waste disposal. Properly designed and implemented wasteforms can result in a risk reduction of many orders of magnitude compared to not using a wasteform. Because the wasteforms are designed (man-made), uncertainties can be better controlled in comparison to the natural system (e.g., retardation, dispersion, dilution). In the IDF, wasteforms provide a stable media to limit void formation after closure. Though DOE anticipates that glass wasteforms will comprise 90% of the waste disposed in the IDF, depending on the performance characteristics of each wasteform, the non-glass wasteforms could be significant to demonstration of compliance with the performance requirements. The following sections describe the NRC staff's evaluation of the performance of non-glass wasteforms with an emphasis on retention of radioactivity and potential release to the environment.

DOE performed extensive calculations related to estimation of the releases from cementitious wasteforms. Those calculations and results were described throughout numerous references (e.g., RPP-RPT-59958, 2018; RPP-ENV-58738, 2015; RPP-RPT-59342, 2016; RPP-CALC-61030, 2017). Many additional references were reviewed by the NRC staff. The NRC staff also reviewed, modified, and analyzed different GoldSim model files to develop independent risk insights.

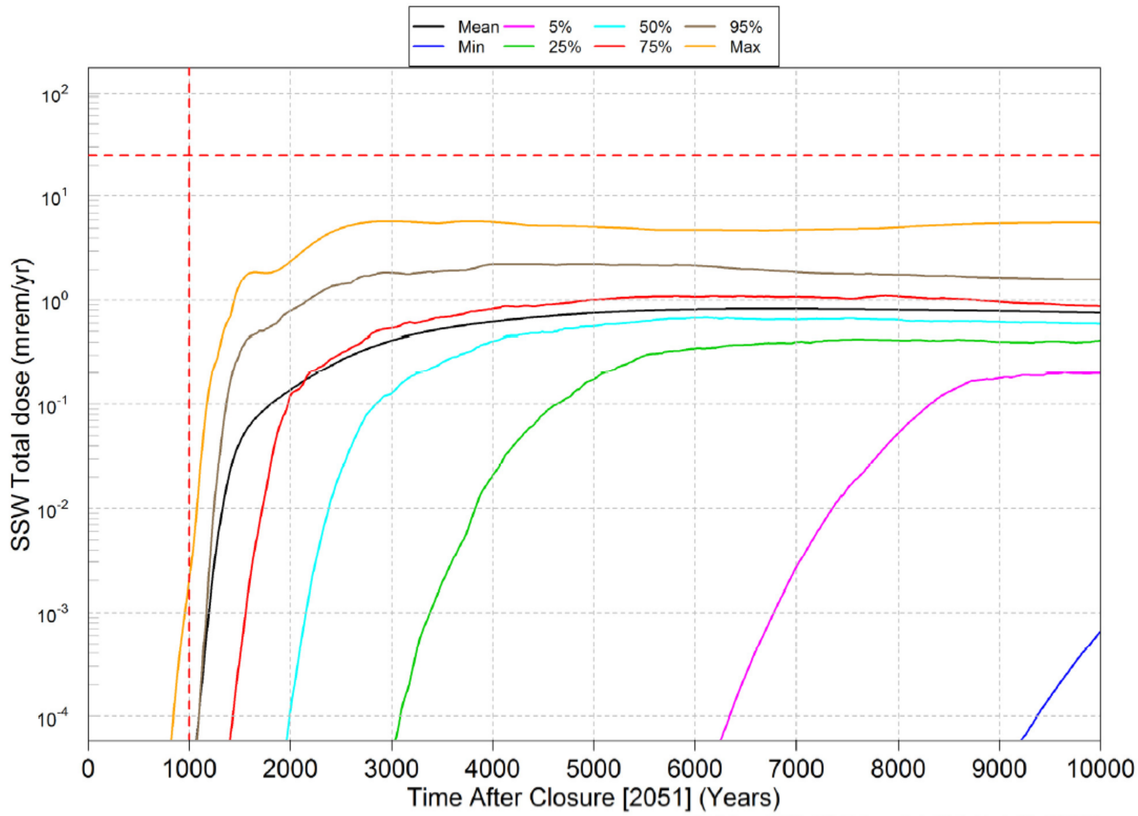


Figure 4-14 Uncertainty in Groundwater Pathway Doses from Secondary Solid Wastes
[Figure 6-136 in the IDF PA (RPP-RPT-59958, 2018)]

Overall, DOE's description of the non-glass wasteform performance was very thorough and many simulations were performed. DOE demonstrated an understanding of how non-glass wasteform performance fit into the broader risk context of the performance assessment. DOE effectively used charts and tables to communicate the information. NRC staff agrees with DOE's statement that the base case, sensitivity cases, and uncertainty analyses all supply useful information to develop the risk context of non-glass wasteform performance. As NRC staff discuss in a later section (Section 4.13.2), proper interpretation of different analyses can be important to understanding the risk context of the evaluation.

The transparency and traceability of the analysis of non-glass wasteform performance for the base case with the STOMP process model could be improved. It can be a challenge to summarize complex analyses that are completed on a large project over an extended timeframe. DOE provided different references (listed above) that in some cases seemed to point to each other for information without one reference providing a clear source of the parameters used in the base case. Though NRC staff does not recommend emphasis of one analysis result of a complex, uncertain system, some stakeholders do like to have a reference point for comparison of information. In the tabulated values provided it was not clear what values were used in the deterministic process modeling for the base case completed with STOMP. A table may have provided the data as well as summary statistics, but which value was used in the base case was not clearly described. It is recommended that DOE provide an appendix to the PA that explicitly identifies the parameters used in the base case and other key cases to improve the transparency and usability of the PA and to facilitate future independent review (Recommendation #11). In addition, this type of information may be used as DOE progresses to operations so that verification that actual performance is within prescribed operational values can be completed.

DOE's use of process modeling (i.e., with STOMP) combined with system modeling (i.e., with GoldSim) was an effective approach to develop an understanding of the performance and importance of non-glass wasteforms. Each approach, process and system modeling, can have strengths and weaknesses. DOE effectively used the different types of modeling in a complimentary manner. DOE identified key assumptions and uncertainties. Besides the exceptions noted in the following sections, NRC staff agrees with the key assumptions and uncertainties identified by DOE.

4.7.2.1 NRC's Evaluation of Secondary Waste Streams Modeled in PA

DOE described the types of secondary wastes that will be generated, their properties (volumes, forms, radioactivity), and how they will be disposed in the IDF. Forms of secondary waste DOE expects to generate from WTP operations include: melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, HEPA filters, and other process-related waste. The WTP SSW waste streams explicitly addressed by DOE included GAC, IX resins, HEPA filters, and silver mordenite (AgM). DOE indicated that these waste streams were explicitly included because they are expected to contain higher radionuclide inventories than the other WTP SSW. DOE appropriately described the cementitious wasteforms that would be generated from the solidification and stabilization of liquid waste from the ETF. The waste streams DOE included in the PA were appropriate.

DOE's final specification of the solidification and encapsulation matrices for SSW from the WTP are currently under development. DOE stated that they intend to select the matrices based on several performance factors, including adequate mechanical strength for handling, transportation, and emplacement, compatibility with other engineered barriers and wasteforms in the IDF, and limited rates of release of contaminants of potential concern into the IDF. NRC staff agrees that these are appropriate factors to consider when establishing specifications for non-glass wasteform performance.

DOE indicated that they expect the LLW disposed at the IDF to be placed in containers similar to those currently being used for disposal at the Hanford Solid Waste Burial Grounds (i.e., 55 gallon [0.25 m^3] drums, B-25 rectangular container, and the waste burial box). These containers are typically made of carbon steel. DOE did not take credit for the containers in its waste release models. NRC staff agrees that it was conservative to not take credit for performance of the containers in the analyses. The strategy for emplacement of waste within the IDF is not finalized. DOE indicated that this is a performance variable that will be considered under the PA maintenance program. NRC agrees that it is appropriate to consider emplacement strategies within the PA maintenance program.

The inventory that will be contained within SSW is uncertain. DOE examined different inventory cases in the PA analyses. Those cases demonstrated the importance of which wasteforms the volatile radionuclides are partitioned. In addition, concentration of certain radionuclides (e.g., ^{90}Sr) in particular waste streams could impact demonstration of compliance with the intruder protection performance objective. DOE has proper processes in place to ensure the inventory assigned to SSW and non-glass wasteforms will be consistent with expected operations. DOE should more clearly describe what measurements will be taken when and where within the operational process to confirm the concentrations of key radionuclides in waste streams and in non-glass wasteforms to enhance confidence.

DOE properly identified uncertainties associated with the waste loading scenarios, including: the spacing between waste containers for different wasteform types, the loading of different wasteforms into the IDF (i.e., the number of containers per lift and the number of lifts), the potential for mixing of different wasteforms vertically or spatially, the trench space taken by each wasteform, and waste container sizes used for each wasteform. NRC staff agrees with DOE that it was intractable to evaluation all of these combinations within the PA and that the cases examined by DOE were appropriate.

4.7.2.2 NRC's Evaluation of Models Used to Calculate Release from Cementitious Wasteforms

DOE clearly described the conceptual model for releases from cementitious wasteforms. DOE expects that recharge in the IDF will be low even after deterioration of the engineered cover. The limited recharge combined with development of cements of appropriate quality will result in diffusion dominated releases of radioactivity from the wasteforms. From the NRC staff's perspective, the key consideration is whether DOE will be able to produce cementitious wasteforms of acceptable quality over the range of wastes that will be processed.

Successful modeling relies on a combination of verification and validation. DOE's empirical testing is a necessary first step in developing appropriate formulations for wasteforms. However, it is not a sufficient condition by itself. Given the complexity of wastes that will be addressed, it is imperative that DOE implement a robust process of verification and validation to ensure that the wasteforms that are produced will achieve acceptable performance.

DOE indicated the key components of the PA considered in their modeling of the release of radionuclides from cementitious wasteforms. NRC staff agree that DOE properly considered the PA context. The significance of some aspects of the PA context are likely influenced by uncertainties that are not yet fully resolved. Some of these uncertainties will be evaluated in DOE's PA maintenance program. DOE stated that they considered degradation of cementitious wasteforms (as shown in Figure 4-12). DOE described their consideration of degradation and analyzed the potential effects of degradation in select sensitivity analysis cases (e.g., by increasing the effective diffusion coefficient). Using this approach, the effects of degradation processes are not present in the base case results. DOE assumed that no significant degradation would occur over the full range of wastes that must be treated and stabilized. Historical operations at Hanford have generated large volumes of diverse wastes. These wastes are in different forms (solid, liquid, slurry), have different major and minor mineral phases present, and have considerably different chemistry. DOE has performed extensive research on the performance of cement wasteforms. However, the translation of that largely experimental research into real-world production of wasteforms from the diverse wastes at Hanford must be done very carefully and cautiously. Internal reaction mechanisms such as alkali silicate reaction (ASR) or ettringite formation are becoming better understood but have many uncertainties. These processes occur in "normal" concrete and cements under certain conditions. It is not yet possible to accurately predict when these processes will occur. The

Cement Barriers Partnership (CBP) was making progress to address this important question, but the program was terminated when funding was not renewed. The wastes that are processed at Hanford may have some unique characteristics and the necessary performance timeframe is certainly atypical of normal applications. The cementitious wasteforms at Hanford could experience similar unforeseen internal reactions. For these reasons it is recommended that DOE perform testing of non-glass wasteforms over the full range of waste variability (including minor species and organics) using testing over extended timeframes or with proper acceleration (Recommendation #12).

DOE's modeling of the release of radionuclides from non-glass wasteforms was challenging because of the diverse waste streams that will be generated (e.g., GAC, AgM) and numerous other sources of uncertainty or variability. DOE simplified the number of combinations that needed to be assessed by recognizing if some parameters were similar, then estimated releases could be generated for a more limited set of waste types. Nonetheless, the assessment had numerous permutations. The key assessment variables included:

- Waste type (e.g., HEPA filter, GAC, ETF-LSW)
- Package type (e.g., B-25 box, 55-gallon drum)
- Modeling type (process or system)
- Analyses type (deterministic, probabilistic, sensitivity, uncertainty)

DOE described the modeling that was performed in a series of references (RPP-RPT-59958, 2018; RPP-ENV-58738, 2015; RPP-RPT-59342, 2016; RPP-CALC-61030, 2017). NRC staff reviewed those references and select underlying references. The staff found the documents to be complete and comprehensive in terms of documentation and interpretation of results. Staff did not identify quality issues. Data sources were provided and appropriately referenced. DOE clearly described the modeling in terms of the software used, assumptions, and model discretization. DOE identified who performance the analyses and that appropriately qualified software was used. There was not as much transparency of the data used in the process modeling or specifically how the results of different calculations were integrated into other calculations. There was much greater clarity in terms of the data used in the system modeling compared to the process modeling. The NRC staff verified that data used in the system modeling matched what was stated in the PA document.

DOE used STOMP to perform the process modeling of releases from non-glass wasteforms. STOMP has been widely used at the Hanford site for a variety of different analyses. DOE used STOMP to simulate the liquid saturation of the various materials in the system as a result of recharge through the barriers overlying the wasteforms. Release modeling was primarily from diffusion. Key parameters were the effective diffusion coefficient of the wasteform, the distribution coefficients of radionuclides in cement, and the diffusive lengths and areas. NRC staff verified the proper calculation of surface areas and diffusion lengths for the B-25 boxes and 55-gallon drums. DOE identified the key parameters associated with diffusive release.

Some wastes will be encapsulated, whereas others will be solidified. The modeling of release in these scenarios is different because in one case, the barrier surrounds the radioactivity and in the other case, the radioactivity is embedded in the matrix. DOE described an approach of averaging K_d values for solidified SSW. DOE represented the solidified matrix with a volume-averaged K_d value considering the amount of waste and cement. DOE asserted the result would be conservative. The integrated release from a volume-averaged K_d wasteform may be conservative but that conclusion should be verified experimentally.

The wasteforms will be emplaced in the disposal facility in different waste containers. The emplacement configuration was not yet determined at the time the analyses was completed. In the system modeling DOE assumed that two lifts of non-glass wasteforms would be stacked and that non-glass wasteforms would be segregated from glass wasteforms. DOE stated that emplacement configurations may be optimized through future analyses such as evaluations completed in their PA maintenance program. The NRC staff agree that it is appropriate to consider different waste emplacement configurations in the PA maintenance program. The approach to modeling waste emplacement was appropriate.

As discussed in Section 4.5.2, the NRC staff did not find that DOE's assessment of model discretization for process modeling was sufficient for near-field flow (and glass release modeling). A similar conclusion is made here with respect to process modeling of releases from cementitious wasteforms. DOE described the discretization that was used but did not provide analyses to demonstrate that the discretization used was sufficient. Lack of proper discretization can represent a source of bias rather than a source of uncertainty or variability. Some confidence is derived from the comparisons of the process model results with the system model results provided by DOE.

Though DOE anticipated that diffusion would dominate the release from cementitious wasteforms, that could change in the future if challenges are encountered with making high-quality cementitious wasteforms for the diverse wastes at Hanford or if recharge is estimated to be higher than anticipated. The NRC staff's comments associated with unsaturated flow modeling of advection in glass wasteforms would also apply to the modeling of advection through cementitious wasteforms.

4.7.2.3 NRC's Evaluation of Parameters Assumed in Cementitious Wasteform Release Calculations

DOE performed calculations using a wide variety of parameter values for the main parameters that can be risk-significant for the release of the radionuclides. These analyses were a good starting point for estimating the potential risk from the disposal of cementitious wasteforms at the IDF and for identifying the most risk-significant parameter values. The NRC staff note that although DOE did provide a lot of information on the calculations performed, it was not always clear what parameters were assumed in each case. In particular, the specific assumptions and parameters used for the base case (process modeling) were not clear from the IDF PA or primary references. In Appendix C of RPP-CALC-61030, DOE provided an example of how values discussed in the references were modified for use in STOMP. This is a good practice, however, as noted in Recommendation #11, the transparency and usability of the PA could be improved if the parameters used in key cases (e.g., the base case) were explicitly listed in a table or appendix to the PA.

DOE primarily used literature values from previous research in its determination of the parameter values assumed in its cementitious wasteform release calculations because the final formulations had not yet been established when the IDF PA was written (RPP-RPT-59958, 2018). DOE provided additional information on the expected cement wasteform formulations in their response to RAIs (DOE, 2021c). Given that the formulations have not been finalized and processing and disposal of this waste has not yet begun, the use of generic literature values as a starting point was reasonable. However, the generic information might not be applicable to the final formulations used to stabilize or encapsulate these waste streams. Additionally, the generic information might not be applicable to the cementitious wasteforms generated with the LAW waste streams. Therefore, site-specific values of the key parameters need to be measured using the final formulations for the cementitious wasteforms and representative waste streams.

DOE used STOMP modeling to estimate the initial water saturations of the low- and high-density backfill as 0.09 and 0.2, respectively. The water saturation of the backfill at closure is likely to be a function of how long the closure process lasts. The backfill will be exposed to the environment and may have unfavorable geometry from the perspective of run-off (surface water flow). If the backfill saturation is not significant to estimated releases, then the approach DOE used was reasonable. However, some modeling platforms will use different forms of power laws to represent tortuosity as a function of liquid saturation. DOE may want to perform sensitivity calculations where the backfill is fixed at select liquid saturation values.

DOE identified the effective diffusion coefficient is a key parameter for estimating releases from a wasteform when releases are predominantly from diffusion. NRC staff reviewed the data provided in various references and summarized in Appendix B to RPP-CALC-61030 (RPP-CALC-61030, 2017). The best estimate effective diffusion coefficients for grout, paste, and mortar DOE used of 3×10^{-8} cm^2/s, 2.9×10^{-8} cm^2/s, and 5.4×10^{-8} cm^2/s, respectively, were reasonable and consistent with the referenced reports. Some tests had much lower or higher values. These lower or higher values could be driven by different formulations or other factors. Unless waste concentrations are extremely low, the composition of the waste can impact the initial quality of the wasteform as well as the long-term durability. The waste at Hanford is very diverse with high compositional variability. Though the tanks used for pre-processing of waste are very large and will help reduce variability, there is significant uncertainty that wasteforms developed in the laboratory will have adequate performance when developed for the actual waste. As described above in Recommendation #12, the testing to develop the key parameters (e.g., effective diffusion coefficient) must be consistent with the range of compositions of the waste that will be immobilized. This includes consideration of what may be minor species that may be enhanced or concentration during processing. The NRC further recommends that the WAC for the SSW and the cementitious wasteforms should be consistent with the performance observed over the full range of testing (Recommendation #13).

The terminology of effective diffusion coefficients and related parameters is mixed in the literature and sometimes confusing. If information is used from literature sources outside the project it is important to determine what processes are included in the results of the measurements. DOE clearly described the terminology used within the project and how it was ensured that correct values were used in the process and system modeling.

For encapsulated waste, DOE used a high value of $D_e = 5.6 \times 10^{-6}$ cm^2/sec because of uncertainty in the form and effective properties of the debris wastes that may be disposed. This effectively resulted in no performance credit being taken for resistance to diffusive transport within the waste zone. Only the encapsulating grout provided resistance to diffusive transport, for which a mean value of $D_e = 2.9 \times 10^{-8}$ cm^2/sec was used. It was appropriate for DOE to use this conservative approach in lieu of collecting additional data. However, caution should be used with this conservative approach if the waste streams being evaluated end up as the risk drivers. DOE has a process to track assumptions made and through the PA maintenance and change control process can collect information and perform additional analyses as needed.

The assumed K_d value, or sorption coefficient, can significantly affect the modeled retention of the radionuclides in the cementitious wasteforms. The chemical retention of radionuclides is sometimes included in diffusion coefficient measurements as "apparent" diffusion coefficients. Both ^{129}I and ^{99}Tc exhibit large differences in sorption under different redox conditions with ^{99}Tc having low sorption (K_d) under oxidizing conditions and high sorption under reducing conditions and ^{129}I having low sorption under reducing conditions and high sorption under oxidizing conditions. DOE examined the base case with oxidizing conditions and an alternate case with reducing conditions. Both cases had acceptable performance. The benefits of using reducing conditions for ^{99}Tc were offset by higher releases of ^{129}I. Because each of these radionuclides may be collected in different parts of the waste processing system, it may be possible to produce reducing wasteforms for ^{99}Tc and oxidizing wasteforms for ^{129}I.

DOE modeled the retention of different radionuclide in cements as a sorption phenomenon in the IDF PA. The K_d values assumed in the PA for the cementitious wasteforms were primarily developed using literature values. NRC verified the K_d values used were consistent with the literature. For the solidified wasteforms (i.e., GAC, AgM, and IX-Resin), the overall K_d value of the wasteform was developed by volume averaging the K_d values of the individual components (e.g., volume averaging the K_d for GAC and cement to determine the K_d for the solidified GAC). This approach was appropriate for initial modeling but is non-physical and should be revised based on future testing.

Different processes can contribute to degradation (aging) of non-glass wasteforms (see Section 4.7.1.2). NRC agrees with the DOE statement that chemical attack of the wasteform is correlated with the amount of water that interacts with the wasteform. The significance of degradation processes is related to the performance of the wasteform. If very low values of effective diffusion coefficients and hydraulic conductivity are used for the non-glass wasteforms then degradation processes could drive release rates. If high values of the parameters are used, then degradation may be of limited significance. Given DOE's performance credit for non-glass wasteforms, their approach to consideration of wasteform degradation was reasonable. DOE should revisit the assessment of degradation processes if more performance credit is assigned to the non-glass wasteforms or as dictated by additional empirical measurements. DOE stated that physical degradation of the wasteform due to deformation cracking may be significant, but the adverse effect of cracks is expected to be minimal with respect to moisture and solute transport due to the low saturation in the surrounding backfill material. This argument is, in effect, taking credit for a capillary barrier effect of the backfill. This result should be verified empirically similar to the flow of moisture around and through cracked glass (see Section 4.6.2). Internal reactions as a result of incompatibility of the waste and wasteform matrix are the degradation mechanism that could result in large differences between expected and actual performance. These processes are being actively studied throughout the world. DOE should continue to stay abreast of the research of degradation by internal reactions and actively incorporate empirical evaluation, if necessary.

DOE appropriately used different diffusion lengths for different wasteforms and configurations. In the RAI, the NRC staff asked for additional information associated with the diffusion length of 0.2 m (0.6 ft) for ETF-LSW waste. In the response to the RAI, DOE stated that the diffusion length in the system model was selected based on calibration/comparison with the results from process modeling. DOE provided the results of sensitivity analyses that showed the fractional release rate of ^{99}Tc and ^{129}I increased as the diffusive pathway length was decreased. DOE indicated that the fractional release rates in the system model with a 0.2 m (0.6 ft) pathway length were considerably higher than the estimated releases in the process model. DOE considered maximum fractional release rates as well as the cumulative release in 10,000 years when calibrating the system model discretization against the process model results. The use of calibration to achieve better agreement in the simulated results decreases the model support that can be derived from model intercomparisons. Matching of peak values and integrated 10,000-year cumulative releases may not yield optimum results. Peak releases, especially if the peaks are very sharp, are significantly dispersed during transport through the engineered barriers and natural system. The cumulative releases over 10,000 years include decreasing releases over many thousands of years. These releases will not contribute a significant fraction of the peak groundwater doses which will be driven by the integrated releases over a certain portion of time. Analysis of different release profiles may better identify the shape of the fractional release curves that drive groundwater doses such that the calibration process is based off risk-significant output metrics.

As with the glass wasteforms, it is important for DOE to perform verification testing of the actual wasteforms. The laboratory tests of surrogates are a necessary but not sufficient condition to ensuring production of wasteforms that will protect public health and safety. Cements can be complex, and the presence of organic matter and other components can affect its performance. NRC staff recommend that DOE should perform verification testing of the cementitious wasteforms at the beginning of operations and after significant process changes occur (Recommendation #14). NRC staff concludes that DOE’s development of specifications for cements/grouts has not accounted for composition variances and process scale-up. It is recommended that DOE account for waste composition variability and process scale-up when developing specifications for non-glass wasteforms (Recommendation #15). DOE has an extensive PA maintenance program. NRC staff strongly supports this program to further develop key parameters supporting the performance of non-glass wasteforms. Both laboratory and field experiments may yield valuable insights that lead to innovation and better decision-making. Lysimeters tests seem to be particularly well-suited to providing model support.

4.7.2.4 NRC’s Evaluation of Calculated Releases from Cementitious Wasteforms

NRC staff reviewed DOE’s analyses used to estimate releases from cementitious wasteforms. DOE used a combination of process modeling with STOMP and system modeling with GoldSim to estimate releases. Releases were estimated for different waste streams (e.g., HEPA, GAC) as well as different stabilization methods (encapsulated and solidified) and container types (B-25 boxes and 55-gallon drums). DOE generated fractional release rate curves as well as cumulative release rate curved to communicate the performance estimates.

As shown in Figure 4-13 for SSW-HEPA, the releases from cementitious wasteforms were characterized by a rapid increase up until shortly before 500 years, followed by a spike at 500 years, with a rapid and then gradual decline. The magnitude of the estimated releases were consistent with the parameters and models used. The early shape of the release curves is likely to be more a function of assumptions made by DOE rather than a physical result. DOE elected to take a pessimistic (conservative) approach to representing some engineered features of the system.

The engineered cover system is not yet fully designed but is anticipated to be similar to the Hanford Barrier that was installed and studied over two decades ago. Because of uncertainties in the long-term performance of the barrier, DOE assumed that the cover would function for the first 500 years and then infiltration rates would have a step change to long-term average values (3.5 mm/yr (0.14 in/yr)) (RPP-RPT-59958, 2018). In addition, the wastes will be disposed in carbon steel containers that are typically painted. These containers will provide a barrier to diffusion and advection until they degrade. Though carbon steel degradation is generally "rapid" with respect to the timeframes considered, the Hanford environment is semi-arid and the engineered cover may create a low moisture environment while it is functioning. When an engineered cover or steel containers degrade, they do so gradually such that loss of function is distributed in time and space. These aspects are not extremely important to the long-term impacts because transport through the natural system results in considerable smoothing of releases from the source term. However, if monitoring of releases from the IDF to the sump or below the liner were to be performed, it should not be expected that releases would match the simulated results. Releases from SSW to the air pathway in the compliance period are likely to have been significantly overestimated. The sharp spikes in the air pathway releases are not dispersed (smoothed) as much as groundwater pathway releases.

The cumulative releases of ^{99}Tc were on the order of 10-60% in the first 1,000 years for most cases examined. The release of ^{129}I was slower than ^{99}Tc but a significant fraction was released after 1,000 years. The release of ^{99}Tc from compacted HEPA filters that will be packed into a B-25 box with a 10 cm (4 in) cement encapsulating layer were the most significant source of the groundwater dose. Disposed HEPA filters were the dominant source of ^{129}I released from SSW waste streams.

For all simulations, DOE estimated that it was very unlikely that more than an insignificant amount of radionuclides would be released to the aquifer within the 1,000 year compliance period. The engineered barriers and unsaturated zone were estimated to delay the arrival of release contaminants for more than 1,000 years. In the 10,000-year timeframe only the most mobile contaminants (^{99}Tc and ^{129}I) were expected to reach a hypothetical well for extraction of groundwater. NRC staff performed independent verification of the calculated arrival times and concur with the DOE results.

DOE provided numerous sensitivity and uncertainty analyses to investigate uncertainty and variability. The magnitude of groundwater doses after 1,000 years was 0.015 mSv/yr (1.5 mrem/yr) from all sources combined. In the sensitivity cases for SSW, DOE demonstrated that no single feature, event, or process would be capable of increasing the dose to the 0.25 mSv/yr (25 mrem/yr) dose standard that is applied in the 1,000-year compliance period. The evaluation of different inventory cases showed that if too much volatile radionuclides are partitioned to SSW then doses exceeding 0.25 mSv/yr (25 mrem/yr) could occur (after 1,000 years). However, DOE indicated that the use of off-gas recycling combined with improvements in glass melter system performance will ensure that the amount of volatile radionuclides partitioned to SSW will be low.

A sensitivity case described by DOE may highlight the need for further evaluation of unsaturated flow around and within cementitious wasteforms. DOE examined uncertainty in the physical property of the solidified mixture of grout and IX-Resin by assuming sand properties for the mixture and assuming the initial saturation in the wasteform to be in capillary equilibrium with the surrounding backfill. This resulted in relatively low water saturation (Sw=0.125). However, the ^{129}I release in this scenario was not much different than the mortar-based solidification. In fact, the release rate for the sand-based solidification was lower at late time due to lower saturation and corresponding lower effective diffusion (RPP-CALC-61030 Figures 7-52 and 7-53). This result would seem to suggest that according to DOE's modeling, use of sand would provide an as effective barrier as the use of cement. NRC staff expects that the result is non-physical and driven by the coarse discretization of the numerical grid and the use of uniform properties for different material types. As discussed in Section 4.6, DOE should carefully examine simulated capillary barrier effects.

Because the development of cementitious wasteforms for SSW is ongoing, it would be advantageous to identify scenarios or performance characteristics of wasteforms where performance objectives (or action guidelines) would not be achieved. For instance, at what effective diffusion coefficient would cementitious wasteform performance be unacceptable? At what hydraulic conductivity or degree of cracking would performance be compromised? There are likely to be unforeseen challenges and having a clear understanding of when an issue could result in a problem will help address and mitigate those challenges as they occur.

4.7.3 NRC's Conclusion of Non-Glass Wasteform Performance

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on non-glass wasteform performance. The NRC staff has the following conclusions with respect to non-glass wasteform performance:

- The non-glass wasteform will provide a significant barrier to the release of radioactivity. The non-glass wasteform will help limit the impacts from disposing of radioactive waste in the IDF.
- Research is ongoing to develop the performance characteristics of non-glass wasteforms. DOE appropriately summarized research completed to date.
- DOE did not include non-glass wasteforms within the scope of the draft waste evaluation, but the performance of the wasteforms was evaluated in the performance assessment.
- Non-glass wasteforms are unlikely to produce releases of radioactivity that will result in exceedance of the performance objectives for the compliance period.
- Wasteform performance does not impact intruder doses because the wasteforms are not assumed to provide a barrier to extraction of radioactivity. If the amount of radioactivity in SSW increased substantially intruder doses could be impacted.
- Doses to the air pathway from SSW are well below the performance objective and have likely been significantly overestimated in the compliance period.
- The conclusions for the non-glass wasteform would be impacted if DOE is not able to attain the partitioning of volatile species to glass stated in response to the NRC RAI.

The NRC staff has the following recommendations with respect to non-glass wasteform performance:

- DOE should provide an appendix to the PA that explicitly identifies the parameters used in the base case and other key cases to improve the transparency and usability of the PA and to facilitate future independent review. (Recommendation #11)
- DOE should determine the performance of non-glass wasteforms over the full range of waste variability (including minor species and organics) using testing over extended timeframes or with proper acceleration. (Recommendation #12)
- DOE should ensure that the WAC for the SSW and the cementitious wasteforms is consistent with the performance observed over full range of testing. (Recommendation #13)
- DOE should perform verification testing of the cementitious wasteforms at the beginning of operations and after significant process changes occur. (Recommendation #14)
- DOE should account for waste composition variability and process scale-up when developing specifications for non-glass wasteforms. (Recommendation #15)

DOE described their plans for future work under the PA maintenance program. NRC staff supports the PA maintenance program in general, and the specific actions DOE described. As DOE progresses from research to operations, it is recommended that DOE enhance the linkages between operational experience and observations and ongoing research. It is unlikely that the research and experimental basis will be able to accommodate all of the numerous variables associated with the complexity at Hanford. It will be very important to verify the quality of the wasteforms that are produced while operations are ongoing. The waste acceptance criteria (WAC) combined with the unresolved disposal question process will have an important role to ensure that highly performing wasteforms are produced. Those program elements must be sufficiently detailed and integrated with the results of the performance analyses to ensure protection of public health and safety.

4.8 Flow and Transport in the Unsaturated Zone

Flow and transport in the unsaturated zone is the movement of water, gas, and contaminants from the disposal facility to the underlying aquifer, in the case of water, and to the environment (air), in the case of gas. The following sections describe DOE's analyses of flow and transport in the unsaturated zone and the NRC staff's review of the information. The approach used by DOE to assess flow and transport in the unsaturated zone for VLAW (IDF PA) was very similar to that used for WMA C which the NRC staff had previously reviewed (NRC, 2020b). In the sections, below the staff provides an overview and discusses any new issues or observations.

4.8.1 DOE's Analyses of Flow and Transport in the Unsaturated Zone

As a result of historical operations at the Hanford Site, the hydrology has been extensively studied. The hydrogeological system at the IDF consists of a vadose or unsaturated zone roughly 70 m to 90 m (230 ft to 295 ft) thick. The unsaturated zone is comprised of three units: the H1 Hanford formation, H2 Hanford formation, and the undifferentiated H3/CCU/RF unit. Historically, both natural and anthropogenic recharge has occurred at the Hanford Site. Large amounts of water were used for site operations and much of that water was discharged to the unsaturated zone. Historical water usage impacted the transport of radioactivity that was discharged to the environment during operations.

DOE stated that the integrated knowledge obtained from previous and ongoing studies provides a conceptual understanding of the geologic, hydraulic, and geochemical environment and its controls on the distribution and movement of contaminants within the vadose zone (RPP-RPT-59958, 2018). The most significant features and processes associated with the unsaturated zone at the IDF are its thickness, low recharge, dispersion, and sorption. DOE estimated there would be approximately 70 m (230 ft) of the H2 sand unit and 14 m (46 ft) of the H3 gravel unit below the bottom of the IDF.

To complete computer modeling of flow and transport, DOE needed estimates of hydrologic and transport properties as well as the variability and uncertainty in those parameters. Data that are necessary for quantifying the water storage and flow properties of unsaturated soil include the moisture characteristics (i.e., soil moisture content versus pressure head, and unsaturated hydraulic conductivity versus pressure head relations) in various geologic units. The vadose zone hydraulic properties for the H2 sand unit were derived from laboratory experiments (core scale) conducted on samples representative of the Hanford H2 unit (RPP-20621, 2004). Boreholes have been used to provide samples for testing. Many samples were obtained from a limited number of boreholes. For the H2 unit, 44 samples were obtained from 3 boreholes. There were no measurements of hydraulic data available for the H3 gravel sequence at the IDF. DOE used borehole samples with high gravel contents from the 100 Area (15 samples) and 300 Area (10 samples) as surrogates to represent the vadose zone properties of the H3 gravel sequence (PNNL-23711, 2015). In comparison, the analyses for WMA C used 17 surrogate samples from near the ERDF and other locations in the 200 Area (NRC, 2020b).

DOE indicated that in arid and semi-arid regions with thick vadose zones, long-term factors like climate change, changes in the annual precipitation rates, and changes in vegetation structure and community are necessary to influence the infiltration into and vertical water flux within the vadose zone. Large seasonal fluctuations in soil water are generally contained within the upper few meters of soil. For this reason, prescribed infiltration rates did not vary spatially or temporally for a particular surface condition and time period (e.g., undisturbed soil, institutional control period).

The geochemical and sorption conceptual model was used to describe the movement and retardation of contaminants in the vadose zone. DOE used an empirical equilibrium sorption-based approach to approximate contaminant sorption during transport in the vadose zone. Due to mixing and buffering by mineral-water reactions, DOE expected IDF-derived water would be comparable to the ambient pore water (from a geochemical standpoint) within a short distance from the base of the IDF. A linear isotherm (constant distribution coefficient (K_d) model) is assumed to be generally applicable when: 1) contaminants are present at low concentrations, 2) the geochemical environment being modeled is not affected by large spatial or temporal changes, and 3) the possible sorption sites occupied by the contaminant remain much less than the sorption capacity over the scale of transport. DOE chose K_d values assuming low-salt, near-neutral waste chemistry (RPP-RPT-59958, 2018).

The Hanford Site was an area of volcanic activity millions of years ago. Lava flow produced clastic dikes in some locations. Clastic dikes are vertical to subvertical sedimentary structures that crosscut normal sedimentary layering (BHI-01103, 1999). Clastic dikes typically occur in swarms. In general, a clastic dike is composed of an outer skin of clay with coarser infilling material. Two clastic dikes were mapped in the IDF trench during construction. These dikes were not known to be present prior to construction. Detailed observations of the clastic dikes intersected during construction of the IDF are provided in PNNL-15237 (2005). Figure 4-15 shows the extent of the intersected dikes with blue lines on the plan view map of the IDF (RPP-RPT-59958, 2018). DOE stated that observations suggest the dikes are relatively lower permeability and, therefore, could act as a barrier to flow instead of providing a fast pathway (RPP-RPT-59958, 2018). When rain or snow fell in the trench during the time the trench was being mapped, the clastic dike remained relatively dry compared to the host sediment. While the dikes may form preferentially faster flow pathways under saturated conditions, they tend to act as barriers to transport under unsaturated flow conditions. The clastic dikes were not included in DOE's base case representation of flow and transport in the unsaturated zone but were evaluated in additional sensitivity cases.

There are no surface water bodies at or in the vicinity of the IDF that could impact flow and transport in the unsaturated zone, though run-on or run-off from large precipitation events could impact an engineered cover. The engineered cover will be designed to reduce infiltration to the disposed waste and reduce recharge of water to the unsaturated zone. Even under a probable maximum flood condition, DOE estimated that the disposal facility would not be impacted by surface water bodies (RPP-RPT-59958, 2018).

The numerical modeling tools used and the approach to implementing those models was nearly identical in the IDF PA compared to the WMA C PA. The travel time to the underlying aquifer for conservative species, those radionuclides that do not experience sorption, was greater than 1,000 years. The DOE performance objective in Order 435.1 specifies the use of a compliance period of 1,000 years.

Gases and vapors diffuse upward through the backfill and the closure barrier to the ground surface. After reaching the ground surface, the contaminants can be transported with the movement of air in the environment. The amount of the contaminants being released as a gas or vapor depends on the partitioning between the solid and liquid phases and the subsequent partitioning between the liquid and gas phases. DOE used two coefficients to describe these ratios: the partitioning between the solid and liquid phases is defined by the distribution coefficient (K_d) and the partitioning between the liquid and gas phases is defined by the Henry's Law constant (RPP-RPT-59958, 2018). The principal transport process is diffusion. DOE identified four potential contaminants that could be released in gaseous form: ^{14}C, ^{3}H, ^{129}I, and ^{222}Rn. DOE made pessimistic assumptions associated with gas transport because the dose impacts from gaseous radionuclides is typically small for the inventory evaluated and the engineered barriers used. Some barriers that would inhibit diffusion were not credited and Henry's Law constants were based on undissociated species. The emanation factor for radon was given a value of unity. Collectively the pessimistic assumptions were thought to produce estimated fluxes of gas to the land surface that were significantly higher than would be observed.

The gas-phase diffusive mass flux arriving at the surface was transported (except for radon), assuming advection and dispersion via wind movement, to the receptor. During the time of institutional controls, starting at the time of facility closure and lasting for 100 years, the receptor was assumed to be in the direction of the prevailing wind, about 20 km (12 mi) to the east-southeast. At the cessation of the institutional control period, the receptor was assumed to be located 100 m (328 ft) downwind from the edge of the IDF excavation, in the direction of the prevailing wind.

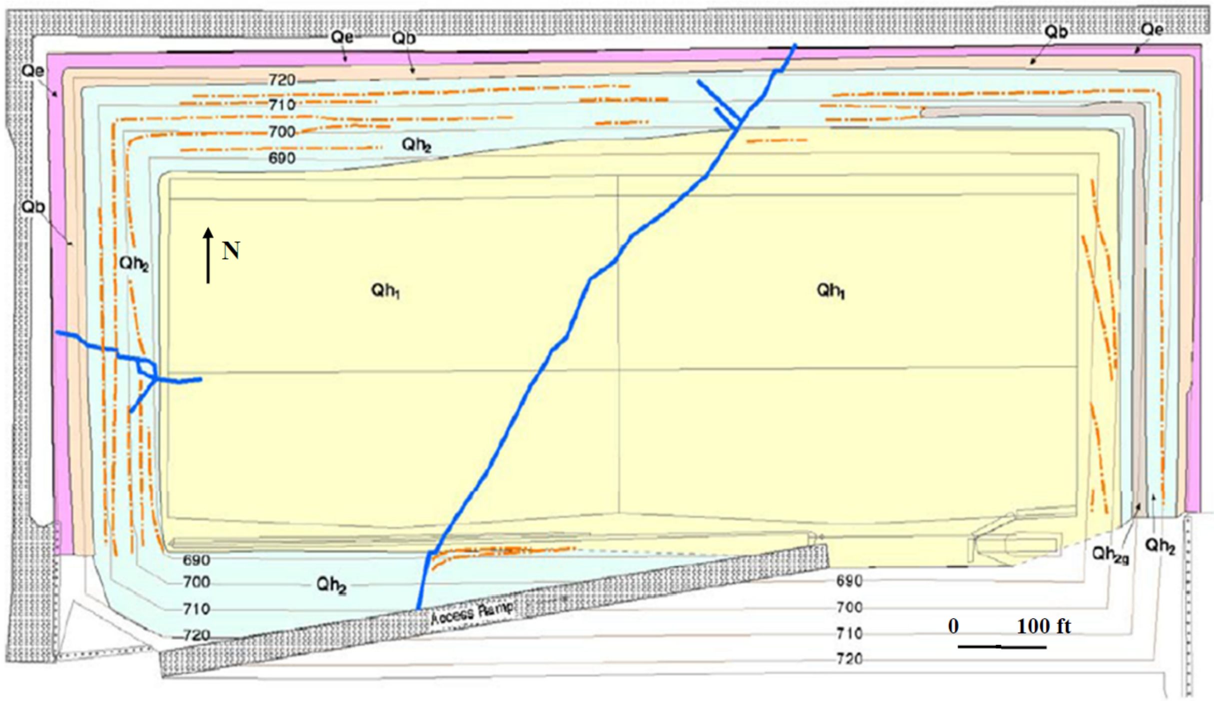


Figure 4-15 Clastic Dikes (shown in blue) Identified During Construction of the IDF
[Figure 3-42 from (RPP-RPT-59958, 2018)]

DOE made use of site-specific data on wind direction, wind velocity, and wind stability to calculate the downwind concentration at the receptor location using χ/Q values determined using the wind information, where χ /Q is the concentration at the receptor (in units of Ci/m^3) per unit release (in Ci/sec) from the source, where χ/Q is normally reported in units of sec/m^3. Using this approach took advantage of site-specific annual-average χ/Q values instead of calculating an equivalent χ/Q based on a fixed wind speed, direction and assumed stability class.

DOE described the data and its sources, the equations and models used, and simplifying assumptions that were made. Gaseous transport was evaluated in one-dimension with a column extending from the waste to the land surface. The lateral boundaries were no flux boundaries. The atmospheric transport calculations were based on standard approaches that are widely used.

4.8.2 NRC Evaluation of Flow and Transport in the Unsaturated Zone

The NRC staff reviewed the information DOE provided on flow and transport in the unsaturated zone. The modeling approach was very similar to that utilized for WMA C, therefore, the staff's review focused on differences as well as the risk context. The analyses used a process model and a system model. The system model was more computationally efficient and was used for sensitivity cases.

As with WMA C, DOE evaluated sensitivity cases to examine the potential effects of changes in key parameters and assumptions. The unsaturated zone flow and transport safety function was examined by using different K_d values. Using a K_d value of ^{129}I of 0.0 ml/g compared to 0.1 ml/g was shown to shift the timing of the peak dose from about 6,300 to 2,800 years and the magnitude of the dose increased from about 0.016 mSv/yr (1.6 mrem/yr) to 0.027 mSv/yr (2.7 mrem/yr). The magnitude of the changes was in line with the NRC staff's expectations. The observed changes were after the DOE compliance period of 1,000 years.

DOE used a process model to simulate a three-dimensional representation of the unsaturated zone extending 1,500 m (4,900 ft) in the east/west and north/south directions and to below the water table in the vertical direction. Minimum grid size spacing was 10 m (33 ft) by 10 m (33 ft) by 0.5 m (1.6 ft). DOE did not provide information with respect to the sufficiency of the grid spacing, though results from the process model and system model were compared and had sufficient agreement. The NRC staff does not expect significant bias in the results from grid spacing in this model. In the future, it would be good practice to complete additional analyses to evaluate the sufficiency of the grid spacing if the unsaturated zone flow and transport modeling were revised. The upper boundary of the simulation model was a defined flux boundary condition, which is appropriate. Because DOE did not know exactly how the wastes would be placed in the facility, they assumed a placement of wastes that would maximize concentrations (impacts) to the groundwater pathway receptor. This was a reasonable approach to mitigate the impact of waste emplacement uncertainty. On the other hand, the approach may have enhanced the perceived impact of SSW because it was confined to a small area compared to how it may ultimately be distributed in the IDF. The NRC staff recommends that DOE identify waste emplacement distributions that will minimize impacts and utilize those for disposal operations (Recommendation #16).

DOE appropriately identified the properties necessary to complete flow and transport modeling of the unsaturated zone at the IDF. DOE clearly described the data sources used and the analyses to modify the data. Documentation was transparent and traceable. DOE did not have data for the H3 unit from the vicinity of the IDF and, thus, used surrogate data. The NRC staff examined the data and determined the surrogate information was appropriate given the lessor importance of the H3 unit compared to the H2 unit. For the H2 unit, DOE used 44 samples from three boreholes to define hydraulic properties. These boreholes were in the vicinity of the IDF. DOE combined the data (i.e., assumed that intra-borehole variability was identical to inter-borehole variability). This approach is reasonable when sparse data are available but there is additional uncertainty as to whether the assumption is valid. In most simulations, the H2 unit was assigned average-upscaled uniform properties. The measurements were made on samples with dimensions of tens of centimeters, whereas the parameter values were assigned to grid blocks in the process modeling that are 100 m^3 (3,500 ft^3) or more. If the one-dimensional system model and the three-dimensional process model were using the same upscaled property sets, then they would be expected to agree with each other. It is possible that neither would agree with reality, which is why it is important to provide adequate model support as discussed in Section 4.12. In the future, an alternative approach that could be considered that may enhance the understanding of the importance of hydraulic data variability would be to use sets of measured data directly without upscaling to preserve structure, such as correlations in the data. A second alternative approach would be to generate geostatistically valid heterogeneity fields of properties to simulate unsaturated flow and transport. However, this second approach would currently be limited in utility because while data exists to define variability in the vertical (z) direction, the limited number of boreholes would likely result in very broad ranges of parameters in the x and y directions.

The use of liners and sumps in the design of the IDF can result in a significant redistribution of recharge within the facility before it exits into the unsaturated zone below the facility. In addition, the engineered cover will redistribute infiltration to the side slopes of the facility. This redistribution will only occur while these systems are functioning. DOE assumed that the systems will not function in the future, but the timing of that assumption is very uncertain. The systems may stop functioning at different times or at different rates. The three-dimensional process modeling can represent a non-uniform flux condition. The one-dimensional system model can represent a temporally varying flux condition but not a spatially-variable one. Given that the one- and three-dimensional models could produce similar results, it does not appear to the NRC staff that the spatial variability in flux is significant to the results. In general, the process modeling should only be used if necessary. The burden in setting up, executing, and interpreting the results from a more complex model can outweigh the benefits. Because parameters are upscaled and uniform values are assigned to a unit, the process modeling does not yield significant benefits over the system model.

DOE's selected K_d values assuming low-salt, near-neutral waste chemistry. The basis provided was that the release from the wasteforms was expected to be slow, therefore, the pore water chemistry should not be significantly impacted. Because there is uncertainty in the degree of mixing of natural pore waters with water containing chemicals leached from the IDF wasteforms, the approach taken in the IDF PA was to use the minimum "best" value for each contaminant in the base case representation. The approach taken for selection of K_d values for use in the unsaturated zone flow and transport model was reasonable. The contaminants that are the current drivers of risk from the groundwater pathway are unlikely to be impacted by changes in chemistry.

Clastic dikes have been observed at the IDF (RPP-RPT-59958, 2018). DOE believes these dikes are of limited vertical and areal extent. To evaluate the importance of a dike, DOE developed a two-dimensional cross-section representation of the unsaturated zone with a dike present. Based on the properties assigned to the dike, DOE observed that the simulations with the dike present resulted in slower movement of contaminants than without the dike present. The NRC staff reviewed the simulation results. DOE's observations are consistent with the properties assigned. Model support for dikes acting as barriers or impediments to transport was not provided. There is uncertainty associated with not only the impact of dikes (currently expected to be minimal) but also the extent of dikes. With the limited number of boreholes at IDF, it is difficult to address the vertical extent of dikes or the potential for the presence of a larger number of dikes at depth. Until the IDF was constructed, dikes were not believed to be present in that area. A vertically extensive dike that reaches the water table could enhance transport either by acting as a fast pathway or by focusing flow next to the dike. Contaminants could be focused along the dike in an area of higher saturation. It would be useful to complete a broader range of sensitivity analyses for dikes to address uncertainty in the properties of the dikes and uncertainty in the frequency and extent of dikes.

The numerical modeling was thorough, well-documented, and clearly described. The numerical modeling results are estimates, including uncertainty, given the conceptual model and supporting data. The NRC staff examined the documentation, the system model, and performed independent calculations of travel times and obtained results comparable to DOE. The NRC staff also reviewed the input parameters for the process and system models, many of which were similar to those used in the WMA C PA.

There are numerous sources of uncertainty and some of the inputs to flow and transport modeling are non-unique. Model support is essential to validate that the modeling was a proper representation (or conservative representation) of the real system. Historical operations at the Hanford Site have resulted in various planned and unplanned releases of radioactivity to the environment. Observation of the movement of that radioactivity provides a rich source of information against which numerical models can be compared. There are numerous challenges associated with using the information such as not knowing the exact timing and magnitude of the releases or the amount of moisture discharged. Because the transport of conservative radionuclides at Hanford is expected to take over 1,000 years to reach the aquifer under ambient recharge, model support cannot be developed under those conditions. If the numerical models developed are valid and are not spurious calibrations based on non-unique parameter sets, DOE should be able to use the models to simulate flow and transport of radionuclides under many different moisture regimes, including high discharges during operations. The purpose of such comparisons is not to accurately match a plume to a minimal percentage difference but rather to check more broadly that numerical models can replicate observations with consistency, considering the uncertainties. A numerical model that cannot replicate observations under high recharge is unlikely to accurately simulate flow and transport under low flow conditions.

DOE clearly described the data, assumptions, and analyses performed to calculate the transport of gaseous radionuclides. The methods DOE used were appropriate and consistent with industry standard approaches. The NRC staff finds that the pessimistic assumptions DOE used have resulted in estimated atmospheric concentrations at the hypothetical receptor locations that are quite likely to be significantly higher than will be observed in the future. Because the release of gaseous radionuclides was likely overestimated, DOE should ensure that radionuclides that can also contribute to the water pathway are not arbitrarily removed in the air pathway if releases to the groundwater pathway were to be included in the same model.

4.8.3 NRC Conclusions on Flow and Transport in the Unsaturated Zone

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on flow and transport in the unsaturated zone. Staff previously reviewed flow and transport in the unsaturated zone for application to WMA C (NRC, 2020b). The NRC staff has the following conclusions on flow and transport in the unsaturated zone applied to waste disposal at the IDF:

- The unsaturated zone is likely to provide a significant delay in the arrival time of contaminants to the underlying aquifer at the IDF.
- DOE's description of the flow and transport analyses performed for the disposal of VLAW at IDF was clear, transparent, and traceable.
- DOE's model estimates of travel times were consistent with the data.
- DOE's selection of K_d values assuming low-salt, near-neutral waste chemistry was appropriate.
- DOE's consideration of the impact of clastic dikes was appropriate and the conclusion that the presence of a limited number of dikes are not likely to significantly impact flow and transport in the unsaturated zone was properly supported.
- DOE's estimated flow and transport of gaseous radionuclides was appropriate.

The NRC staff has the following recommendation with respect to flow and transport in the unsaturated zone:

- DOE should identify waste emplacement distributions that will minimize impacts and utilize those for disposal operations (Recommendation #16).

4.9 Flow and Transport in the Saturated Zone

Flow and transport in the saturated zone is the movement of water and contaminants in the aquifer underlying the disposal facility. The following sections describe DOE's analyses of flow and transport in the saturated zone and the NRC staff's review of the information. The approach DOE used to assess flow and transport in the saturated zone for disposal of VLAW at the IDF was very similar to that used for WMA C which the NRC staff had previously reviewed (NRC, 2020b). Therefore, the staff provides an overview and discusses any new issues or observations. Flow in the saturated zone at the IDF is very important because the flux of water through the aquifer is very high in comparison to the vertical flux of water into the aquifer. As a result of mixing of these two fluxes, significant dilution occurs which reduces the concentrations of contaminants in the aquifer at the receptor well location.

4.9.1 DOE's Analyses of Flow and Transport in the Saturated Zone

Figure 4-16 shows a three-dimensional representation of DOE's combined unsaturated/ saturated flow modeling. DOE summarized the details of the geology and hydrostratigraphy in the vicinity of the IDF in a geologic framework model report (RPP-RPT-59343, 2016). The aquifer underlying the IDF is present in the Hanford formation and possibly the Ringold Unit E formation in the southwest corner. The Ringold Unit E is a fluvially-deposited pebble-to-cobble gravel with a sandy matrix. It is characterized by complex interstratified beds and lenses of sand and gravel with low to moderate degrees of cementation. The contact between Ringold Unit E and the Hanford formation is important because the saturated hydraulic conductivity for the gravel-dominated sequence of the Hanford formation is typically multiple orders of magnitude higher than the more compacted and locally-cemented Ringold Unit E. From a PA standpoint, flow rates through the Ringold Unit E will be slower, therefore, concentrations of contaminants would be higher relative to the Hanford formation. The Hanford formation has a calibrated hydraulic conductivity of 17,000 m/day (56,000 ft/day) and of the Ringold Unit E has a calibrated hydraulic conductivity of 5 m/day (16 ft/day) (RPP-CALC-61032, 2017). Uncertainty exists as to the delineation of the contact between the Hanford and Ringold formation because it is based on a limited number of boreholes, especially in the west and southeast portions of the IDF study area (see Figure 4-17).

DOE stated that groundwater flow in the vicinity of the IDF has been extensively studied since the 1940s through routine monitoring and aquifer tests and through multiple numerical modeling efforts since about the 1990s. The successive numerical modeling efforts integrated accumulated site knowledge to achieve the best possible representation of groundwater flow. The Central Plateau Groundwater Model (CPGWM) (RPP-CALC-61016, 2017) provided the basis and framework for saturated flow at the IDF (RPP-CALC-61032, 2017). Historical operations at the Hanford Site discharged large amounts of water creating a mound in the water table. Presently, with better control of surface run-off and less operational discharges, the water table has been relaxing.

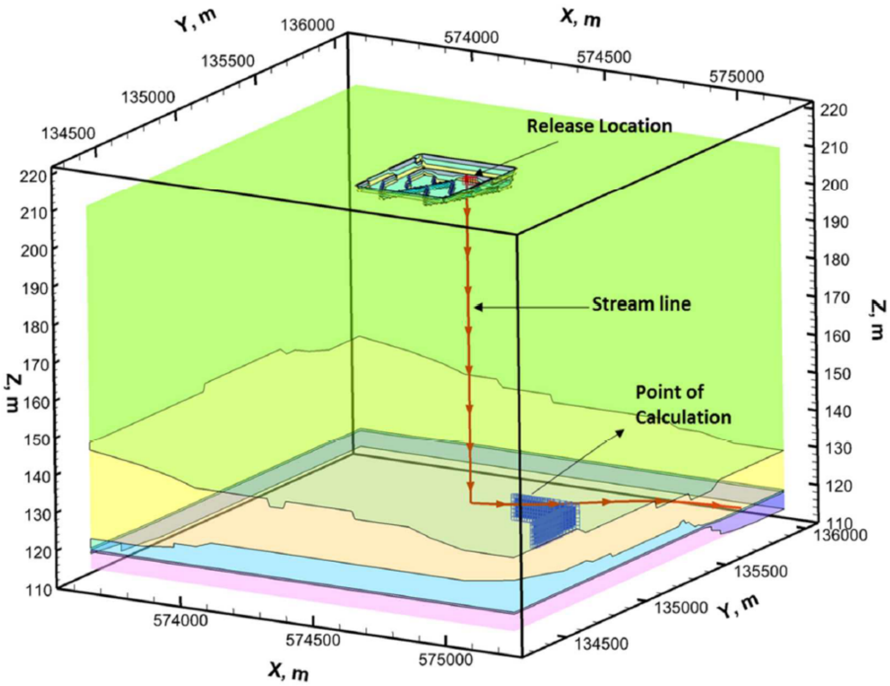


Figure 4-16 Three-Dimensional Representation of Flow and Transport in DOE's Model
[Figure 2-17 from (RPP-CALC-61032, 2017)]

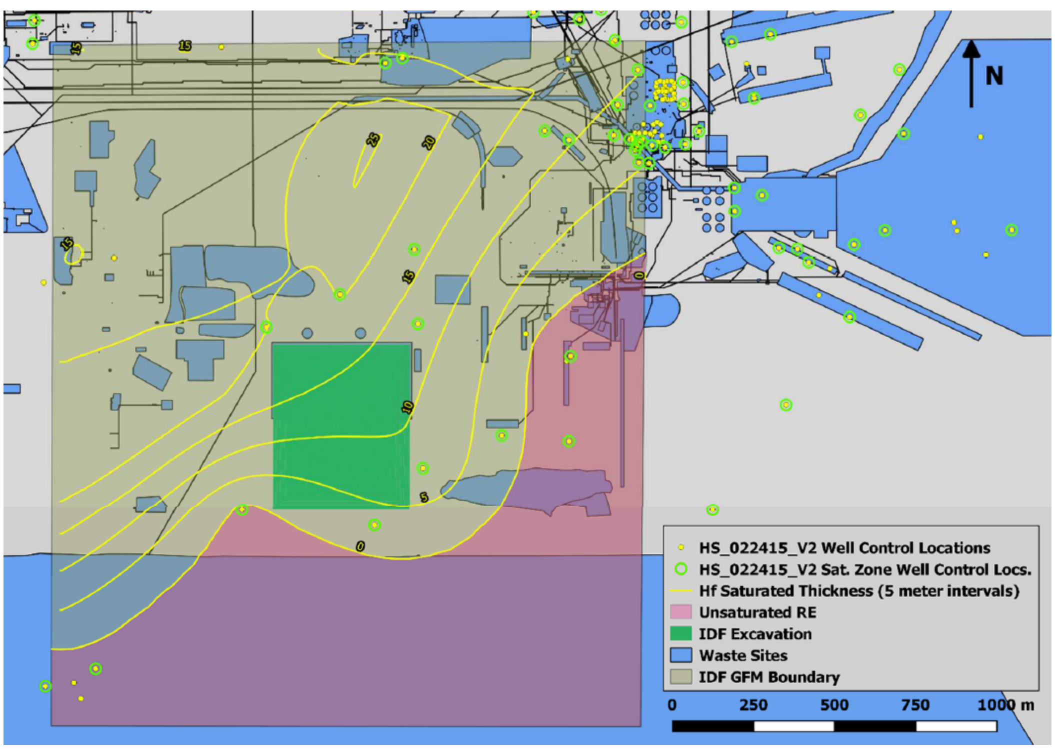


Figure 4-17 Distribution of the Hanford and Ringold Units with Boreholes
[Figure 3-52 from (RPP-RPT-59958, 2018)]

Specific discharge (the product of gradient and hydraulic conductivity) is important for contaminant transport and has also been relaxing. Near the IDF, there is essentially no vertical hydraulic gradient ($2x10^{-5}$ m/m) because of the very high hydraulic conductivity of the Hanford formation. At the IDF, DOE estimated the specific discharge to be approximately 100 m/yr (330 ft/yr). Transport times in the aquifer of conservative species from the release point to the potential groundwater well of a receptor are rapid but are insignificant from a PA standpoint.

DOE described the steps used to perform a simulation of the saturated zone with STOMP (RPP-CALC-61032, 2017). To perform the analyses, many steps were required for the different simulations. The analyses required integration of results from other reports and analyses such as for infiltration and release from wasteforms. When simulations were performed for the saturated zone (without the unsaturated zone), the effects of delay and dispersion in the unsaturated zone were ignored. This was done to provide steady-state inputs to the saturated zone to better isolate the impacts of changes to the saturated zone. DOE performed technical checking of each model simulation and associated pre-processing or post-processing by an independent checker(s) who did not participate in those steps for the relevant simulation. The independent checker resolved identified issues and provided documentation of the checking.

DOE used both a three-dimensional model representation in STOMP and a one-dimensional model representation in GoldSim (RPP-RPT-59958, 2018). The three-dimensional calculations were much more computationally intensive requiring 8-14 hours for a saturated zone-only simulation or 1-2 weeks for a combined unsaturated/saturated simulation. The one-dimensional model was used to perform uncertainty analyses, which would have been impractical with the STOMP model. DOE stated that results from the three- and one-dimensional models were compared and showed good agreement. Groundwater concentrations at the edge of the 100-m buffer zone were controlled by dilution of the unsaturated zone mass flux by the saturated zone (groundwater) flux, which is the product of the hydraulic gradient and the hydraulic conductivity.

For saturated zone sensitivity cases, the initial conditions for pressures were obtained from restart files from historical flow simulations. Arbitrary initial conditions were equilibrated with the specified boundary conditions over thousands of years to achieve steady initial conditions before the restart time of calendar year 2006. The sensitivity cases evaluated the impact from changes in model inputs for saturated zone layers, waste location, and contaminant release rates on contaminant concentrations in a 5-m vertical interval at the 100-m buffer boundary. The saturated zone-only cases were intended to aid in the selection of parameters for the combined saturated/unsaturated simulations.

The travel time in the saturated zone over 200 m (660 ft) was calculated to be about 0.3 years (compared to a travel time, depending on the case and parameters, of approximately 1,800 years through the unsaturated zone). The most risk-significant process in the saturated zone is dilution. In sensitivity cases, the amount of dilution was demonstrated to be directly proportional to changes in saturated hydraulic conductivity (the hydraulic gradient was held constant). The saturated zone sensitivity cases were useful to examine the placement of different wastes in the IDF. If the plumes from concentrated (or poorly performing) wasteforms overlap, then concentrations of contaminants are larger, whereas if the wasteforms are placed throughout the facility concentrations can be reduced.

4.9.2 NRC Evaluation of Flow and Transport in the Saturated Zone

The NRC previously reviewed flow and transport in the saturated zone for application to WMA C (NRC, 2020b). DOE used essentially the same models and modeling approaches for modeling saturated zone flow and transport for the disposal of waste at IDF. This section provides an overview of the NRC staff's review and places emphasis on differences and key observations.

The NRC staff reviewed the DOE documentation, supporting references, and computer model files and performed select independent checking and independent calculations. DOE clearly described the analyses and the information provided was transparent and traceable. DOE provided adequate documentation that proper QA procedures were used and implemented. It is good practice to make available any records that demonstrate independent checking was performed (by whom, what, when).

The DOE conceptual model was an appropriate interpretation and representation of the available information. Sensitivity and uncertainty analyses were used to explore uncertainties. The dilution that occurs in the saturated zone from the mixing with vertical influxes from the unsaturated zone is a very important safety function. The NRC staff agrees with DOE's assessment that the magnitude of the reduction in concentrations is likely to be large and the effect is likely to be durable. An uncertainty associated with the aquifer dilution safety function is the potential for changes in surface water usage and application. Historical operations at Hanford resulted in a large mound in the water table in the area where the water was being discharged. With less discharges and improvements to run-off control, the water table mound has been relaxing over time. Agriculture, particularly in arid or semi-arid regions, can use large amounts of water. Changes to water usage have the potential to change the direction and magnitude of contaminant velocities and to effect concentrations.

Transport times in the saturated zone from the release point to the receptor location for a conservative species were estimated to be very short, in general, less than one year. Transport time in the saturated zone could be more risk-significant for strongly sorbing species but those species are also likely to be strongly sorbing in the unsaturated zone, which has much lower transport velocities, making the saturated zone sorption safety function redundant.

In the NRC staff's RAI, NRC staff stated that multiple changes to the saturated zone hydraulic conductivity estimates that have occurred over time may demonstrate that the values used in deterministic analyses may not be reliable (NRC, 2020a). In response, DOE acknowledged that there is uncertainty in the hydraulic conductivity of the aquifer sediments underneath the IDF, therefore, they included an evaluation of this uncertainty in the IDF PA uncertainty analysis. DOE indicated that the change in hydraulic conductivity estimates over time was driven by field measurements and improvements to the model calibration process. Though hydraulic conductivity estimates have changed over time, the specific discharge values have changed less and have been consistent. DOE has activities planned to address uncertainties in saturated zone-specific discharge values. DOE has completed activities to provide additional confidence in the hydraulic conductivity values used. Confidence-building activities to verify the representativeness of the hydraulic conductivity of the saturated sediments beneath the IDF include (1) pumping tests conducted in 2015; (2) the calibration of hydraulic conductivity parameters in the groundwater models using a tritium plume, and (3) a drawdown test performed on two wells near the IDF (DOE, 2021c). The CPGWM has been updated multiple times since completion of the IDF PA.

The information DOE provided fulfilled the RAI and was sufficiently detailed. Because risk is driven by specific discharge, it is recommended that future analyses emphasize the product of the two parameters rather than each parameter individually because they are not independent. It is expected that data may change over time, especially as more research is conducted. A critical evaluation of the changes in data over time can identify when an estimate is reliable and robust. If a parameter is changing with more research or analyses, then it is much harder to argue that the "correct" value has been determined.

In the NRC staff's RAI, staff discussed the information associated with geologic uncertainty (NRC, 2020a). There are limited boreholes to define the extent of the Ringold E unit under the IDF footprint. Previous analyses had included the Ringold E, while the most recent analyses had eliminated the Ringold E. Borehole characterization is the main technique that was used to delineate the contact locations between different hydrostratigraphic layers across the Hanford Site. The boreholes nearest to the IDF are the most important boreholes to define the local geology. The boreholes near the IDF used for each analysis did not change - only the analytical methods to interpret and model the data changed. DOE provided a detailed summary of the extensive revisions that have occurred. At least two errors were identified and corrected. As science evolves and new methods are developed, better results may be produced, but the correctness of the outcome cannot be determined in the absence of validation. This is extremely difficult and expensive when it comes to geologic uncertainty. A better approach is to include the uncertainty in the model explicitly or to take a pessimistic approach. Figure 2-14-9 of DOE's response to the NRC staff's RAI is an excellent representation of the uncertainty associated with reinterpretation of hydrologic and geologic information (DOE, 2021c). DOE should further examine the geologic uncertainty associated with the Ringold E Unit. If characterization data is not sufficient to make a confident interpretation, then a pessimistic representation should be taken (Recommendation #17).

4.9.3 NRC Conclusions on Flow and Transport in the Saturated Zone

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on flow and transport in the saturated zone. The NRC staff previously reviewed flow and transport in the saturated zone for application to WMA C (NRC, 2020b). Staff has the following conclusions on flow and transport in the saturated zone applied to the IDF:

- The saturated zone is likely to provide significant dilution of contaminant concentrations (fluxes) entering the aquifer from the unsaturated zone.
- DOE's description of the analyses performed of flow and transport in the saturated zone for the disposal of VLAW at IDF was clear, transparent, and traceable.
- DOE's model estimates of travel times in the saturated zone of less than one year were consistent with data.

The NRC staff has the following recommendation associated with flow and transport in the saturated zone:

- DOE should further examine the geologic uncertainty associated with the Ringold E Unit. If characterization data is not sufficient to make a confident interpretation, then a pessimistic representation should be taken (Recommendation #17).

4.10 Biosphere and Dose Assessment

The biosphere is the environment where humans and biota may be exposed to radiation that is released from the disposal facility. The dose assessment is the analyses to convert the released radiation into a dose to a receptor (human). Multiple pathways can contribute to release and many different pathways for exposure or uptake of radiation can be active in a site-specific dose assessment. Typically, a few pathways contribute to most of the estimated doses. The following sections describe DOE's analyses of the biosphere and conversion of released radiation into hypothetical doses (dose assessment) and the NRC staff's review of the information. DOE's approach to representing the biosphere and completing the dose assessment for disposal of VLAW at the IDF was very similar to that used for WMA C, that the NRC staff reviewed (NRC, 2020b). In the sections below, the NRC staff provides an overview and discusses any new issues or observations.

4.10.1 DOE's Analyses of the Biosphere and Dose Assessment

DOE described an exposure pathway as the physical course that a contaminant takes from the point of presence in a specific environmental media to a receptor. The route of exposure is the path through which a contaminant enters a receptor. An exposure scenario included data and exposure parameters that described how exposure occurs.

During the institutional control period, DOE assumed the receptor would reside 20 km (12 mi) from the edge of the disposal facility. After the institutional control period, the receptor was assumed to reside 100 m (328 ft) from the edge of the facility excavation. DOE requirements in DOE Order 435.1 are similar to NRC requirements in 10 CFR 61.41. An all-pathways farmer scenario was implemented to calculate the total effective dose equivalent (TEDE) for comparison against a dose limit of 0.25 mSv (25 mrem). In DOE's approach, the dose limit was the TEDE in a year from all exposure pathways, excluding the dose from radon and progeny in air. Calculations were performed based on predicted radionuclide transport through the groundwater and atmospheric pathways, and exposure at the point of contact.

For the groundwater transport pathway, a farmer and his family were assumed to reside 100 m (328 ft) from the edge of the facility excavation. They were assumed to use contaminated water from a well located in the center of the saturated zone plume located 100 m (328 ft) from the edge of the facility excavation. The well was assumed to be located at the point of maximum concentration. Dilution processes associated with drawing water through the well were not considered. The receptor was assumed to be an adult who uses the water for drinking, irrigation of crops, and to water livestock. DOE included ten exposure pathways for the resident farmer. The ingestion pathways included ingestion of water, garden vegetables, beef cattle, milk, eggs, poultry, and incidental contaminated soil. The inhalation pathways included contaminated dust and water vapor. External exposure to radiation was also included. DOE provided the equations used as well as the data (RPP-ENV-58813, 2016). DOE applied 95th percentile intake rates to provide additional conservatism and to maintain consistency with other Hanford Site analyses (RPP-RPT-59958, 2018). DOE checked the reasonableness of the resulting dose conversion factors (i.e., how much dose a unit concentration of a radionuclide would produce) against the results from previous analyses and found the dose conversion factors to have reasonable agreement.

The compliance period used in the analyses was 1,000 years following closure of the IDF. DOE included analyses beyond 1,000 years to calculate the maximum dose and the time of that dose. The analyses of results after 1,000 and up to 10,000 years was included to increase confidence in the modeling outcome. DOE included an additional calculation in the PA analyses to demonstrate that the peak dose from contaminated groundwater would be expected to occur in the first 10,000 years.

Some contaminants were expected to be released to the atmosphere. DOE's analysis of the dose from atmospheric release was similar to the groundwater analyses with appropriate changes to exposure pathways. The air pathway analysis started at the time of closure. DOE assumed that the disposal containers are not airtight and that gaseous contaminants may be released from the partially water-saturated wasteforms and be transported by gaseous diffusion through the backfill to the surface.

DOE used the concept of a representative person for the dose assessment. DOE stated that the all-pathways representative person scenario was consistent with the scenario described in the supporting data package (RPP-ENV-58813, 2018) and with the scenario implemented in the PA for WMA C (RPP-ENV-58782, 2016) that was previously reviewed by the NRC staff (NRC, 2020b). Current guidance from DOE and the International Commission on Radiological Protection (ICRP) recommends the use of a "representative person" to describe a hypothetical member of the public for use in projections of future dose. The representative person is described as a person that is representative of the more highly exposed individuals in the population. The ICRP replaced the concept of an average member of the critical group used in older radiation protection guidance with the representative person concept. The reference person is a hypothetical aggregation of age- and gender-weighted human (male and female) physical and physiological characteristics for the purpose of standardizing radiation dose calculations (DOE, 2011a).

DOE considered the potential for accumulation of two key radionuclides (^{129}I and ^{99}Tc) in soil. DOE estimated that a significant percentage of the dose from ingestion came from animals ingesting contaminated water, rather than the ingestion of contaminated plants or inadvertent soil ingestion. For ^{129}I, increasing the concentrations in soil by a factor of 10 resulted in an increase in dose of only 5 percent. For ^{99}Tc, the increases could be larger but because over 80 percent of the dose from ^{99}Tc comes from ingestion of contaminated water, the increase in dose was insignificant in a PA context. DOE also considered the results from alternative dose models. In one modeling study, DOE assumed 100 years of accumulation along with a K_d for ^{99}Tc of 2.0 mL/g. For comparison, the K_d assigned to the H2 unit was 0.0 mL/g. The estimated increase in dose was between a factor of 6.4 and 7 (HNF-SD-WM-TI-707, 2007). DOE stated that assumptions used in the analyses biased the outcome to higher results, therefore, the alternative model was not implemented in the IDF PA.

4.10.2 NRC Evaluation of the Biosphere and Dose Assessment

The NRC staff previously reviewed analyses of the biosphere and dose assessment for application to WMA C (NRC, 2020b). DOE used essentially the same approach to the biosphere and dose assessment for the disposal of waste at the IDF as was used for WMA C.

This section provides an overview of the NRC staff's review and places emphasis on differences and key observations.

The NRC staff reviewed DOE's documentation, supporting references, and computer model files, and performed select independent checking and calculations. DOE clearly described the analyses that were performed, and the information provided was transparent and traceable. DOE provided adequate documentation that proper QA procedures were used and implemented.

Access to the site is controlled by DOE and no members of the public live on the site. After the institutional control period of 100 years following closure, DOE assumed that the public could access the site. The assumption of a maximum of 100 years of effective institutional control was appropriate and consistent with NRC guidance and the 10 CFR Part 61 regulations. DOE may be present at the site for a longer time period to remediate past contamination. DOE used the dose assessment to take the estimated fluxes of contaminants to the environment and convert them into projected radiological impacts to receptors.

The two pathways for contaminant release evaluated by DOE were the transport of contamination through groundwater to an offsite receptor and the atmospheric release of gaseous radionuclides. These two pathways were reasonable and complete. Surface water bodies are not anticipated to be present in the vicinity of the IDF. The Columbia River and other surface water bodies are present but are very distant from the IDF. More distant potential exposure points at the surface water bodies would result in lower doses than those evaluated by DOE. Disruptive events that could result in a new exposure scenario are not anticipated. DOE appropriately considered the potential for seismic, volcanic, and other disruptive events and did not identify any that were likely to result in a more risk-significant exposure scenario than those evaluated. Erosion is a disruptive process that could potentially lead to alternative exposure scenarios. Because of the depth at which waste will be disposed at the IDF, if erosion were to occur, it is unlikely to result in a new exposure scenario. However, erosion could result in enhanced infiltration and changes to the exposure scenarios DOE evaluated. The impacts to hypothetical inadvertent intruders that DOE evaluated are discussed in Section 4.11.

The potential for buildup of contaminants in the biosphere from irrigation or differences in geochemical environments at the point of exposure compared to the point of release (or during transport) is a complex process. DOE provided a clear summary of what was considered and why buildup was not included in the base case of the PA. The effects with respect to ^{129}I are likely to be minor, however, the effects with respect to ^{99}Tc could be more significant. Technetium is usually weakly sorbed under oxidizing conditions but can be strongly retained under reducing conditions. It is recommended that DOE develop a more complete assessment of the impacts of the buildup of ^{99}Tc in the biosphere (Recommendation #18).

4.10.3 NRC Conclusions for the Biosphere and Dose Assessment

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on the biosphere and dose assessment. The NRC staff previously reviewed the biosphere and dose assessment as applied to WMA C (NRC, 2020b) and has the following conclusions:

- DOE considered appropriate exposure scenarios and had technical basis for eliminating other scenarios.
- DOE's description of the analyses performed for the biosphere and dose assessment for the disposal of VLAW at the IDF was clear, transparent, and traceable.
- There were no sources of uncertainty associated with the biosphere and the dose assessment that were unique to disposal of VLAW at the IDF.

The NRC staff has the following recommendations:

- DOE should develop a more complete assessment of the impacts of the buildup of ^{99}Tc in the biosphere. (Recommendation #18)

4.11 Inadvertent Human Intrusion

Inadvertent human intruders are members of the public who may unknowingly use the portion of the site at some time in the future after active institutional controls are no longer being implemented. DOE's assessment of inadvertent human intrusion for disposal of VLAW at the IDF used a very similar analysis method as was completed for closure of WMA C (from a scenario and computation standpoint; the wastes were different). The sections that follow provide an overview of the analyses and focus on differences between the evaluations.

4.11.1 DOE's Analyses of Inadvertent Human Intrusion

DOE developed the inadvertent intruder scenarios and dose calculations to support the development of the WAC for the IDF and to assess compliance with the requirements for protection of the inadvertent intruder. DOE described the conceptual model for the intruder analysis and the associated mathematical model that was implemented in GoldSim in report RPP-CALC-61254 (RPP-CALC-61254, 2018). Doses were calculated for four inadvertent intruder scenarios, represented by one acute exposure scenario (40 hours over a 5-day period) and three chronic exposure scenarios. The intruder analysis used performance measures of 100 mrem (1 mSv) in a year for the chronic scenario and 500 mrem (5 mSv) TEDE (excluding radon) in a year for the acute scenario.

For the purposes of the DOE analyses, and to be consistent with NRC guidance, DOE used an active institutional control period of 100 years. Although 100 years of institutional control was assumed, DOE indicated that the likelihood for an inadvertent intrusion at that time would be small for a variety of reasons. DOE expects to be present at the site for longer than 100 years to remediate groundwater and contaminated soil. DOE considers that land used for waste management is a permanent commitment such that DOE will remain present for at least the next 150 years. In addition, various Records of Decisions (ROD) in the area require long-term institutional control because of the presence of long-lived contamination. DOE described different actions that have or will be undertaken to ensure that institutional controls are durable. DOE will install markers to warn people of the presence of dangerous materials. Site information will be provided through various means (e.g., library records, internet). The engineered cover will be large and will not have the appearance of a natural feature, thereby alerting future generations to some previous actions by man. At the apex, the cover is expected to be 15 m (50 ft) above the land surface. Under current conditions, the land is not viewed as favorable for human development.

Consistent with current regional practices, DOE assumed that inadvertent intruders would engage in normal behaviors associated with a rural lifestyle. DOE considered two acute scenarios: installation of a well and building of a residence with a basement. Because of the depth that waste will be disposed at the IDF, the building of a residence would not contact the waste, therefore, this scenario was not carried forward for analyses. DOE evaluated the dose received during the drilling of the well and subsequent exposure to waste extracted in the drill cuttings (acute scenario); exposure was evaluated over a short time (i.e., 40 hours). DOE evaluated chronic scenarios of the post-intrusion dose received from spreading the drill cuttings on the land surface, after which an individual was assumed to live or work on that area. In the intruder analyses, DOE credited institutional controls for the first 100 years following closure of the disposal facility. During this time, intrusion cannot occur because of the active controls that will be in place.

Much of the waste that will be disposed in the IDF (e.g., the vitrified glass wasteform) will be contained in a stainless-steel package. DOE did not take any credit for the steel packages because the waste containers will degrade over time. DOE discussed that it is possible that the packages may degrade slowly such that a well driller would recognize the presence of non-native material and take investigative actions. DOE assumed a well will be installed to the depth of the water table. As the well is drilled through the IDF, the disposed waste would be intercepted and brought to the ground surface in the form of drill cuttings, assuming the drilling occurred at the location where a waste container or containers had been placed. DOE did not include doses to the intruder from consuming contaminated water. A summary of the intruder scenarios DOE considered is provided in Table 4-4.

DOE's analyses of the potential impacts to inadvertent intruders included 43 radionuclides as well as daughters, where applicable. The waste streams considered by DOE in the IDF PA were those expected to be generated by the vitrification process (VLAW, spent glass melters, solidified (non-debris) and encapsulated (debris) SSW, solidified ETF-treated LSW) as well as other waste streams that were not a result of the vitrification process. For the inadvertent intruder dose calculations, DOE assumed that VLAW was co-located with spent melters, ETF-LSW was disposed in a separate area of the IDF, and that all SSW was disposed together but segregated from the VLAW and ETF-LSW. With the inadvertent intruder model, DOE simulated an intrusion into three separate waste streams: VLAW glass, ETF-LSW, and SSW. DOE evaluated intruder doses using the average concentration of radionuclides in SSW calculated by dividing the total inventory by the total disposed volume.

DOE provided the NRC staff detailed descriptions of the calculations, data used, and technical basis for the analyses. DOE also provided the GoldSim model files that showed how the calculations were implemented. Key technical information used in or resulting from the analyses included:

- For the external exposure pathway, DOE used dose conversion factors assuming the contaminated soil that results from the waste that will be extracted by the driller is infinitely thick.
- A soil mass loading value of 6.66×10^{-5} g/m^3 was assumed.
- For chronic scenarios, contamination was assumed to be spread uniformly over the land area associated with the scenario.

Table 4-4 Intruder Scenarios Considered by DOE

Scenario	Description
Acute Exposure: Well Driller	Dose is the result of drilling through the IDF. Exposure routes include external exposure, inhalation of soil particulates, and incidental soil ingestion. Exposure occurs during the drilling operation while in contact with the drill cuttings. Resulting dose does not depend on the borehole diameter.
Acute Exposure: Basement Intrusion	Dose is considered highly unlikely due to the thickness of the closure cap. A basement excavation would not disturb the waste. No further discussion of this scenario is included.
Chronic Exposure: Rural Pasture	Dose is the result of drilling a well that serves a rural pasture. Contaminated drill cuttings are mixed with the soil over the pasture area. Exposure routes include external exposure, inhalation of soil particulates, incidental soil ingestion, and milk consumption.
Chronic Exposure: Suburban Garden	Dose is the result of drilling a well that serves a suburban garden. Contaminated drill cuttings are mixed with the soil over the area where a residence and a garden are constructed. Exposure routes include external exposure, inhalation of soil particulates, incidental soil ingestion, and fruit and vegetable consumption.
Chronic Exposure: Commercial Farm	Dose is the result of drilling a well that serves a commercial farm. Contaminated drill cuttings are mixed with the soil over the commercial farm area. Exposure routes are external exposure, inhalation of soil particulates, and incidental soil ingestion.

4.11.2 NRC Evaluation of Inadvertent Human Intrusion

The NRC staff previously reviewed human intrusion analyses for application to WMA C at the Hanford Site (NRC, 2020b). DOE used essentially the same approach to analyses of potential human intrusion for the disposal of waste at IDF as was used for WMA C. This section provides an overview of the NRC review and places emphasis on differences and key observations.

The NRC staff reviewed DOE's documentation, supporting references, and computer model files, and performed select independent checking and calculations. DOE clearly described the analyses that were performed, and the information provided was transparent and traceable. DOE provided adequate documentation that proper QA procedures were used and implemented. The results of the human intrusion calculations are discussed in Section 4.15.1 for comparison with NRC's performance objective.

The scenarios DOE considered for the inadvertent intruder were appropriate. The calculations and data used were described and suitable for the application. For inadvertent human intrusion at a waste disposal site, the potential for construction of a home with a basement is a common scenario that is evaluated. The depth of basement construction from the land surface is typically 3-4 m (10-13 ft). However, the minimum thickness from the top of the engineered cover to the waste is more than 10 m (33 ft) in DOE's current design. Therefore, DOE concluded, and the NRC staff concurs that excavation to construct a residence is not a credible intruder scenario at the IDF.

In the IDF PA, DOE used a 100-year institutional control period for disposal of waste at the IDF and presented information to support the use of this duration. In response to NRC's RAI question on waste variability, DOE referenced an institutional control period lasting until 2278, which translates to 227 years, assuming the baseline did not change (DOE, 2021c).

Considering all factors, a longer period of institutional control may be suitable for discrete areas at the Hanford Site where the risk is driven by short-lived radioactivity. However, it is difficult to accurately forecast the human component of the decision analyses for the length of the institutional control period. Will a decision-maker continue to fund active institutional controls or understand the implications of NOT continuing to fund the controls? For commercial waste disposal facilities, NRC requires a durable funding source (financial assurance) to ensure the active controls will be provided. The durable funding source provides the necessary resources for the whole institutional control period. Passive controls like deed restrictions, markers, and state or federal government ownership of the land are also required. Given the defense-in-depth approach to land usage and control, it is likely that active and passive controls are effective and future intrusion does not occur. Given the uncertainty in long-term human actions, a cautious approach is warranted. DOE should clarify in the final WIR evaluation what institutional control period is being used. If institutional controls are assumed to fail after 100 years, the waste variability analyses provided in DOE's RAI response demonstrates that the waste acceptance criteria (concentrations) would likely be exceeded unless other mitigating factors or approaches are considered. DOE should consider waste variability when establishing waste acceptance criteria (Recommendation #19).

DOE described the spatial scale and shape of the disposal facility (cover) as well as the environment where the facility is located as being a deterrent to future intrusion. Whether future generations will recognize the cover as the top of a disposal facility will depend in part on the final design and markers that are placed. A naturalized cover design may be more difficult to recognize as being man-made. A sparsely populated area will have a lower probability that intrusion will occur in the future. Humans could settle in the vicinity of the disposal site but installation of a well would be at a discrete point that may or may not intersect buried waste. It was appropriate for DOE to consider the disposal facility design and the environment where it is located when considering the likelihood of future human intrusion.

In the NRC staff's RAI, staff asked for additional information on the use of the average waste concentrations in the intruder assessment (NRC, 2020a). The NRC staff asked DOE to provide the range of dose impacts to an inadvertent intruder from each waste stream disposed in the IDF. In response to the RAI, DOE discussed the analysis process and the approach used to develop the WAC (DOE, 2021c). Because it is envisioned that containers will be stacked up to four containers high, some individual containers could exceed allowable concentrations in the WAC if others in the stack were below allowable concentrations.

DOE also evaluated the potential variability in the wastes that would be processed (DOE, 2021c). Figure 4-18 was provided to show the monthly average concentrations of LAW that are expected to be processed. Much of the variability is driven by ^{90}Sr. DOE did not address the variability in concentrations of radionuclides that will occur in secondary waste packages resulting from the processing that is performed to produce glass. The secondary solid waste (SSW) includes different wastes that perform different functions (e.g., GAC, HEPA filters). These wastes will have different concentrations of radionuclides. The wastes are likely to be packaged in different containers (not mixed in a container) and disposed in discrete locations. DOE's use of average values calculated based on the total activity and volume of secondary waste is not sufficient to identify wastes that, if disposed, could be difficult to demonstrate will meet 10 CFR 61.42. For commercial LLW disposal, comparison to the performance objective for the intruder (i.e., 10 CFR 61.42) is usually done with an average waste concentration but with constraints on variability provided in the Branch Technical Position on Concentration

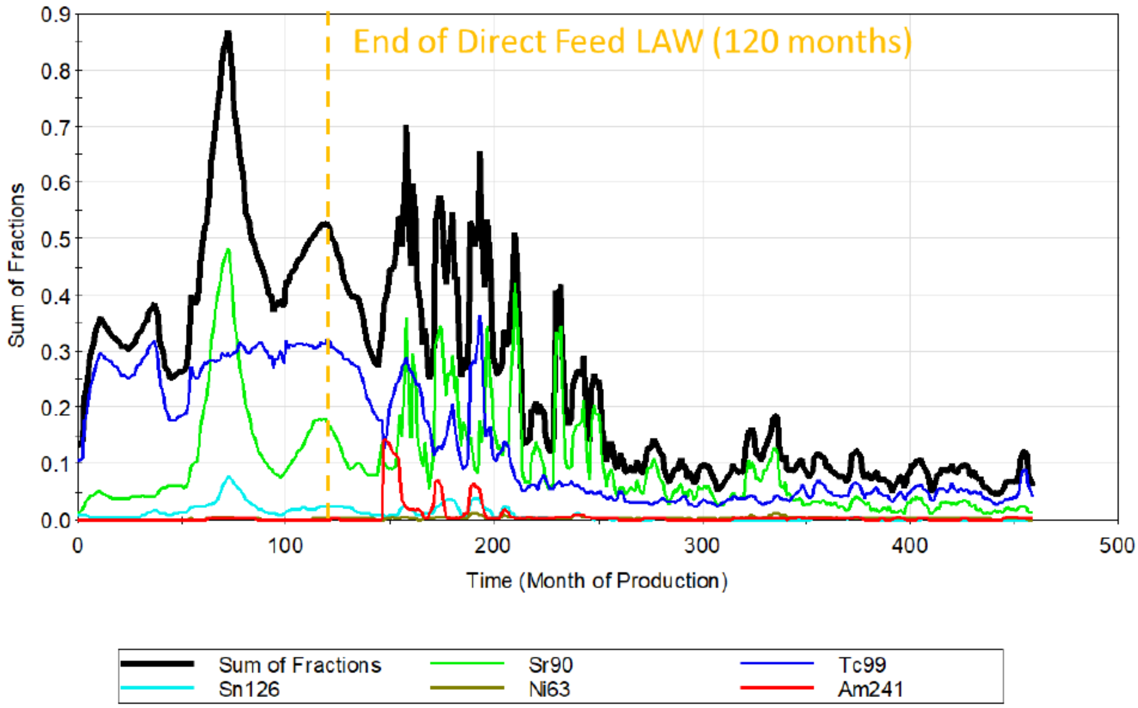


Figure 4-18 Variability in Waste Processed with VLAW
[Figure 2-17-1 in (DOE, 2021c)]

Averaging (NRC, 2015). It was not clear to the NRC staff that DOE would be providing constraints on waste variability in a revision to the WAC. Understanding the potential range of intruder doses that could result from different waste types within SSW would allow for risk-informed decision-making. Though waste containers were not credited in the analyses, if problematic SSW containers were identified, they could be emplaced under stainless-steel containers (e.g., to prevent drilling into the problem waste) that may result in future real-world risk reduction. It is recommended that DOE evaluate the potential impacts of different concentrations of radioactivity in different types of SSW rather than only considering an average over all SSW (Recommendation #20).

The recommendations provided in the review of WMA C and associated with various parameters used in the inadvertent human intrusion assessment should be considered in future revision to the IDF PA for disposal of VLAW and the final WIR evaluation.

4.11.3 NRC Conclusions on Inadvertent Human Intrusion

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on the inadvertent human intrusion, which the NRC staff previously reviewed as applied to WMA C (NRC, 2020b). Staff has the following conclusions:

- DOE provided clear documentation and thorough analyses of potential human intrusion of VLAW disposed at the IDF. The analyses were reasonable and appropriate for the application.

- It was appropriate to eliminate the intruder-excavation scenario from further analyses because the minimum thickness from the top of the engineered cover to the waste is more than 10 m (33 ft) in DOE's current design.
- An institutional control period of 100 years is consistent with NRC guidance and is appropriate for disposal of VLAW in the IDF.

The NRC staff has the following recommendations:

- DOE should clarify in the final WIR evaluation what institutional control period is being used. If a 100-year institutional control period is used, the waste variability analyses provided in DOE's RAI response demonstrates that the waste acceptance criteria (concentrations) would likely be exceeded unless other mitigating factors or approaches are considered. DOE should consider waste variability when establishing waste acceptance criteria. (Recommendation #19)
- DOE should evaluate the potential impacts to inadvertent human intruders of different concentrations of radioactivity in different types of SSW rather than only considering an average over all SSW. (Recommendation #20)

4.12 Model Support

Model support is essential for all modeling projects, but it is especially important for complex analyses used to provide projections over long timeframes. Verification and validation are complementary but different tasks. Whereas verification is determining that the equations were solved correctly, validation is determining that the correct equations were solved. The latter task is much more difficult. For PAs that are used to evaluate complex systems and estimate potential impacts (doses) to human receptors well into the future, traditional model validation is not possible. Performance assessment models must be supported by multiple, diverse sources of information. Even with strong supporting information, there may be remaining uncertainty. Performance assessment models are collections of other models (e.g., process models), where inputs and the effects (outputs) of the process models are integrated. Even though the overall PA model may not be validated in the traditional sense, some of the individual process models may be validated. NRC staff reviewed the information DOE provided on verification and support for the models and analyses completed for the performance assessment of the IDF. The sections that follow provide DOE's model support of the PA for the disposal of VLAW at the IDF and the NRC staff's review of the information.

4.12.1 DOE's Model Support

DOE indicated that the IDF PA was based on decades of scientific investigations conducted by Hanford researchers. The results of these scientific investigations have been published in numerous reports that formed the basis for the conceptual models and parameters used in the IDF PA. The reports used, arranged by model components of the IDF system, were summarized in Table 8-1 of the IDF PA (RPP-RPT-59958, 2018). Many additional reports supported the documents listed in Table 8-1. The data packages were utilized in the development and qualification of models used to describe and forecast the performance of the IDF. The models were prepared, checked, and reviewed under DOE's QA program. In addition, additional research is ongoing under the PA maintenance program.

DOE used PA 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 PA 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 PA models cannot be validated in the traditional manner of other numerical models, PA models must have adequate support of the results for the model's intended purpose.

DOE presented the information used to develop the modeling in the very extensive IDF PA (RPP-RPT-59958, 2018). Technical studies have been completed for decades at the Hanford Site on a wide range of topics (e.g., infiltration, waste release, hydrology), including studies to evaluate the performance of systems (e.g., engineered cover performance, unsaturated zone hydrology). DOE is also planning to complete additional studies, such as glass lysimeter studies. To date, most of the technical work has been used to develop parameter values for inputs to the various models rather than to develop confirmatory information supporting the results of the PA models or the underlying conceptual models.

In the NRC staff's RAI, staff stated that DOE provided limited support that the conceptual models were implemented appropriately with the numerical models in the PA as well as limited support that the model projections would likely bound anticipated impacts (NRC, 2020a). The NRC staff stated that DOE should provide support for the key intermediate results of the numerical modeling, such as for observed transport rates of contaminants such as ^{90}Sr. DOE stated, in response to the NRC staff's RAI, that an important part of the model confidence-building activities was to compare the model assumptions and predictions developed in the 2017 IDF PA to model assumptions and predictions developed in previous analyses that had been subject to regulatory review and acceptance (DOE, 2021c). DOE referenced statements made by regulators with respect to acceptance of previous analyses. Model confidence (i.e., model support) was also provided by comparing the intermediate results of the numerical modeling to relevant observations. DOE provided information on:

- **Secondary Minerals Formed During Glass Corrosion** - Model support for the selection of the base case SMRN adopted in the PA was described in PNNL-21812 and references cited therein (PNNL-21812, 2013). For example, a geochemical model was used in those investigations to infer the existence of secondary minerals that formed in Product Consistency Tests (PCTs) at 90°C. This confirmatory identification was based on comparisons between model predictions with observed changes in solution chemistry as a function of an assumed amount of glass corrosion. This provided guidance for the selection of the SMRN adopted in the IDF PA to simulate glass corrosion and calculate fractional release rates (FRRs). In addition, research studies conducted at Pacific Northwest National Laboratory (PNNL) and the Vitreous State Laboratory (VSL) used different techniques (e.g., x-ray diffraction, scanning electron microscopy, and energy dispersive X-ray spectroscopy) to identify the secondary minerals that are formed during corrosion experiments. A related confidence-building activity in the IDF PA involved comparisons of calculated FRRs that were calculated using GWB and STOMP, two independent computational models. DOE stated there was general good agreement obtained between GWB and STOMP results.

- **Water Contacting Wasteform** – DOE stated that it assumed in the IDF PA that all the water that percolates through the surface cover enters the disposal trenches and can contact the wasteform (DOE, 2021c). DOE assumed that some fraction of the water seeping through the backfilled trenches can enter the fractured glass, grouted SSW, or Effluent Treatment Facility (ETF)-liquid secondary waste (LSW) wasteforms depending on the hydraulic characteristics of the wasteform. The remaining fraction of the percolating water was assumed to flow in the backfill around the wasteforms. DOE indicated that there were no direct observations of the moisture content in the wasteform or backfill with which to compare the model-predicted values. Because there was uncertainty in the hydraulic characteristics of the wasteforms and the backfill, DOE conducted sensitivity analyses to determine the extent to which the water contacting the waste impacted the release rates and associated performance.
- **Vadose Zone Transport** – DOE developed support for the natural system models used in the IDF PA from different observations of fate and transport of water and contaminants in the vadose zone and saturated zone in the Central Plateau of the Hanford Site. One such comparison was that of observed and predicted ^{99}Tc concentrations associated with releases from a past leak in the 241-C Tank Farm (RPP-RPT-59197, 2016). The comparison confirmed that the vadose zone fate and transport model could predict ^{99}Tc transport time to the water table in the decades following the release event. DOE also compared observed and predicted vadose zone transport in areas of past discharge of liquid wastes to cribs and trenches in the Central Plateau. For example, DOE documented comparisons of observed and model-predicted ^{99}Tc concentrations near the BC cribs and trenches (PNNL-14907, 2004). DOE stated that observed contaminant migration in areas of past liquid discharge can provide support for the vadose zone fate and transport models used in the IDF PA. DOE indicated it is important to caveat the general conclusion regarding the use of operational data and observations by recognizing that the vadose zone flow regime beneath past liquid discharge sites is characterized by focused high volumes of liquid discharged as well as background net infiltration rates that are much greater than the long-term average steady-state recharge rate in the area near the IDF.

  The moisture content predicted using the vadose zone flow and transport model is another example of an intermediate result that was used to support the model. The moisture content of sediments in the vadose zone had been measured in boreholes drilled in undisturbed and disturbed areas near the IDF and comparisons of observed and predicted vadose zone moisture contents in undisturbed and disturbed areas near the IDF were presented.

  DOE indicated that it would be possible to use the vadose zone flow and transport model to replicate past observations of ^{90}Sr transport at the Hanford Site. However, the amount of ^{90}Sr released and the amount of liquid involved in the discharge(s) may be very uncertain and difficult to quantify. DOE also indicated that the releases of ^{90}Sr from a covered waste facility under very low flow conditions is unlikely to be analogous to the conditions for past releases of ^{90}Sr.
- **Saturated Zone Dilution** - The amount of dilution afforded by the saturated zone beneath the IDF is a function of the specific discharge (i.e., the hydraulic conductivity times the hydraulic gradient) for the groundwater regime beneath the IDF. DOE stated that the hydraulic conductivity in the high-conductivity zone beneath the IDF has been inferred from hydraulic tests as well as from the calibrated hydraulic properties in the Central Plateau groundwater flow model (RPP-RPT-59958, 2018). The current Central Plateau model was

used to make comparisons of model-predicted and observed plumes of tritium and ^{129}I to provide confidence in the model results. DOE stated that additional PA maintenance activities are planned to confirm the saturated zone model and parameters used in the IDF PA.

Prior to the development and application of the models used in the IDF PA, DOE identified additional activities to support DOE's strategy and plan for future verification activities. These activities were presented in the IDF PA Maintenance Plan (CHPRC-03348, 2019). The activities identified in the maintenance plan are those developed to confirm and improve the technical basis for the key assumptions made in the IDF PA. The key assumptions and uncertainties were summarized in Section 2.8 and 8.4 of the IDF PA (RPP-RPT-59958, 2018). The maintenance activities identified in the maintenance plan were categorized by technical area and then into two groups of research and development activities, including: 1) activities planned to continue the evaluation of assumptions related to the design basis used for the IDF PA and the related scientific studies of wasteform and site characteristics, and 2) activities planned to conduct focused testing on key assumptions related to the conceptual models and parameter values used in the forecasts of the IDF performance. Several of the research and development activities related to the NRC staff's RAIs on the Draft WIR Evaluation for VLAW and the IDF PA (NRC, 2020a).

4.12.2 NRC Evaluation of Model Support

The NRC staff previously reviewed model support for application to WMA C at the Hanford Site (NRC, 2020b). DOE used a similar approach for the disposal of waste at the IDF. Model support is one of the most risk-significant aspects of a PA. The NRC staff reviewed DOE's documentation, supporting references, and computer model files, and performed select independent checking and calculations. Staff also reviewed technical reports and literature not directly referenced in the IDF PA, such as incident reports from operational history, to compare against the information DOE provided. Model support is one of five general technical review procedures identified in NRC's guidance document for reviewing waste determinations (NRC, 2007). The goal of the review procedures is to ensure that the output from DOE's PA can be supported by comparison to independent data. The specific review procedures include:

- Verification that DOE has compared the results with an appropriate combination of site characterization and design data, process-level modeling, laboratory testing, field measurements, analogs, and independent peer review
- Examination of the output from the mathematical models for consistency of the response with the expected response, given the conceptual model description
- Verification that the PA is reasonably supported by observations from the site, if available
- Use of independent analyses to confirm results
- Performing simplified calculations to examine DOE outputs
- Confirming that DOE has identified and implemented adequate procedures to construct and test its mathematical and numerical models

In the bulleted list above, the second and last two bullets are more focused on verification rather than validation. Verification is necessary but only a component of developing model support; both verification and validation must be considered. DOE's verification activities were more mature than the validation activities. Validation activities tend to be more costly and time consuming. The best method to achieve validation is to attempt to refute the results rather than trying to find information that supports the results.

The NRC staff determined that DOE compared the results of intermediate outputs with an appropriate combination of site characterization and design data, process-level modeling, laboratory testing, and field measurements. In the case of long-term climate information, analogs were considered. DOE also considered analogs for some aspects of glass release modeling. The DOE Low-Level Waste Federal Review Group (LFRG) performed an independent review of the DOE analyses, but the depth of that review is difficult to determine from the documentation provided. Because the Hanford Site has a long history, numerous independent reviews have been performed by other stakeholders such as the Washington State Department of Ecology, the State of Oregon Department of Energy, the Confederated Tribes and Bands of the Yakama Nation, the Confederated Tribes of the Umatilla Indian Reservation, and the Nez Perce Tribe. Information provided and discussions with these stakeholders was very useful to the NRC staff.

DOE clearly described the technical basis for the analyses that were performed, and the information provided was transparent and traceable. DOE provided adequate documentation that proper QA procedures were used and implemented. The information flow between the different models used to perform the calculations was summarized. DOE described the conceptual models and the basis for them, the numerical and mathematical models, the results of process models, and the integrated system model. DOE also described the uncertainty and sensitivity analyses that were performed.

DOE described the technical basis for the IDF PA in Section 8.1.2 (RPP-RPT-59958, 2018). The various sections of the IDF PA collectively provided the information necessary to evaluate DOE's support for the analyses. Because DOE did not provide a specific section dedicated to model support or a stand-alone document the review was challenging for the NRC staff. The documentation for the IDF PA and the Hanford Site includes numerous reports. The regulatory review to identify supporting or conflicting information for numerous different technical areas was a complex task. DOE did an excellent job of detailing how different models were developed, the sources for the inputs, and descriptions and interpretation of the outputs.

Demonstrating that the outputs are expected to represent future reality (or bound future impacts) is a more difficult question that was not directly addressed by DOE. It should be noted that ensuring safety does not always mean accurately estimating future impacts. A purposefully pessimistic calculation that overestimates potential future impacts is acceptable to the NRC and most regulators and can be a useful way to address uncertainties that may be difficult, expensive, or impossible to reduce. As discussed in Section 4.13, for a complex system it can be difficult to subjectively assess pessimistic choices for parameters or models.

In the NRC staff's RAI, staff stated that DOE provided limited support that the model projections would likely bound anticipated impacts (NRC, 2020a). The NRC staff stated that support should be provided for the key intermediate results of the numerical modeling, such as for observed transport rates of contaminants such as ^{90}Sr. Key intermediate results of the PA modeling included, but were 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. In response to the NRC staff's RAI, DOE provided detailed information addressing model support for the PA (DOE, 2021c). The information provided was comprehensive for the specific intermediate models mentioned by the NRC staff, however, those intermediate models were meant to be examples rather than a comprehensive list. In response, DOE described past, and in some cases, future activities that have been or will be completed to support key intermediate outputs.

One of the key components of the PA is wasteforms performance. For the glass wasteforms, as discussed in Section 4.6, maintenance of very low release rates depends in part on avoiding Stage III behavior. Stage III behavior can be triggered by the minerals present in the SMRN. The modeling of glass release assumed a secondary mineral that was not observed during glass leaching experiments because it provided better agreement between the modeling and experimental data. In the NRC staff's view, this represents a calibration rather than a validation. DOE did have a good practice of using two different tools (STOMP and GWB) to estimate and compare glass release rates. This type of comparison lends support to the results, however, more in terms of verification rather than validation. Inter-model comparisons help identify if some error occurred in one of the codes especially a numerical or data input error. If the two codes used the same data that was in error or both codes did not include a significant process, then the results would agree with each other but differ from reality. DOE described their field-scale lysimeter tests to provide support for the performance of the wasteforms. As discussed in Section 4.6.2, this is an excellent activity to provide model support for a key aspect of the PA and it is strongly supported by the NRC staff. It may be useful to provide uncertainty estimates in the prospective modeling of the lysimeters because the results will invariably differ between the projections and the measurements.

The engineered cover system is yet to be fully designed and implemented but will be a variant of that implemented for the Prototype Hanford Barrier experiment (PNNL-18845, 2011). DOE based the expected performance on lysimeter studies, tracer tests, and computer simulations (PNNL-14744, 2004) as well as the 15-year performance of the 200-BP-1 Prototype Hanford Barrier (PNNL-18845, 2011). Data gathered from the Prototype Hanford Barrier, installed in 1994 over the 216-B-57 Crib, indicated that an IDF cover design similar to the Prototype Hanford Barrier could be sufficiently robust and perform as designed for the long-term.

The unsaturated flow and transport modeling is important with respect to the timing of contaminants reaching potential receptors but of lesser importance with respect to the magnitude of the impacts that may result. The wasteforms in the IDF are anticipated to provide slow rates of release but those releases will persist, once started, for a long period of time. Dispersion is of lesser importance compared to dilution. Nonetheless, the observed transport times from past releases are an important source of information that can be used for confidence building. The unsaturated flow modeling uses many inputs that are determined from characterization and modeling. The inputs that are derived could be non-unique. DOE provided comparisons of moisture content depth profiles with model simulated values. The comparisons were reasonable, however, they showed limited sensitivity to changes in recharge rates. For the relatively same moisture content profile, the estimated travel time could be substantially different.

DOE's model support is focused on flow, whereas the PA is used to calculate flow and transport. Past releases provide a source of information to test the completeness of the flow and transport modeling. DOE raised challenges associated with trying to model past releases and those challenges are valid. The flow and transport model used in the PA should be robust such that reasonable agreement between observed and modeled results can be achieved in different geologic settings and under different flow regimes. There are high-flow scenarios that are evaluated in the PA analyses. DOE can enhance their support for unsaturated flow and transport modeling by using the PA models to simulate past releases of conservative and non-conservative contaminants.

DOE also discussed support for the dilution expected in the saturated zone. This is a key factor in the PA. The dilution of contaminant fluxes transported vertically from the unsaturated zone by the high flow of water in the aquifer was justified by DOE. Additional PA maintenance activities are planned to support saturated zone flow modeling.

Because model support activities are planned and will be completed in the future, there is uncertainty that new information may become available that shows the current models are not supported. In this event, DOE has a process called the Unresolved Disposal Question Evaluation (UDQE) where the significance of the information is evaluated, and the result could lead to a new assessment. A process is in place to mitigate the impact of this type of uncertainty. Because of the importance of model support to the decision-making process, a dedicated plan, strategy, and document summarizing model support for the IDF PA could enhance confidence that the numerical models adequately project or bound future impacts (Recommendation #21).

4.12.3 NRC Conclusions on Model Support

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, PA documentation, additional references, the PA model, and had technical discussions with DOE staff and contractors on the model support for the IDF PA. Staff performed independent verification of DOE's results. Staff modified DOE's inputs and calculations to examine additional cases, such as examining the impact if the long-term infiltration rates were to be much higher than anticipated. Staff examined numerous uncertainties associated with model support. The NRC staff has the following conclusions:

- DOE provided documentation of model support. The model support provided was appropriate for regulatory decision-making.
- Verification activities were better supported than validation activities.
- The wasteform lysimeter test is an excellent example of a validation activity of a key intermediate output.

The NRC staff has the following recommendation on model support:

- Because of the importance of model support to the decision-making process, a dedicated plan, strategy, and document summarizing model support for the IDF PA could enhance confidence that the numerical models adequately project or bound future impacts. (Recommendation #21)

4.13 Uncertainty

The characterization and evaluation of uncertainty is especially important for the disposal of radioactive waste at the Hanford Site because of the complex operational history and the potential for long-lived radionuclides to be present in the waste. Once uncertainties are understood and evaluated, actions can be taken to mitigate the impact of those uncertainties. DOE identified, characterized, assessed, and mitigated uncertainties in the PA for WMA C, which the NRC staff previously reviewed (NRC, 2020b). The sections that follow provide a summary of DOE's analyses of uncertainty with emphasis placed on differences between WMA C (i.e., closure of below ground storage tanks with residual waste) and disposal of VLAW in the IDF.

4.13.1 DOE's Analyses of Uncertainty

DOE recognized the importance of the impacts of uncertainty on the results of performance assessment analyses used for decision-making. Though sensitivity and uncertainty analyses were discussed in Section 6 of the PA document DOE, the discussion and analyses of uncertainties was common throughout the PA document and supporting references (RPP-RPT-59958, 2018). DOE stated that the intent of the uncertainty and sensitivity analysis was to identify the assumptions and parameters that have the greatest impact on the projected doses and evaluate the consequences of the uncertainties relative to the performance objectives. Precise estimates of future impacts are not truly quantifiable, and some sources of uncertainty remain unquantifiable because they include elements of subjectivity (RPP-RPT-59958, 2018). DOE's guidance for completing the uncertainty and sensitivity analyses (DOE Guide 435.1-1) states that dose assessments have uncertainties, and a discussion of the uncertainties should be included in a PA. The guidance further states that an estimate of the magnitude of uncertainty is needed for the analysis of impacts that may occur after the 1,000-year compliance period.

Key uncertainties identified by DOE included wasteform 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. Within a topical area, for instance infiltration, DOE characterized uncertainties and translated them into ranges for key parameters. DOE used both sensitivity analyses and uncertainty analyses in a complimentary manner. DOE utilized sensitivity analyses to quantify the cause-and-effect relationships due to single-parameter or limited number of multiple-parameter changes in the parameter estimates. The results of the sensitivity analysis identified those parameters for which the variability in their estimates, either because of lack of knowledge or foreknowledge, limited data, or inherent randomness, introduced the greatest uncertainty into estimated radionuclide concentrations in the environment. Uncertainty analyses were used to evaluate how uncertainty in inputs collectively affect uncertainty in the analysis outcomes (for example, an estimate of dose to a receptor). As part of the uncertainty analysis, uncertain inputs were evaluated together within the system model to estimate a plausible range of outcomes. DOE stated that a probabilistic uncertainty analysis helps evaluate how the combination of various parameters could lead to different outcomes (for example, high dose or low dose).

In the system model used for sensitivity and uncertainty analyses the following assumptions were made for releases to groundwater:

1) No flow processes occurred within a wasteform or laterally between the backfill and wasteforms.

2) Advection took place only in the vertical direction in backfill regions.
3) No sorption processes occurred in backfill.
4) All contaminants were assumed to be soluble.
5) Diffusion from and within wasteforms was in all directions.
6) The model side boundaries were "zero-solute flux" boundary conditions due to symmetry.
7) The model top boundary for diffusion was "zero-solute flux" boundary condition.
8) The top boundary was open to infiltrating water at a specific rate equal to the product of the infiltration rate and the horizontal area including both waste and backfill.
9) The bottom boundary was open for advection, representing the unsaturated zone.
10) The carbon or stainless-steel containers were degraded instantly at the end of institutional control period (100 years).
11) The degraded containers, about 1 cm (0.4 in) thick, were treated as backfill.

DOE described the sensitivity analyses that were performed in Section 6.2 of the IDF PA (RPP-RPT-59958, 2018). For the groundwater pathway, a starting set of assumptions and parameters was established, and sensitivity cases were defined to evaluate the effects of changing those assumptions and parameters. The sensitivity cases were identified as key parameters and assumptions potentially affecting specific safety functions. A summary of the starting assumptions was provided in Table 6-18 (RPP-RPT-59958, 2018). The starting analysis used inventory case 7 and included both glass and non-glass wasteforms. The fractional release rate from glass was based on a regression equation that had a nominal value of 2.5×10^{-7} 1/yr (RPP-CALC-61192, 2017). DOE provided a summary of the 31 sensitivity cases examined in Table 6-19. Process models were not used for the sensitivity and uncertainty analyses because the computational burden was too large. DOE provided comparisons of the projected groundwater pathway doses for key radionuclides to demonstrate that the system model results were comparable to the process modeling results for the starting case.

DOE stated that each sensitivity case represented alternative input assumptions compared to the starting case, and therefore could be thought of as alternative conceptual models for processes in the model rather than single parameter "one-off" sensitivity analyses (RPP-RPT-59958, 2018). In many cases a single parameter may have been varied but the range for the variation in the parameter was selected to represent multiple uncertainties. For instance, the surface barrier flow safety function case was evaluated by changing the flow through the cover. The changes in flow represented different assumptions about the long-term behavior of the cover and how it may degrade over time. DOE provided numerous charts and detailed explanation of the results for each sensitivity analysis. As an example, Figure 4-19 is the results of using different values for the long-term infiltration rate. Higher infiltration rates produce larger and earlier doses while lower infiltration rates result in smaller and doses which occur later in time.

The uncertainty in the overall inventory (number of Bq (Ci)) disposed in the IDF was not evaluated in a sensitivity case as DOE indicated the dose results scaled linearly with inventory. DOE concluded that there was uncertainty in the inventory that would be disposed (DOE, 2021c). Development of waste acceptance criteria is the method used by DOE to ensure that

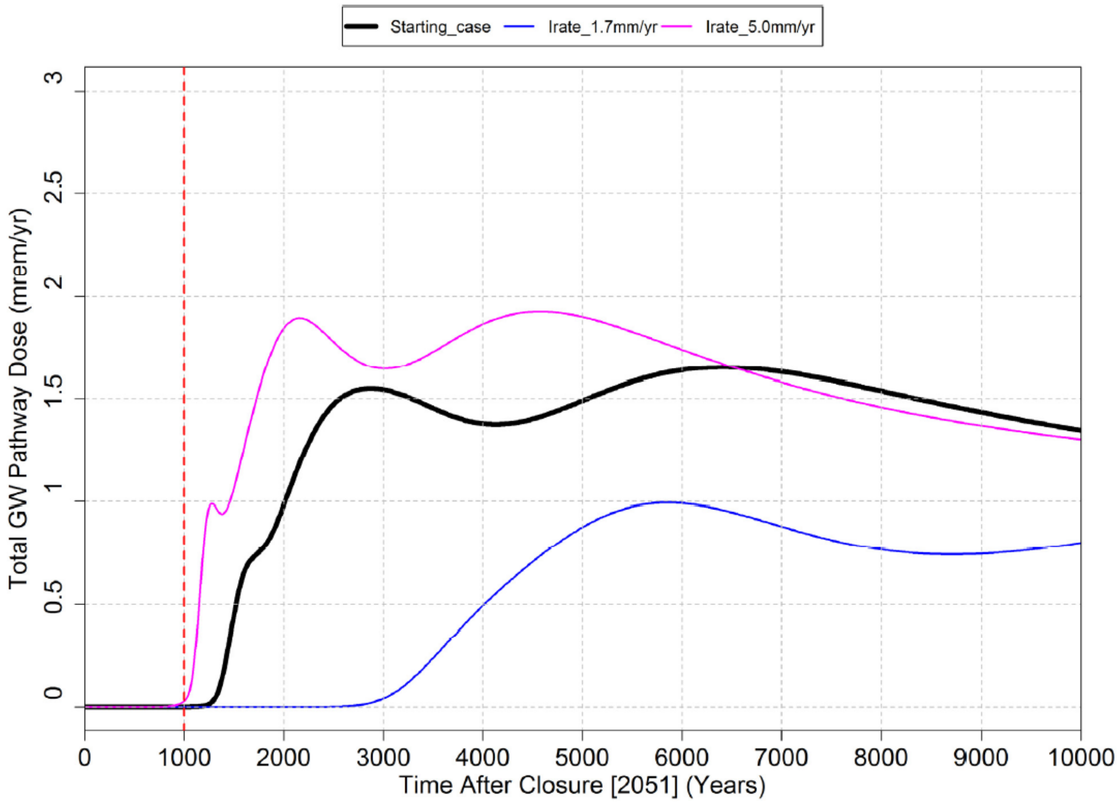


Figure 4-19 Results of the Infiltration Rate Sensitivity Case
[Figure 6-64 of (RPP-RPT-59958, 2018)]

the acceptable inventory that can be disposed is not exceeded[18]. The most significant uncertainty identified with sensitivity analyses was the quantity of key radionuclides contained in the glass or the non-glass wasteforms. In DOE's models, the glass wasteform is estimated to provide lower release rates than the non-glass wasteforms and therefore higher partitioning of key radionuclides to the glass wasteform during processing results in lower doses. The inventory cases were designed to examine different assumptions with respect to the quantity of key radionuclides retained in the glass in each pass through the melter as well as if recycle of the off-gas stream was used. The inventory cases resulted in doses less than 1 x 10^{-4} Sv/yr (0.01 mrem/yr) in the 1,000-year compliance period used by DOE. Inventory cases 10A and 10B resulted in doses after the compliance period exceeding 1.0 Sv/yr (100 mrem/yr) and case 7b was close to the compliance period performance objective. In response to the NRC RAI, DOE stated they will be using recycle to produce VLAW and have been completing additional research to increase the single-pass retention of key radionuclides (DOE, 2021c).

DOE examined different aspects of the grouted wasteform performance safety function with sensitivity analyses. For example, DOE changed the distribution coefficients (K_d)s to represent a change to oxidizing conditions from reducing conditions in the wasteform. In another case DOE changed the diffusion coefficients to represent alteration of the wasteform. The projected

[18] Waste acceptance criteria provide much more than allowable concentrations; they are general and provide many characteristics the waste must or cannot have.

doses increased by approximately 30% from the starting case. A variety of other analyses were presented in Section 6.2 of the PA document (RPP-RPT-59958, 2018). Most sensitivity cases showed moderate increases and decreases in response to changes in input parameters. One of the larger increases was associated with changes to the Darcy flux in the aquifer, with the peak groundwater dose after 1,000 years increasing from about 0.015 mSv/yr (1.5 mrem/yr) to a bit over 0.05 mSv/yr (5 mrem/yr). Figure 4-20 provides a summary of the peak doses from the groundwater pathway sensitivity cases completed by DOE (doses are expressed in mrem per year - to convert to mSv multiply by 0.01).

DOE examined other groundwater pathway sensitivity cases throughout the PA document (RPP-RPT-59958, 2018). For example, in Section 5.1 of the PA document, DOE described different analyses cases completed to evaluate near-field flow and source-term release. DOE examined the timing of engineered barrier "failure" by examining a case where the surface cover and liner had a step change in properties at 500 years post-closure. DOE identified only moderate sensitivity of the doses to changes in failure assumptions of the engineered barriers. 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) can have a significant impact on release rates if differing materials are present and simulation of capillary barrier effects occurs.

DOE examined sensitivity of doses from the atmospheric pathway (RPP-RPT-59958, 2018). The atmospheric pathway dose is larger than the groundwater pathway dose during the compliance period. The atmospheric pathway dose, with what DOE asserted was conservative analyses, was well below 0.01 mSv (1 mrem) during the compliance period for the base case. The largest sensitivities were to the inventory cases followed by the deposition velocity. None of these sensitivity cases resulted in a dose above the limit in the compliance period or during the sensitivity and uncertainty analyses period.

DOE used probabilistic analyses to evaluate the global impacts of uncertainty. DOE stated that underlying the probabilistic approaches presented in the PA was a conservative bias (i.e., pessimism) that had been applied in the selection of many of the models and input parameters. This conservative bias was not quantified in the uncertainty analysis but should be acknowledged when interpreting the results. DOE defined probability distributions for uncertain parameters and used GoldSim to sample the distributions in a Monte Carlo analysis. Latin Hypercube Sampling (LHS) was the sampling method used. DOE used guidance provided by EPA (EPA/630/R-97/001) in the selection of probability distributions (EPA, 1997). For probability distributions involving epistemic uncertainties DOE consider the principle of maximum entropy with due consideration of the effects of risk dilution. The total number of uncertain parameters considered in the system model was 98. DOE used different measures to examine parameter uncertainty including correlation coefficients, standardized regression coefficients, partial correlation coefficients, and importance measures. DOE provided results for different time periods. For the 1,000-year compliance period the most important parameters were the background infiltration rate, the K_d for Tc in the H2 unit, a parameter related to glass release, and a parameter related to the amount of dilution expected in the saturated zone. For

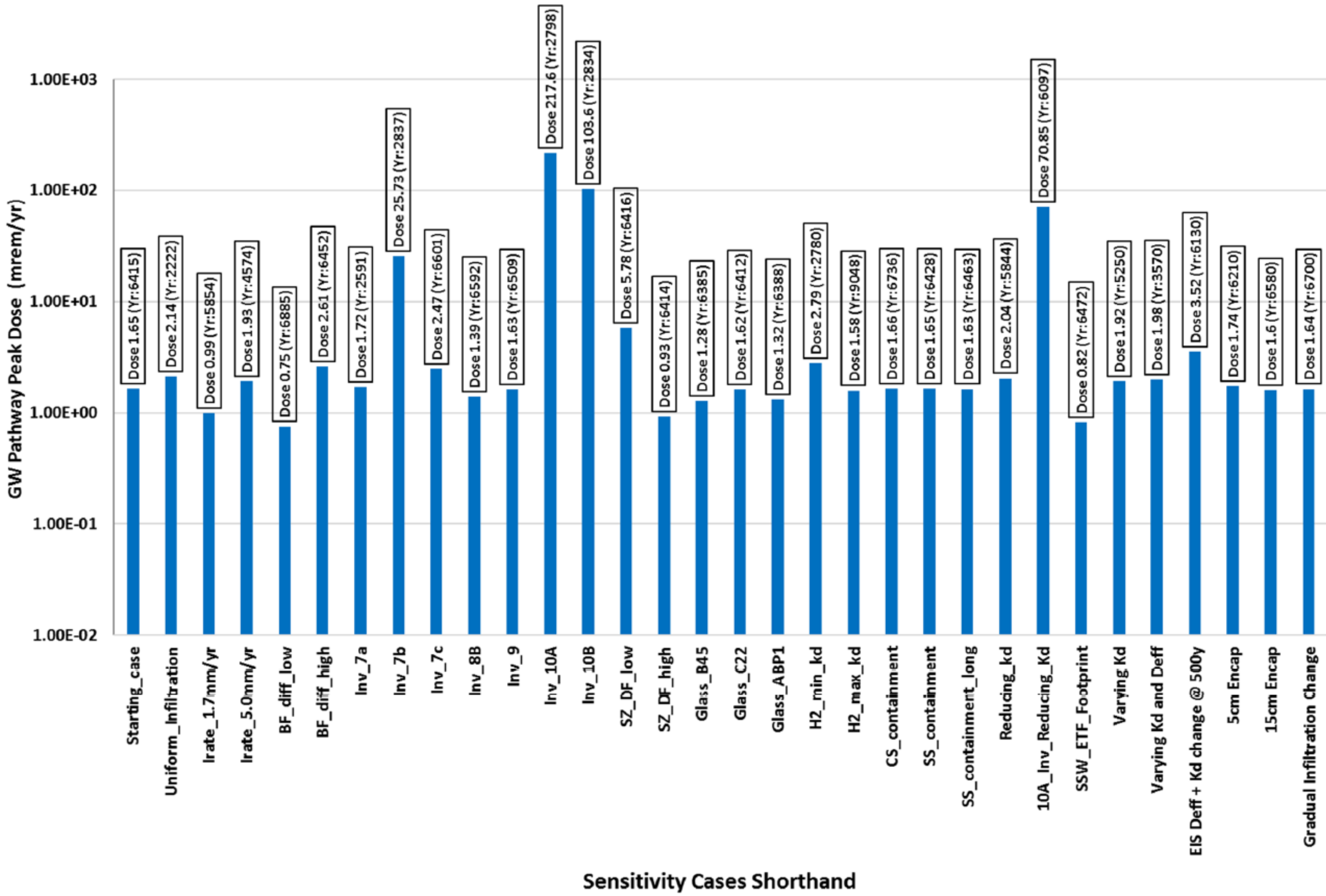


Figure 4-20 Results of DOE Sensitivity Cases
[Figure 6-110 of (RPP-RPT-59958, 2018)]

3,000 and 6,000 years the parameters were similar with additional glass dissolution parameters becoming important.

NRC requested additional information on DOE's uncertainty and sensitivity analyses (NRC, 2020a). NRC stated that DOE should expand the sensitivity and uncertainty analyses to reflect sources of uncertainty that were not evaluated or carried forward from other technical reports. Examples provided were glass release uncertainties, inventory partitioning/splits, and inventory magnitude. In response to the NRC RAI, DOE stated that they believed the range of sensitivity and uncertainty analyses performed were sufficient (DOE, 2021c). DOE indicated the analyses were used to identity the activities included in the PA maintenance plan (CHPRC-03348, 2019). DOE provided additional sensitivity analyses in response to the RAI, these included:

- Near-field hydrology and the effect of the surface barrier degrading faster than the liner system.
- Hydraulic characteristic, notably the MCC and the effects on glass release.
- SMRN effects on glass release.
- Other sources of uncertainty and the effects on glass release.
- Inventory partitioning effects on groundwater pathway concentrations and doses.
- Inventory effects on groundwater pathway concentrations and doses.

One scenario mentioned by the NRC and evaluated by the DOE was a "bathtub effect" scenario resulting from a liner system that functions longer than the cover system. DOE evaluated the scenario and estimated that the peak groundwater concentration that would result would be approximately 25% of the peak dose estimated in the starting case.

With respect to hydraulic characteristics of the glass wasteform, DOE stated that they agreed that the hydraulic characteristics of fractured glass are uncertain and that the effect of the uncertainty in MCCs on the predicted vertical Darcy flux and moisture content should be evaluated (DOE, 2021c). The evaluation should consider the potential impact of uncertainty in van Genuchten-Mualem parameter values—namely the saturated hydraulic conductivity, porosity, residual moisture content, and van Genuchten parameters alpha and n—used in the STOMP glass release model. DOE stated that because of competing effects between Darcy velocity and moisture content the fractional release rates from the glass wasteform were relatively insensitive to changes in net infiltration rates. DOE provided analyses to examine the uncertainty associated with MCCs for fractured glass. DOE used GWB because it could be executed more rapidly, and the estimates of release rates were comparable to STOMP. The results (presented in Table 2-12-4 of the RAI response) showed a close to linear relationship between changes in the residence time (resulting from different MCCs) and the estimated FRRs (DOE, 2021c).

The NRC staff requested additional information about the uncertainty in assumptions about the SMRN (NRC, 2020a). Though DOE evaluated different cases in the PA, the NRC staff stated that those cases should be expanded. Staff indicated in the basis for the RAI, that DOE did not use secondary minerals observed in experiments in the PA because acceptable glass degradation rates could not be achieved. The NRC staff recommended that the SMRN cases should be expanded and, if possible, supported by information from experiments and the literature. DOE recognized the importance of uncertainty in SMRNs and included it within the scope of the PA maintenance plan (DOE, 2021c). To quantify the effect that alternative SMRNs may have on the predicted glass release rates, additional sensitivity analyses termed “COMBX” were performed by DOE. DOE also included other glass release rate parameters and uncertainties in the evaluation as well as MCCs (residence times). The COMBX cases indicated a total range of FRR results from 1.28×10^{-8} to 3.80×10^{-6} yr-1 for the cases when the residence time was fixed at the reference case value of 13.1 years and from 1.22×10^{-8} to 2.41×10^{-5} yr-1 when the residence time was assumed to be either 5 times or 1/5 times the reference case value. The maximum (pessimistic) predicted FRRs were a factor of 66 and 420 greater than the reference case FRR of 5.75×10^{-8} yr-1 while the minimum (optimistic) predicted FRRs were a factor of 4.5 and 4.7 less than the reference case FRR. DOE stated this non-symmetric distribution reflected the observation that the total corrosion rate is controlled both by the dissolution rate of the glass matrix, in accordance with the TST-based rate law, and by the rate of the ion-exchange reaction controlling glass hydration. Figure 4-21 shows the dose results with respect to different FRRs.

DOE indicated that the glass release rate sensitivity analyses, conducted as part of the IDF PA (Section 5.1.2 of RPP-RPT-59958) as well as the additional sensitivity analyses conducted in the RAI response, illustrated the significance of uncertain glass corrosion and ion-exchange parameters as well as the hydraulic conditions in the fractured glass and other aspects of the near-field hydrogeochemical environment on the predicted FRR from the glass wasteform.

4.13.2 NRC Evaluation of Uncertainty

The NRC staff previously reviewed DOE’s approach to uncertainty when reviewing the Draft WIR Evaluation for WMA C (NRC, 2020b). DOE used a similar approach to uncertainty for the disposal of waste at the IDF. Staff reviewed DOE’s documentation, supporting references, and computer model files as well as technical reports and literature not directly referenced in the IDF PA or Draft WIR Evaluation for VLAW. NRC’s review of uncertainties associated components of the PA (e.g., infiltration) are found in the respective sections of this report.

DOE described the methods used for sensitivity and uncertainty analyses. The information provided on sensitivity and uncertainty analyses was complimentary and aided in an understanding of performance of the IDF. Sensitivity and uncertainty analyses were completed using the system model rather than the process models utilized for the deterministic base case. DOE compared the results for the system model to the process models to demonstrate that the system model could be an appropriate surrogate. NRC agrees that the system model is a sufficient representation of the process models. Given the uncertainties inherent in the assessment, agreement between a system model or abstraction and a process model does not need to be exact.

The process models can be useful to examine the impact of discrete features (such as a sump/drain) and higher dimensionality. However, if the process models do not represent discrete features with appropriate resolution or include higher order dimensionality that is significant to the assessment then the increased computational and interpretational burden (as well as QA) associated with the process models is generally not worth the excess burden. In addition, the process modeling usually has severe limitations when it comes to evaluating uncertainties. If the process modeling is not validated (supported), then the increased resolution, effects, etc. are not of sufficient value in the regulatory decision-making process. DOE should carefully consider when process modeling is necessary and otherwise err on the side of choosing methods that allow for a more complete assessment of uncertainties.

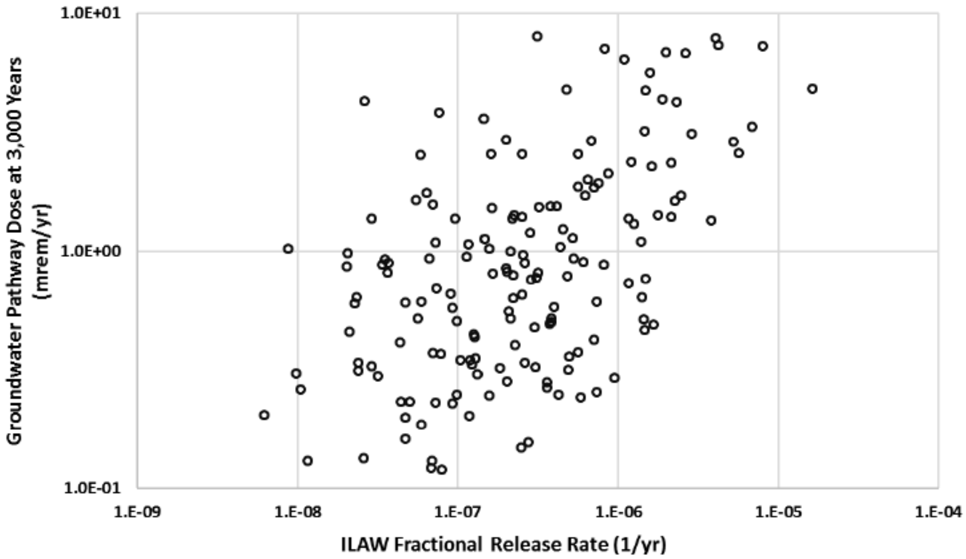


Figure 4-21 Sensitivity of Groundwater Pathway Dose at 3,000 years to the Fractional Release Rate from ILAW
[Figure 2-12-2 from (DOE, 2021c)]

For sensitivity analyses, 31 cases were examined by DOE to evaluate uncertainties that could impact safety functions (RPP-RPT-59958, 2018). These 31 cases covered many of the topical areas that NRC believes are important to the performance of the IDF. NRC agrees that sensitivity analyses have an important role to play in performance assessment analyses and DOE used them effectively to communicate how the PA model was working. However, caution is needed in using sensitivity analyses to *address* uncertainties. Sensitivity analyses can be very useful to identify potentially important uncertainties because if the uncertainties are important on a local basis, they may also be important on a global basis. However, because a parameter is locally insensitive does not necessarily mean it will be insensitive in a global uncertainty assessment. The complexity of the model and if the uncertainties have non-linear effects or complex interactions with other uncertainties influences the significance of an uncertainty.

Numerous instances were observed throughout the PA analyses and underlying reports where DOE used an argument about a local impact and relative change to conclude the technical issue was not significant on a global basis. The issue was then dropped from further consideration. For instance, this occurred with respect to discretization of the models for release rates. If the doses are very small in relation to the performance standard, then it would take many uncertainties with effects in the same direction (e.g., most increasing) to create a change in the conclusions made from the analyses. Because the analysis was large and complex with many inputs and uncertainties, NRC staff recommend that DOE compile the uncertainties addressed with local sensitivity arguments and include them in a global uncertainty analysis (Recommendation #22). This would ensure that the likelihood of different uncertainties increasing or decreasing the output metrics would be reflected in the results. Otherwise, the significance of uncertainties can be difficult to interpret to perform risk-informed regulatory decision-making and stakeholders may misinterpret information.

In the 1,000-year compliance period DOE estimated that doses were anticipated to be very low (small fractions of a mrem) primarily because of the delay in transport afforded by the thick unsaturated zone and low infiltration rates. After the contaminants arrive at the receptor location the doses are much higher in the sensitivity and uncertainty period, though they are still a small fraction of the performance standard. DOE's use of local sensitivity analyses to evaluate uncertainties is not likely to impact the demonstration of compliance in the 1,000-year period because of the robustness of the unsaturated zone as a delay mechanism. The margin between the estimated results and the performance objectives are likely to be smaller than estimated by DOE and interpretation of the significance of future uncertainties may be more challenging. For instance, if research identified Stage III performance of glass disposed at IDF was likely to occur or if a significant fraction of volatile species were deposited inside the glass canister during the vitrification and cooling process the importance of those changes would be difficult to ascertain. Numerous changes of 30 to 50% can add up quickly in a large, complex analysis. A more complete system model, from an uncertainty standpoint, would allow DOE to be better prepared to address uncertainties that are inevitably going to arise as DOE progresses from research to production. For this reason, the NRC staff recommend that DOE should place more emphasis on uncertainty analyses using the system model rather than local sensitivity analyses (Recommendation #23).

The local sensitivity analyses have two significant shortcomings when applied to an analysis of a large, complex system. First, the observed output of the sensitivity analysis (e.g., how much the dose increased or decreased) is a local measure of the effect with everything else held constant. Complex responses are common where a parameter may not be sensitive until something else happens (threshold effect) elsewhere in the model. Local sensitivity analyses work well when the number of uncertainties investigated are limited but are much less reliable when there are many uncertainties. Second, the local sensitivity analyses do not provide a measure of likelihood. The risk context is not clear, and the results are more difficult to interpret. If a parameter is increased or decreased by a factor of 5, for example, what is the likelihood that the parameter could have increased or decreased by that magnitude? As DOE explained, the local sensitivity analyses were used to examine conceptual model uncertainty (DOE, 2021c). According to DOE the range over which a parameter may have been varied was to account for multiple sources of uncertainty. It was not clear to the NRC the basis for the ranges and how the significance of the results was determined.

NRC reviewed DOE's uncertainty analyses and concluded that DOE clearly described what was done, the methods used, and the results of the analyses. DOE developed probability distributions for uncertain parameters considering the principle of maximum entropy while taking into consideration risk dilution. In general, the probability distributions that were developed were appropriate. Some distributions were truncated, and the type of distribution selected (e.g., uniform, triangular) was in many cases arbitrary, but the overall results of the system model uncertainty analyses were not likely to be highly sensitive to these aspects of assigning probability distributions. LHS was used as the sampling method, which was appropriate. DOE examined 98 parameters in the uncertainty assessment which may be perceived a large number. The number of uncertain parameters was on the small side for the type of analyses performed. Many uncertainties were not carried forward by DOE from the underlying analyses to the system model. NRC asked for additional information associated with uncertainty and sensitivity analyses. In response to the RAI, DOE provided additional sensitivity analyses (DOE, 2021c).

The partitioning of volatile species during glass production is a key uncertainty. DOE stated that the cases evaluated encompassed decisions about design of the system that were yet to be established at the time of the analyses, in other words decision analyses rather than uncertainty analyses per se. Use of recycle is projected to result in higher amounts of volatile species being retained in the glass wasteform (DOE, 2021c). If recycle is the selected process design, there are still considerable uncertainties associated with how effective it will be on an operational scale (as discussed previously). Based on present information and independent interpretation of the data and uncertainties, NRC estimates that retention of ^{99}Tc and ^{129}I in glass could be as low as 90% or greater than 99%. It is recommended that this uncertainty should be part of system model uncertainty analyses (Recommendation #24).

The MCCs assigned for glass and backfill combined with the coarse discretization and assignment of uniform material properties for a given material type results in about 90% of the flow that reaches the glass wasteform being diverted around the glass wasteform. There was not experimental or information from analogs/surrogates to provide model support for the result. The number of samples used to derive the MCCs was limited, or data were assumed. The hydraulic properties and other uncertainties associated with the flow of water through the glass wasteform for release modeling should be included in uncertainty analyses with the system model (Recommendation #25).

An uncertainty that DOE addressed in the RAI response was associated with differing failure rates of engineered materials (DOE, 2021c). NRC was concerned as to what would be the impact if the liner system performed much longer than the cover system leading to a potential “bathtub effect”. This has been observed at early commercial low-level waste disposal sites leading to performance issues. DOE performed an analysis to estimate what the impacts would be (DOE, 2021c). This analysis was well-thought out and clearly described. However, the issue of very uncertain failure rates and times of engineered barriers is more generic than the potential for the bathtub effect. DOE took credit for 500 years of performance for the engineered cover and explored different failure times and rates in sensitivity analyses. The stainless-steel canisters containing the glass were given no performance credit. The sump and drain system under the liner was assumed to be operational during the institutional control period. The liner was assumed to provide containment for 500 years. DOE analyzed cases with the liner functioning and focusing flow to the drains and the liner not functioning. Overall, the assumed degradation rates and failure times of these engineered barriers are likely to be pessimistic. Though credit should not be taken in a regulatory compliance analysis for something that is unknown, exploration of the full range of performance states of engineered barriers (combinations) is useful in an uncertainty analysis using the system model. This can be done by assigning different performance states to each component either discretely or continuously and analyzing the system probabilistically.

A key uncertainty, which DOE is continuing to investigate, is the potential for Stage III behavior of glass. DOE stated that because of the conditions in the IDF that they do not expect Stage III behavior to occur (RPP-RPT-59958, 2018). Stage III behavior would be ideal to include in the system model uncertainty analyses to determine how significant the uncertainty could be because data is being developed on the potential magnitude of the effect with the zeolite seeding experiments. DOE provided many different analyses on additional uncertainties associated with glass performance (DOE, 2021c). The NRC staff finds that these analyses were reasonable. The uncertainty of glass performance reflected in DOE’s uncertainty analyses in the PA was limited to compositional differences. The uncertainty of glass performance is significantly larger than included in the uncertainty analyses for the PA.

DOE stated that inventory uncertainty does not need to be part of the uncertainty evaluation because the impact is more or less linear. NRC concurs that the impact is more or less linear, but one useful result of including all known and potentially significant sources of uncertainty in the uncertainty analyses is the results can be used to inform stakeholders of the potential and likelihood of extreme outcomes. DOE used combined sensitivity cases to look at the effects of certain uncertainties if they were compounded. These are informative but also can be misinterpreted if the likelihood of the combination is not addressed. The uncertainty analyses can be used to communicate an estimate of the likelihood of certain results.

4.13.3 NRC Conclusions on Uncertainty

NRC reviewed DOE's Draft WIR Evaluation for VLAW, PA documentation, additional references, the PA model, and had technical discussions with DOE staff and contractors on the characterization, assessment, and analyses of uncertainties for the IDF PA. The NRC staff carefully examined DOE's performance assessment model and the results of sensitivity and uncertainty analyses. NRC has the following conclusions:

- The types of uncertainties examined were reasonable and consistent with NRC's understanding of the system.
- The significant uncertainties identified with the system model were consistent with NRC's understanding and experience with similar systems.
- It was reasonable and appropriate to use the system model for sensitivity and uncertainty analyses.
- DOE evaluated an appropriate set of parameter values and provided numerous charts and tables to explain the results.
- The GoldSim software package was acceptable to use to perform the uncertainty analyses and the sampling approach was sound.
- While local (e.g., single parameter) sensitivity analyses can provide useful information the results can be difficult to interpret when the sources of uncertainty are numerous.
- Combination cases for sensitivity analyses lack the proper context of likelihood/probability and therefore may be misinterpreted. Uncertainty analyses are preferred.
- Methods for consideration of failure rates and times of engineered barriers could be improved.

NRC has the following recommendations related to uncertainty:

- DOE should compile all uncertainties that were "dropped" from further consideration based on local sensitivity analyses and include them in a global uncertainty analyses (Recommendation #22).
- DOE should place more emphasis on uncertainty analyses using the system model rather than local sensitivity analyses (Recommendation #23).

- Based on present information and independent interpretation of the data and uncertainties, NRC estimates that retention of ^{99}Tc and ^{129}I in glass could be as low as 90% or greater than 99%. It is recommended that this uncertainty should be part of system model uncertainty analyses (Recommendation #24).
- The hydraulic properties and other uncertainties associated with the flow of water through the glass wasteform for release modeling should be included in uncertainty analyses with the system model (Recommendation #25).

4.14 Quality Assurance

Quality assurance (QA) is an essential component of technical analyses. QA is used to ensure the analyses are correct, can be replicated, and can be independently reviewed and verified. NRC had previously reviewed DOE's QA procedures and results for implementation to WMA C (NRC, 2020a). Essentially all aspects of QA were identical for disposal of waste in IDF compared to WMA C although a different team of analysts were involved. The sections that follow provide an overview of the DOE approach and the results of NRC's review with an emphasis on differences and new observations.

4.14.1 DOE's Quality Assurance

DOE described their QA processes, approach, and implementation in Section 10 of the PA document (RPP-RPT-59958, 2018). The IDF PA development and implementation was performed under:

- ANSI/ASME NQA-1-2008, *Quality Assurance Requirements for Nuclear Facility Applications* and NQA-1a-2009, *Addenda to Quality Assurance Requirements for Nuclear Facility Applications*
- DOE O 414.1D (DOE, 2011b)
- Title 10, CFR, Part 830, "Nuclear Safety Management" (10 CFR 830), Subpart A—Quality Assurance Requirements
- EM-QA-001, *EM Quality Assurance Program (QAP)*
- Title 10, CFR, Part 830, Subpart A, § 830.120, "Scope"

The same QA requirements were applied to subcontractors working on the IDF analyses. DOE used a project plan that included problem definition and background, quality objectives and criteria for measurements and data acquisition leading to model inputs and outputs, data validation and usability, references, documentation and records management, special training requirements and certifications for modelers, and assessments and reports to management. The model documentation requirements identified during project planning aligned with DOE management expectations.

The development, application, and preservation of environmental models used to support regulatory decision-making and analysis was conducted under a general project plan that implemented the requirements of DOE Order 414.1D, the direction related to modeling in EM-QA-001, as well as EPA guidance provided in EPA/240/R-02/007 (EPA, 2002). This plan provided for modeling to be performed in a QA framework for the full lifecycle, with integrated control of models, implementing software, applications, and supporting information. Highlights of the general plan requirements under which the PA was developed included:

- Training was stipulated in the general plan for modelers that ensures the requirements and QA processes for model development and application were communicated.
- Software used to implement environmental models was controlled.
- A process for model documentation, control, and preservation was specified.
- Full checking and senior review of model package reports and calculation files was required as part of the process.

Software used for model implementation was managed following a controlled software management procedure. The controlled software management procedure directed management of all software including configuration control, evaluation, implementation, acceptance and installation testing (verification and validation), and operation and maintenance. Software used to implement the models and perform calculations was approved for use under the controlled software management procedure. DOE described the implementation of QA to STOMP, GWB, GoldSim, and other project software such as some legacy systems.

DOE stated that the QA procedures addressed the four basic model components necessary to provide traceable, reproducible models including 1) the basis for the model inputs, including data packages, 2) the models, 3) the applications of the models, and 4) the implementing software. The DOE contractor used a system called EMMA to maintain traceability and reproducibility for model components by providing for version documentation and preservation of the model basis, inputs, and outputs, along with identification of the software packages and specific versions used.

DOE used different methods for data quality depending on how old the data was. Data prior to 2000 was not verified or used in the IDF PA (RPP-RPT-59958, 2018). Data between the years 2000 to 2005 were identified as questionable and targeted for review to determine acceptable quality. DOE procedures provided guidance for qualifying and using existing data for supporting engineering activities. Existing data was defined as data determined necessary for conducting activities but developed by methods outside of those normally recognized NQA-1 standards or prior to NQA-1. The procedure provided several methods are to qualify pre-existing data. The two methods used with the IDF PA data generated from 2000 to 2005 were a data corroboration method and QA program equivalency.

DOE self-identified a significant error associated with glass release modeling after completing the final draft of the IDF PA. The glass researchers at PNNL identified an error in the values of *riex* that had been reported in the data package used to support the IDF PA, PNNL-24615 and the key references cited therein, notably PNNL-14805 (PNNL-14805, 2004). This error in *riex* reported in PNNL-26594 was the result of an improper calculation of the *riex* values due to an incorrect unit conversion (PNNL-26594, 2017). This error affected the *riex* values used for both LAWA44 and LAWC22 glasses but did not affect the *riex* value for LAWB45, which was 0.0. The error correction resulted in a reduction of the *riex* by about a factor of 7 for LAWA44 and a factor of 10 for LAWC22. The use of the corrected value would result in a decrease in the FRR by about a factor of 10.

4.14.2 NRC Evaluation of Quality Assurance

NRC reviewed DOE's QA for data, software selection and implementation, technical analyses, model support, and review and checking. NRC considered the guidance provided in chapter 8 of NUREG-1854 when reviewing DOE's QA program and implementation of the program (NRC, 2007).

DOE used appropriately qualified staff to perform the analyses. DOE described their qualifications and experience. In interactions with the staff who performed the technical analyses, NRC had high confidence in their technical expertise. The contractor staff were receptive to questions and constructive criticism. DOE used dedicated personnel for roles such as maintaining configuration control on software and ensuring qualified software was used by appropriately qualified and trained personnel.

The software selected was appropriate for the analyses. In the case of STOMP, the verification tests performed to accept the software does not completely align with how the software is used in the analyses. The glass release modeling and unsaturated flow and transport between the backfill and the wasteforms and within the wasteforms is not directly analogous to the verification tests. This is not a circumstance unique to DOE's assessment of the IDF performance. For many complex process models, it can be difficult to identify verification tests that can, by their very nature, be verified with a different calculation. With respect to glass release modeling DOE attempted to mitigate this by comparing two models – STOMP and GWB. This was an appropriate approach to help address the suitability of the verification tests. DOE may want to consider data from outside the U.S. as some countries may have generated relevant data from lab or field experiments that could be used as verification test cases. Excessive run time for certain calculations can by symptomatic of numerical issues. Extra effort to examine converge issues is usually worthwhile.

DOE's PA made use of a variety of software products (RPP-RPT-59958, 2018). The primary software products were STOMP, GWB, and GoldSim. DOE used other software tools available at the Hanford Site in the preparation of model inputs. DOE described select aspects of QA applied to these legacy systems but did not provide a full accounting of the QA status of these legacy systems or how they were determined to be suitable for use in the program. For instance, in the review of WMA C NRC discussed significant quality assurance issues that were identified by a DOE contractor when the HDW model was reviewed (NRC, 2020b; RPP-19822, 2005). DOE indicated TopSIM 3.0 provided key information for the assessment but did not discuss its qualification status (DOE, 2021c). It is understood that many of these legacy systems provide tangential or secondary information for the assessment, but the qualification status of all software used should be clear and the reliability of the information provided by that software should be ensured.

NRC reviewed the PA documentation and select system models developed in GoldSim. NRC did not identify a single difference between the documentation of parameters and their ranges and that which was used in the system model. NRC also reviewed select process model input and output files and did not identify any inconsistencies with the documented information. NRC traced data cited in the PA to source references and did not identify any differences, though as discussed in the previous section there were uncertainties that were not carried forward into the PA. Tables and figures provided source references when applicable. In the response to the NRC RAI DOE provided updated values for the retention of ^{99}Tc and ^{129}I in glass (DOE, 2021c).

Tracing DOE's basis for the values proved challenging. One large reference would lead to another large reference (that the NRC did not have and would need to request and then there would be a significant lag before receiving) and this process was repeated many times. The draft waste evaluation and response to the RAIs did not have the same traceability of information as the PA. In the future it would be more efficient for the NRC to review the DOE information if all primary references were provided to the NRC or were otherwise available to the public.

After completing the final draft of the IDF PA, the glass researchers at PNNL self-identified an error in the values of *riex* that had been reported in a data package used to support the IDF PA (PNNL-24615 and PNNL-14805). The error resulted in glass release rates being overestimated by a factor of 7 to 10 depending on the glass composition. Though this error was ultimately in DOE's favor, the magnitude of the error was significant. Outside of the inventory cases there were very few individual parameters that were observed to increase or decrease the doses by a factor of 10. This error appears to be isolated and was addressed by DOE. It highlights the difficulty of ensuring the correctness of the information used in a PA. The challenge is more difficult at Hanford because the site has been in operation for a long period of time and the amount of information generated is enormous. This error highlights the importance of using a robust QA program and sometimes errors may still occur. DOE has a program in place to assess and address the significance of errors that are identified. Model support can be used as a defense-in-depth to ensure correctness of the information used.

4.14.3 NRC Conclusions on Quality Assurance

NRC reviewed DOE's Draft WIR Evaluation for VLAW, PA documentation, additional references, the PA model, and had technical discussions with DOE staff and contractors on the quality assurance applied to the IDF PA. The NRC staff carefully examined DOE's performance assessment model and the supporting documentation. NRC has the following conclusions:

- DOE's quality assurance procedures and implementation of the procedures for the disposal of VLAW at IDF were appropriate.
- NRC did not identify any errors.
- The PA documentation was transparent and traceable. The reports were of high quality.
- Qualified staff were used to complete the analyses. Staff were appropriately trained.
- DOE provided records that independent review and checking was completed.

NRC does not have any recommendations associated with quality assurance.

4.15 Demonstration of Compliance with NRC's Performance Objectives

NRC's performance objectives for the disposal of low-level radioactive waste are found in 10 CFR Part 61. The performance objectives, along with other regulatory requirements, are designed to ensure that public health and safety will be protected with reasonable assurance in the present day and into the future.

DOE Order 435.1, *Radioactive Waste Management* and DOE Manual 435.1-1, *Radioactive Waste Management Manual*, requires all radioactive waste subject to the Order to be managed as either LLW, TRU waste, or HLW. DOE Manual 435.1-1 states that waste resulting from reprocessing spent nuclear fuel determined to be WIR is not HLW and shall be managed under DOE's regulatory authority. The criteria for determining if the waste is not HLW, and can be managed as LLW, include criterion B:

(B) It will be managed to meet safety requirements comparable to the performance objectives set out in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 61, Subpart C;

NRC had previously performed a consultive review for WMA C at Hanford (NRC, 2020b). Examination of the criteria DOE uses under DOE Order 435.1 and comparison to NRC's performance objectives was completed in that review and will not be repeated here. In addition, some aspects of DOE's demonstration of compliance with the standards found in DOE Order 435.1 are nearly identical for disposal of VLAW at the IDF as was provided for WMA C. This review provides a summary of the previously reviewed aspects and places emphasis on differences and new material.

The NRC staff previously determined that the standards DOE applied to demonstrate compliance with DOE Order 435.1 were comparable with the requirements found in 10 CFR Part 61 (NRC, 2020b). In addition, DOE's specification of "reasonable expectation" when compared to NRC's use of "reasonable assurance" was not materially different from a technical perspective. The requirements in 10 CFR 61.40 include that land disposal facilities be sited, designed, operated, closed, and controlled after closure such that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in 10 CFR 61.41 through 10 CFR 61.44. For NRC's review of the disposal of waste at the IDF, site selection was not part of the evaluation because the facility is already constructed, however, the NRC staff reviewed the site characteristics. Site closure and institutional control will occur in the future. DOE provided information on anticipated plans and actions associated with site closure and institutional control.

4.15.1 DOE's Demonstration of Compliance with NRC's Performance Objectives

In the draft waste evaluation DOE provided information for the NRC to evaluate whether the vitrified glass from the DFLAW approach will be managed to meet safety requirements comparable to the performance objectives 10 CFR Part 61, Subpart C for disposal of low-level radioactive waste (DOE, 2020a). DOE stated that disposal of VLAW in the IDF will meet DOE safety requirements for low-level radioactive waste disposal, which are comparable to the 10 CFR Part 61, Subpart C performance objectives.

4.15.1.1 Protection of the Public After Closure

The previous sections of this document described DOE's information associated with the PA analyses that was used to estimate potential future doses to a member of the public. This section focuses on the results of those analyses and the integration of the technical information (e.g., waste release, flow and transport) to estimate doses to a member of the public.

DOE used multiple sources of information to collectively provide the basis that offsite members of the public would be protected from disposal of wastes at IDF (DOE, 2020a). DOE used a base case analysis (deterministic), sensitivity analyses, and probabilistic uncertainty analyses to provide estimated doses to a hypothetical member of the public (RPP-RPT-59958, 2018). The member of the public was assumed to reside 100 m (328 ft) from the edge of the facility excavation and draw contaminated water from a well located in the center of the saturated zone plume at the end of the institutional control period. The use of contaminated water can result in radiological doses through a variety of pathways such as consumption of the contaminated water.

The scope of the evaluation requested by DOE in the IA for NRC's consultative review was to evaluate the impacts from the disposal of waste resulting from the separation, pretreatment, and vitrification of approximately 89 million liters (23.5 million gallons) of VLAW, from underground tanks at the Hanford Site in Washington. For the low-activity tank waste considered in the draft WIR evaluation, DOE plans to use a direct-feed low-activity waste (DFLAW) approach (DOE, 2020a). The DFLAW approach is a two-phased approach that will separate and pretreat supernate (essentially the upper-most layer of tank waste that contains lower concentrations of long-lived radionuclides) from some of the Hanford tanks, to generate a low-activity waste stream. The DFLAW approach will result in approximately 13,500 canisters of vitrified glass that will be disposed onside in the IDF. The DFLAW that will be disposed is about 29,000 m^3 (1,000,000 ft^3), which is roughly 10 percent by volume of the total VLAW glass that will be produced. In the Draft WIR Evaluation for VLAW, DOE did not provide dose estimates and charts for the DFLAW fraction (DOE, 2020a). Results were provided for the PA analyses of the IDF with some information on the portion of the doses attributed to different types of waste. The IDF will contain among the largest amounts of waste in the DOE complex and will have one of the largest inventories of long-lived radionuclides in a LLW disposal facility.

Figure 4-22 provides the all-pathways dose results for the deterministic base case analyses (RPP-RPT-59958, 2018). The solid red line shows the air pathway doses. The green dashed line is the groundwater pathway doses. The black dashed line shows the combined doses, and the red dash lines show the performance periods and dose limit. During the 1,000-year compliance period the doses are predominantly from the air pathway. The transport through the unsaturated zone is slow with breakthrough of most of the activity occurring around 1,300 years. The doses in the compliance period are more than 100 times below the limit of 0.25 mSv/yr (25 mrem/yr). In the sensitivity and uncertainty analyses period the peak doses are estimated to be approximately 0.02 mSv/yr (2 mrem/yr) or more than 10 times below the compliance period limit. The groundwater dose is primarily from ^{99}Tc and ^{129}I.

Figure 4-23 provides the concentration of ^{99}Tc in groundwater resulting from the disposal of different waste streams. Although it is a small fraction of the total waste volume to be disposed, the SSW is the dominant contributor to the peak groundwater concentration of ^{99}Tc. DOE showed similar behavior for ^{129}I. The DFLAW fraction of the vitrified glass waste identified on the figure as ILAW would contribute less than 5% to the peak dose.

In addition to the deterministic base case modeling that directly utilized process modeling with STOMP, DOE also developed a probabilistic system model that relied on abstractions of the more complex process models. DOE compared select results for the deterministic base case model and the probabilistic system model to demonstrate that the results were sufficiently similar. The probabilistic system model was used to examine the global impacts of uncertainty as discussed in Section 4.13. Figure 4-24 provides the results of the probabilistic analyses.

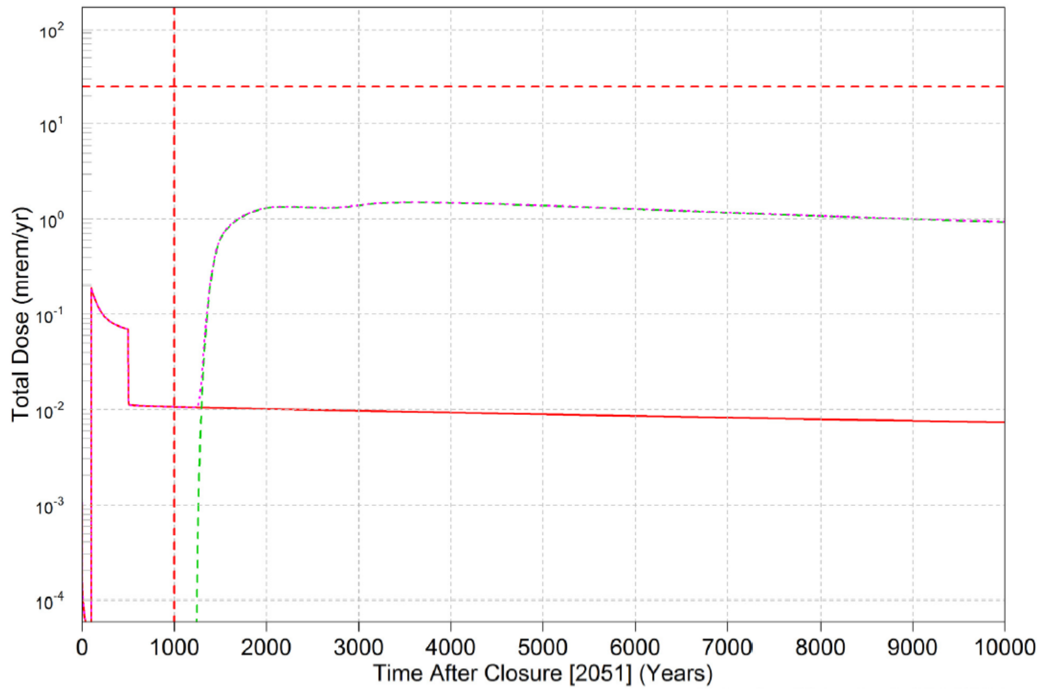


Figure 4-22 All-pathways Dose Results for the Deterministic Base Case
[Figure 1-13 from (RPP-RPT-59958, 2018)]

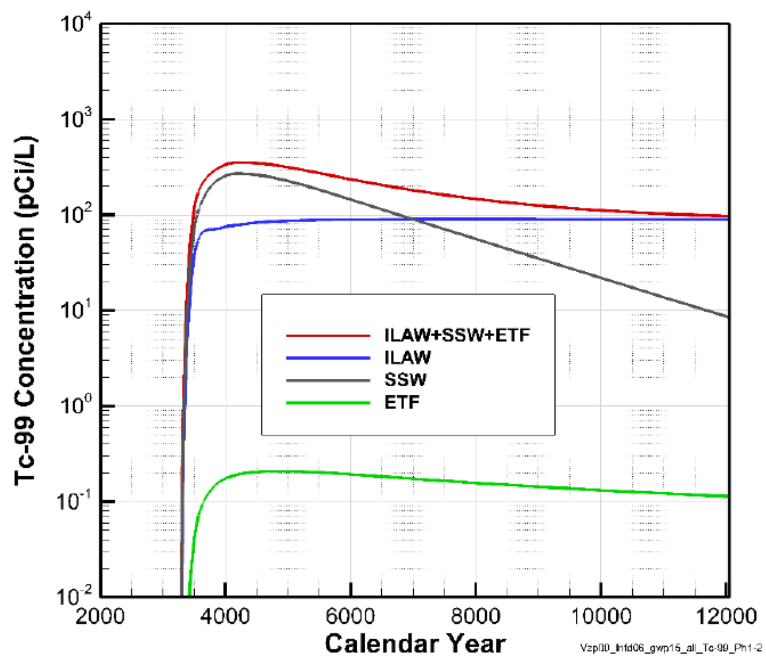


Figure 4-23 Contributions of Different Waste Streams to ^{99}Tc Concentrations in Groundwater
[Figure 1-7 from (RPP-RPT-59958, 2018)]

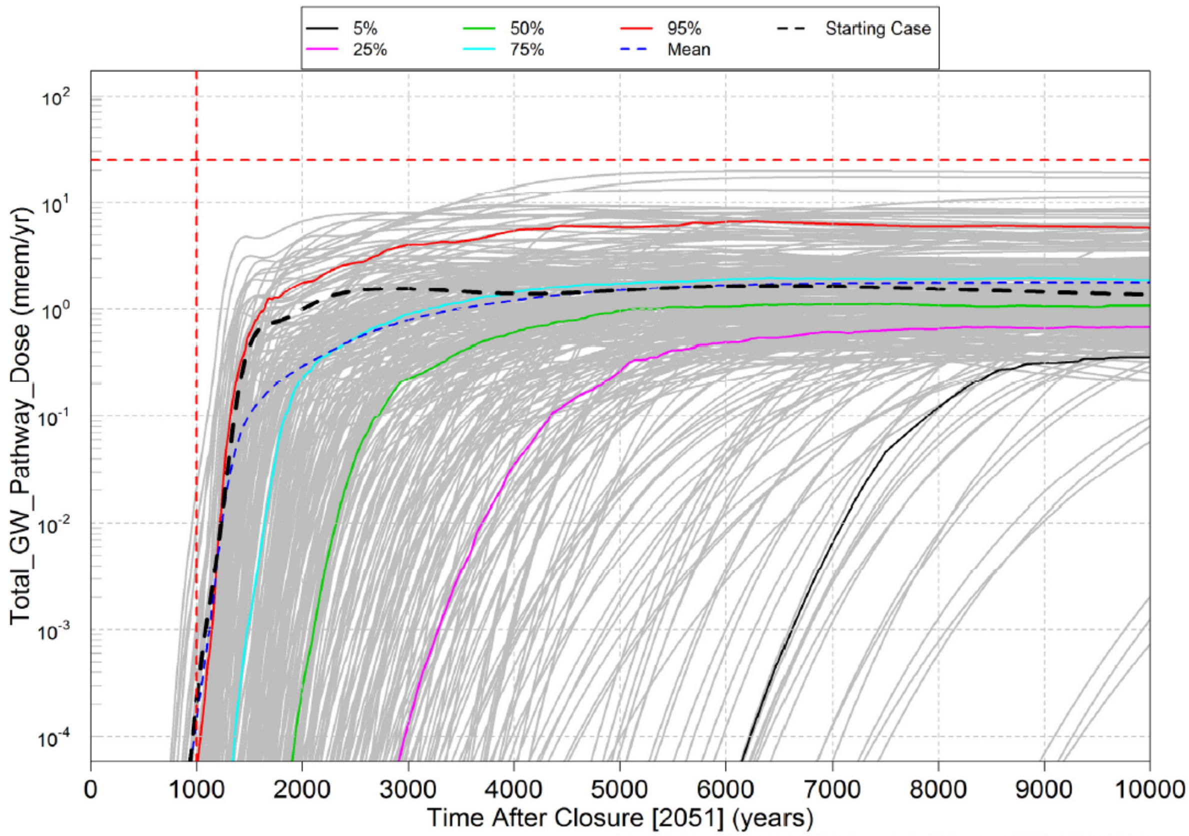


Figure 4-24 Probabilistic Analyses Results Using the System Model
[Figure 1-16 from (RPP-RPT-59958, 2018)]

The figure includes the combined results for all wasteforms disposed in the IDF. Estimated doses in the compliance period were less than 1 x 10^{-3} mSv/yr (0.1 mrem/yr) for the groundwater pathway during the compliance period. In the sensitivity and uncertainty analysis period the peak of the median result was less than 0.01 mSv/yr (1 mrem/yr) and the peak of the 95^{th} percentile result was about 0.06 mSv/yr (6 mrem/yr). None of the realizations in the probabilistic assessment exceeded 0.25 mSv/yr (25 mrem/yr). Two radionuclides contributed most of the dose (^{99}Tc and ^{129}I). As discussed in Section 4.13, a limited number of parameters contributed significantly to the spread or uncertainty in the probabilistic results. For the 1,000-year compliance period the most important parameters were the background infiltration rate, the K_d for Tc in the H2 unit, a parameter related to glass release, and a parameter related to the amount of dilution expected in the saturated zone.

DOE examined different sensitivity cases to explore the influence of changes in data and models or interpretation of data. DOE stated that some of these cases could be thought of as assessing alternative conceptual models rather than examining changes to individual parameters. The sensitivity cases explored many different topical areas pertinent to the PA such as changes to glass performance. The results of those analyses were previously shown in Figure 4-20. In Table 6-23 of the PA document DOE provided a summary of peak dose results for the sensitivity cases (RPP-RPT-59958, 2018). No sensitivity case exceeded the 0.25 mSv/yr (25 mrem/yr) dose performance objective in the 1,000-year compliance period. In all cases, the peak dose within 1,000 years occurred at 1,000 years after closure. In the sensitivity and uncertainty analysis period (1,000 to 10,000 years post-closure), only the Inventory Cases 7b, 10A, and 10B approached or exceeded the 0.25 mSv/yr (25 mrem/yr) dose performance objective (Figure 6-110 of the PA document (RPP-RPT-59958, 2018)). These cases had a common assumption of a much greater fraction of either the ^{99}Tc or ^{129}I inventory being assumed to partition to the secondary wasteforms. DOE indicated that the inventory cases mostly reflected analyses of choices associated with whether to use recycle in glass production. DOE provided the results of many analyses of individual sensitivities to help build an understanding of the system and to develop confidence in the robustness of the results. For example, Figure 4-25 provides the results for sensitivity analyses with respect to glass performance. The different analyses and supporting technical basis collectively provided the demonstration of compliance with the performance objectives.

In addition to the groundwater pathway, DOE estimated potential releases to and resultant doses from the air pathway. Normally an all-pathways dose would be compared against the performance objectives but in the case of disposal of waste at IDF the gaseous releases were estimated to occur well before the releases to groundwater and therefore it was useful to examine them separately. Within the period of compliance, the radionuclides that can migrate to the surface by gaseous diffusion included ^{222}Rn, ^{3}H, ^{14}C, and ^{129}I. DOE assumed the waste packages are no longer airtight immediately following closure of the facility. The radionuclide ^{3}H was released fastest and resulted in an estimated dose to a receptor located 20 km (12 mi) away of 8.8×10^{-6} mSv/yr (8.8×10^{-4} mrem/yr). Within two years the ^{3}H is dispersed. The radionuclides ^{14}C and ^{129}I are released more slowly than ^{3}H but result in a larger dose because the exposure occurs more than 100 years after closure and the receptor is therefore located 100 m (330 ft) from the facility. The peak dose from the air pathway was estimated at 1.9×10^{-3} mSv/yr (0.19 mrem/yr) which was well below DOE's 0.1 mSv/yr (10 mrem/yr) performance objective.

DOE estimated radon fluxes across the surface of the facility. DOE did not include doses from radon in the all-pathways dose assessment. DOE estimated that the flux of radon at the surface of the disposal facility will be more than three orders of magnitude below the DOE performance objective of 20 pCi/m^2/s. The ETF-LSW and SSW wasteforms were the primary contributing sources to the radon flux, in part because of the greater diffusive release of ^{222}Rn from these wasteforms and the smaller surface area occupied by these wasteforms.

In addition to the technical analyses, the technical basis for the analyses provided key information supporting the demonstration of compliance with the performance objectives. DOE's technical basis for the performance assessment was summarized in the previous sections of this report including the section on model support.

4.15.1.2 Protection of the Inadvertent Intruder

Protection of the inadvertent intruder following the institutional control period is provided in NRC requirements in 10 CFR 61.42, "Protection of individuals from inadvertent intrusion". It is not envisioned that the land above disposed radioactive waste will be used by a member of the public in the future. Because of uncertainties about societal use of resources and the durability of passive controls, protection of a member of the public who unknowingly intrudes into disposed waste is provided by technical assessment of a hypothetical intrusion scenario and associated dose limits.

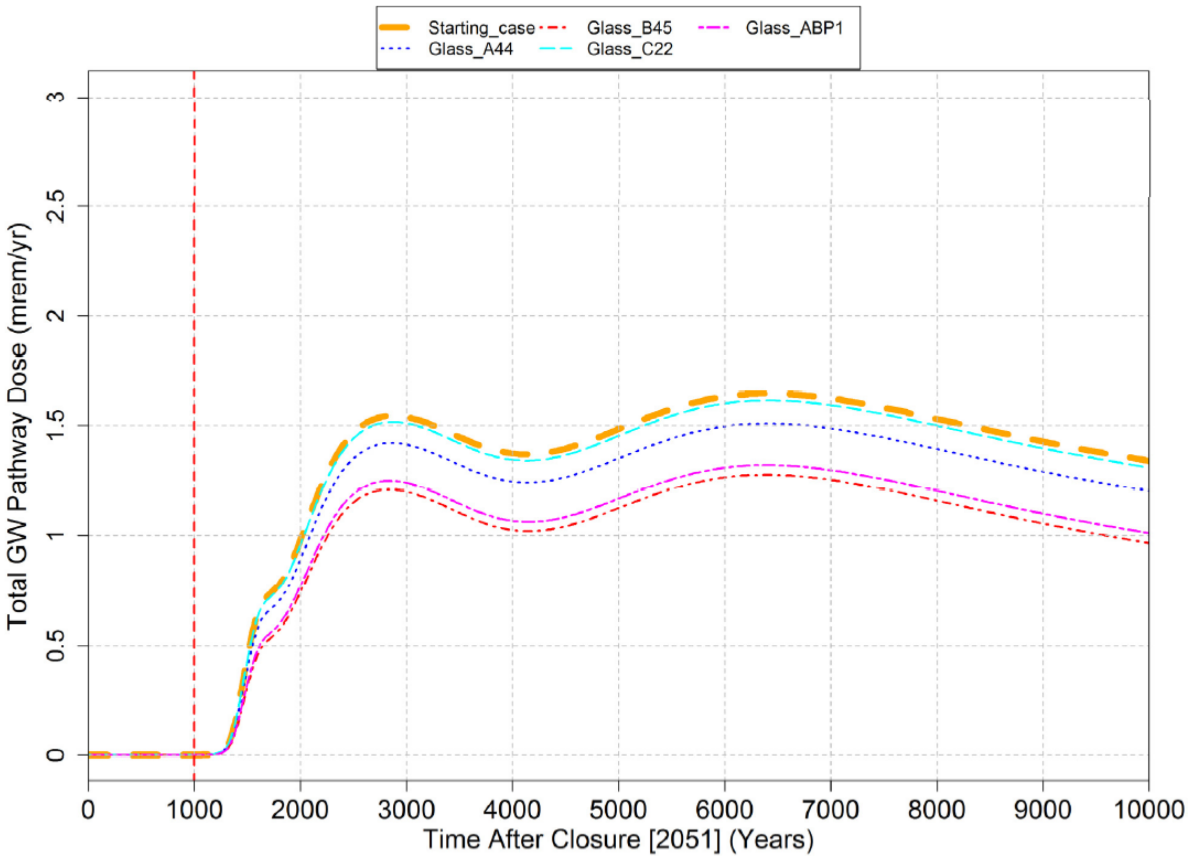


Figure 4-25 Sensitivity of Groundwater Pathway Doses to Glass Composition
[Figure 6-95 from (RPP-RPT-59958, 2018)]

NRC requirements state:

> "Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed."

The DOE requirements for protection of individuals from inadvertent intrusion are found in DOE M 435.1-1, Chapter IV.P.(2)(h). DOE stated that their performance measures for the hypothetical human intruder are more stringent than the dose limit used by NRC. Typically, NRC applies a dose limit of 500 mrem/yr to assess compliance with the requirement at 10 CFR 61.42 (NRC, 2007), whereas DOE imposes a 1 mSv/yr (100 mrem/yr) and 5 mSv/yr (500 mrem/yr) total effective dose (excluding radon in air) for chronic and acute inadvertent human intruder exposures, respectively. DOE used technical analyses in the IDF PA to show that there is reasonable expectation that the performance measure related to inadvertent intrusion will be met. For intruder analyses, DOE assumes institutional controls will be effective in deterring intrusion for no more than 100 years following closure.

For waste that is buried shallow (within 3 m (10 ft) of the land surface) an excavation scenario would be assessed but is not relevant for disposal of waste in the IDF. An excavation scenario can result in much more waste being disturbed and, therefore, higher concentrations for exposure of an intruder. Because waste will be buried and the disposal facility will have a thick engineered cover, the disturbance mechanism by the intruder was assumed to be installation of a well to use water resources. A member of the public, without knowledge of the disposal site, could drill a water well through the waste and disposal container bringing drill cuttings to the ground surface, resulting in an acute dose to the driller. After the drill cuttings are brought to the surface, a member of the public could reside in the area and perform normal present-day activities such as gardening or farming. DOE evaluated three types of chronic scenarios. The most limiting scenario was the rural pasture resident because it has the lowest amount of mixing of cuttings with uncontaminated soil and therefore the highest concentrations of radionuclides. The projected dose for the rural pasture scenario was about a factor of 2 to 5 higher than the other chronic scenarios. DOE used the average concentration of radionuclides that was anticipated to be produced for each wasteform type (e.g., glass, cement). DOE provided the inputs used for the calculations in Section 7 of the PA document (RPP-RPT-59958, 2018).

Figure 4-26 shows the projected potential doses to an inadvertent intruder for the DOE base case. Figure 4-26(a) is the projected acute doses whereas Figure 4-26(b) is the projected chronic doses. The highest doses (acute and chronic) to a hypothetical human intruder occurred immediately following the assumed loss of institutional controls. Doses decrease with time due to radioactive decay. The estimated peak acute dose was 0.093 mSv/yr (9.3 mrem/yr). The estimated peak chronic dose was 0.433 mSv/yr (43.3 mrem/yr). The estimated doses were below DOE's performance objective and well below NRC's performance objective which is less restrictive for chronic doses.

The peak dose for the acute scenario was driven by the external exposure pathway and was principally due to ^{137}Cs, which contributed over 90% of the total dose. This result shows the importance of using TSCR (ion-exchange columns) to remove most of the ^{137}Cs from the waste streams. For the chronic scenarios, the highest dose came from the glass wasteform. The peak dose was 0.433 mSv/yr (43.3 mrem/yr). The peak dose was driven by the milk ingestion pathway which contributed over 90% of the dose. The total dose was principally due to ^{90}Sr and ^{99}Tc, which contributed 0.298 mSv/yr (29.8 mrem/yr) and 0.104 mSv/yr (10.4 mrem/yr), respectively, to the total dose. Because short-lived radionuclides (e.g., ^{137}Cs and ^{90}Sr) contributed a significant fraction of the total dose the results were sensitive to the assumed length of the institutional control period.

DOE performed sensitivity and uncertainty analyses for the inadvertent intruder assessment. DOE did not evaluate all parameters in the uncertainty analyses. DOE demonstrated (Section 7.5.2 of the PA document) that the chronic dose received after an inadvertent intrusion was proportional to the concentration in the surface soils after the contaminated cuttings were spread and tilled into the soil (RPP-RPT-59958, 2018). Parameters that only affected the soil concentration, (e.g., the initial inventory in the waste, the area over which the cuttings are spread, tillage depth), would have a linear effect on the concentration (dose). Parameters that affected the soil concentration were not evaluated in the uncertainty analyses. Results showed the largest sensitivity was to soil bulk density. For the parameters and ranges evaluated, none of the results exceeded the DOE performance objectives.

(a)

(b)

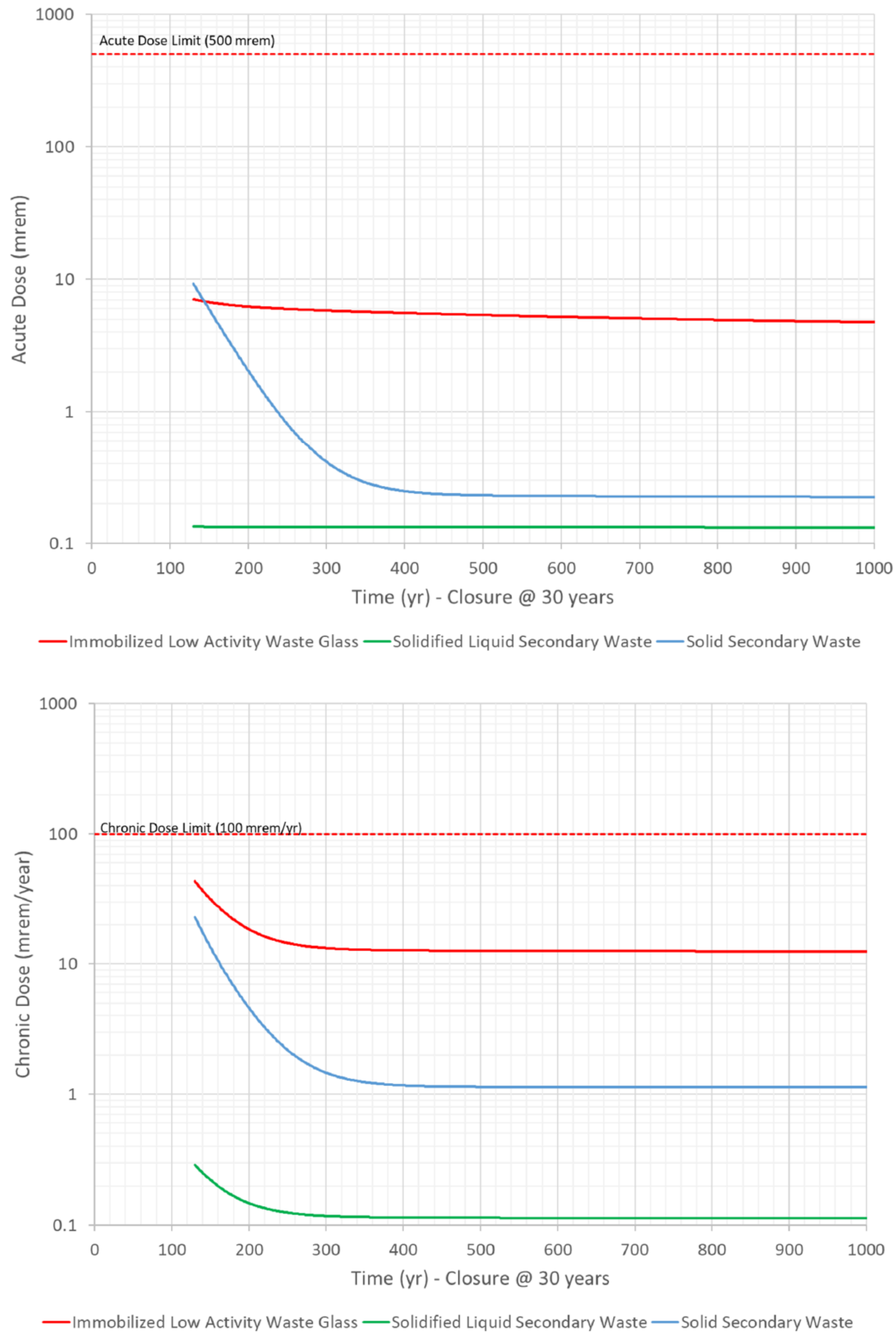


Figure 4-26 (a) Projected Acute Doses to an Inadvertent Intruder and (b) Projected Chronic Doses to an Inadvertent Intruder
[Figure 1-8 from (RPP-RPT-59958, 2018)]

4.15.1.3 Protection of the Public During Operations

Operations at the Hanford site have been ongoing for over 80 years. DOE requirements for occupational radiological protection are provided in 10 CFR Part 835, "Occupational Radiation Protection" (DOE, 1993), and those for radiological protection of the public and the environment are provided in DOE Order 458.1.

DOE provided a detailed comparison of the DOE requirements applicable to protection of the public during operations in Section 5.2.5 of the draft waste evaluation (DOE, 2020a). Members of the public could be workers or non-workers (offsite individuals, visitors, etc.). DOE has regulatory and contract requirements for DOE facilities and activities to ensure compliance with DOE's regulations at 10 CFR Part 835 and relevant DOE Orders that establish dose limits for the public and the workers during operations. DOE's regulation at 10 CFR 835.101 (c) requires that each radiation protection program include formal plans and measures for applying the ALARA (as low as is reasonably achievable) approach to occupational exposures. The Hanford radiological protection programs include a wide range of controls such as established dose limits, administrative control levels, monitoring of individuals and work areas, control of radiation and contamination areas, use of warning signs and labels, radiation safety training, and formal plans and measures for implementing the ALARA process. For the period 2011 to 2015, the average dose for an exposed worker was 0.52 mSv/yr (52.2 mrem/yr) which is well below the equivalent NRC standard (WRPS-1603585, 2016).

The nearest members of the public (non-workers) during operations are 20 km (12 mi) from the facility in the direction of the prevailing wind. Releases during operations could be to the air, water, or soil. Because the IDF is not yet operating, releases have not occurred. DOE has established monitoring programs to identify releases if they were to occur. Emissions are not likely during operations because the wastes will be immobilized in a solid physical form thereby limiting the release of contained radioactivity. The most likely exposure pathway for a member of the public is air because the transport is relatively rapid to an offsite location and releases to air generally cannot be contained once they occur. Because of the large distance to a member of the public significant dispersion is likely reducing concentrations at the point of exposure. The estimated dose per year from all operations at the Hanford Site from airborne emissions to the maximally exposed individual member of the public located at or beyond the Hanford Site boundary ranged from 7.9×10^{-5} to 1.2×10^{-3} mSv (0.0079 to 0.12 mrem) from 2004 through 2013 (RPP-RPT-59958, 2018).

DOE stated that operations at the IDF will be conducted in compliance with the standards for radiation protection set out in 10 CFR Part 20 and 10 CFR Part 835. Every reasonable effort will be made at the IDF to maintain radiation exposures as low as is reasonably achievable. Measures that provide reasonable expectation that operations at the IDF will comply with the applicable dose limits and with the ALARA provisions include the documented radiation protection program; design, regulatory, and contractual enforcement mechanisms; and access controls, training, and dosimetry.

4.15.1.4 Site Stability

Site stability is the maintenance of as-disposed conditions to the extent practical to limit dispersion and release of the waste to the environment. For low-level waste disposal facilities, the primary considerations associated with stability are to limit erosion and to limit subsidence.

Erosion can lead to direct releases or exposure of contained waste while erosion and subsidence can lead to enhanced releases through other pathways such as air or water.

The IDF is an expandable, RCRA-compliant landfill with double-lined trenches and a leachate collection system. The facility will be filled with waste and backfill, and then covered with an engineered cap designed to limit infiltration to the buried waste and prevent direct exposure with the waste. Stability will be provided by the wasteforms, backfill, engineered cover, and the site characteristics. DOE stated that the stability of the IDF after closure is enhanced by its location within the 200 East Area in the Central Plateau region near the center of the Hanford reservation, far from site boundaries, and approximately 200 m (650 ft) above mean sea level. At this location, the potential for surface water or groundwater contacting the waste and facilitating contaminant migration is low. For example, the semi-arid environment has an average annual rainfall of approximately 17.0 cm/yr (6.7 in/yr). The IDF site would be unaffected by flooding of the Columbia River, which is more than eleven kilometers (seven miles) from the IDF site. The potential for groundwater to affect the buried waste is low. The thickness of the unsaturated zone beneath the buried waste is over 80 m (200 ft).

Since the early 1940s, a large volume of information on the geology, seismology, and volcanology at the Hanford site has been collected and evaluated (DOE/RL-2002-39, 2002; PNNL-14586, 2005; SGW-48478, 2012). That information has been evaluated by numerous regulatory and other stakeholder groups. The high level of oversight has helped ensure a rigorous understanding of bounding geologic, seismic, and volcanic risks.

The two types of volcanic hazards that affected the Hanford site in the past 20 million years were continental flood basalt volcanism and volcanism associated with the Cascade Range. Continental flood basalt volcanism is no longer active in the Hanford region. Volcanism associated with the Cascade Range is active but if volcanism were to occur it is not anticipated to impact the performance of the IDF because of the distance from the IDF. DOE stated that the Central Plateau region is relatively stable seismically. The largest known earthquake in the Columbia Plateau occurred in 1936 near Milton-Freewater, Oregon, over 100 km (70 miles) from the IDF and on the far side of the Columbia River with a magnitude of 5.75 on the Richter scale. Figure 4-27 is a map of known seismic events at and around the Hanford Site between 1890 and 2005 (PNNL-6415, 2007). DOE examined a rectangular area around the Hanford site 200 East and 200 West areas and identified 78 earthquakes that have been recorded in the vicinity of the IDF with a magnitude between -0.8 and 2.8. Earthquakes of this magnitude are anticipated to have minimal impact on the stability of the disposal facility after closure. Range fires occur regularly throughout the Hanford site. DOE evaluated the impact of fire on the performance of a prototype engineered cover system and did not identify significant impacts to the performance of the cover (PNNL-18934, 2009).

The glass wasteform container is designed to provide stability of the disposal facility. The vitrified wasteform will have high compressive strength and the roughly 10% void space inside each disposal container will be filled with inert material such as sand. Limiting void space in the design will help to prevent subsidence. Soil used as backfill will be compacted to limit subsidence. The surface barrier that will be emplaced over the waste and backfill is designed to limit damage caused by wind and water erosion. DOE has developed a preliminary closure plan for the IDF. DOE described the use of a modified RCRA Subtitle C multi-layer barrier a minimum of 5 m (16.4 ft) thick above the upper-most level of wastes. The plan was updated in

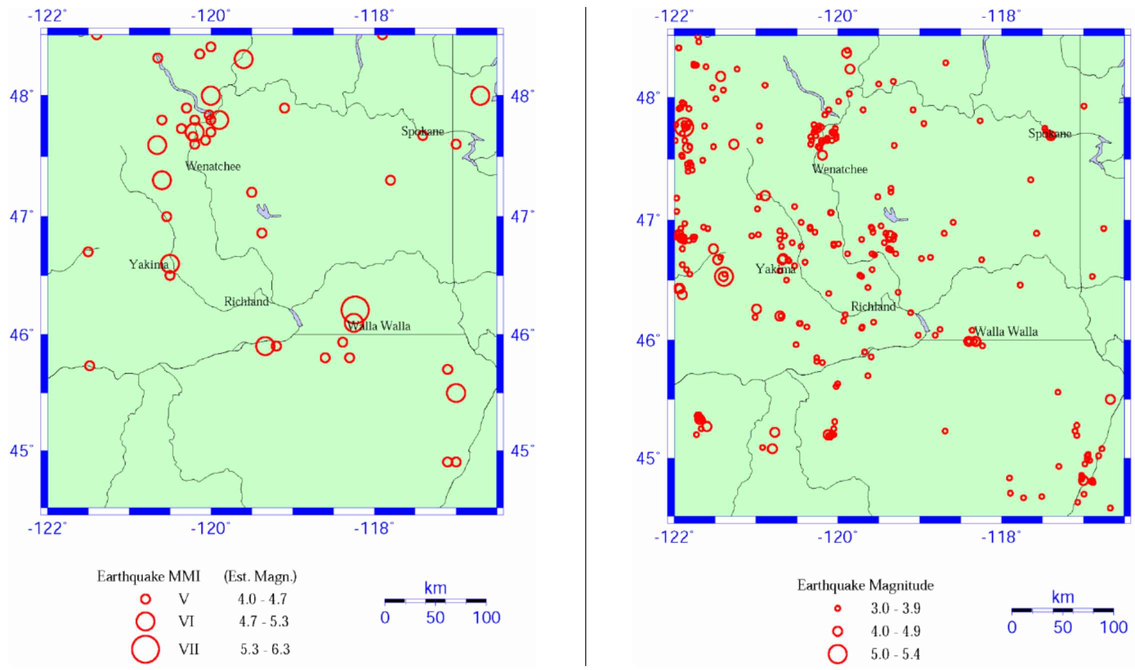


Figure 4-27 Distribution of Earthquakes Between 1890 and 2005 at Hanford (Left 1890 to 1970, Right 1970-2005)
[Figure 3-14 from (RPP-RPT-59958, 2018)]

2019 to align with the IDF PA analyses (CHPRC-03407, 2019). DOE stated that the plan will be updated, as necessary, throughout the operational life of the disposal site. DOE indicated that the setting for the IDF and implementation of the preliminary closure plan and its updated versions will ensure that the applicable requirements in DOE Manual 435.1-1 concerning site stability will be met for the IDF.

4.15.2 NRC Evaluation of DOE's Demonstration of Compliance with NRC's Performance Objectives

The NRC previously reviewed DOE's demonstration of compliance with NRC's performance objectives for application to WMA C (NRC, 2020b). Some aspects were essentially identical for disposal of VLAW at the IDF and WMA C. Those aspects included the demonstration that DOE's requirements were essentially equivalent to NRC's requirements, demonstration of compliance with 10 CFR 61.42 (in terms of the assessment methods and data used), and demonstration of compliance with 10 CFR 61.43. NRC reviewed the DOE documentation, supporting references, and computer model files. NRC performed select independent checking and independent calculations. NRC also reviewed technical reports and literature not directly referenced in the PA to compare against the information provided by DOE. The sections that follow provide the results of NRC's review.

4.15.2.1 Protection of the Public After Closure

The NRC staff reviewed DOE's all-pathway dose, groundwater dose, air dose, and radon flux analysis results developed with deterministic, probabilistic uncertainty, and sensitivity analyses.

The NRC staff considered the different analyses when determining if DOE demonstrated compliance with 10 CFR Part 61.41.

DOE specified a compliance period of 1,000 years for demonstration that the performance objectives will be met. For commercial low-level waste disposal NRC does not specify a compliance period and values longer than DOE's 1,000 years are currently applied for the assessment of the disposal of long-lived waste. In this consultive review under the interagency agreement NRC reviewed the results against DOE's compliance period considering the longer-term impacts provided in the sensitivity and uncertainty analyses period.

DOE's deterministic base case performance assessment modeling resulted in all-pathway doses during the compliance period of well less than 0.01 mSv/yr (1 mrem/yr) to an offsite member of the public. The dose was from the air pathway. The modeling DOE used had numerous conservative features. The containers were assumed to release gaseous radionuclides instantaneously at the time of closure. Carbon, and especially stainless steel, are likely to provide containment of waste for some time after disposal. Some containers may have defects or may leak, but most will likely not leak. Steel containers that degrade will not form perforations at the same time. A failure time distribution is common and the spread in the distribution could be quite broad depending on the rate of corrosion. DOE elected not to take credit for these aspects when estimating gaseous releases. The largest contributor to the air pathway dose in the compliance period was ^{129}I. DOE assumed a Henry's Law constant corresponding to an undissociated species of I_2. This assumption led to a greater predicted release rate for ^{129}I. NRC agrees that the air pathway doses from release of gaseous radionuclides have likely been overestimated and that DOE has demonstrated that the air pathway performance objective of 0.1 mSv/yr (10 mrem/yr) has been met.

Transport of radioactivity through the water pathway takes longer than 1,000 years under conditions of the base case analyses. Most of this time of transport is in the unsaturated zone. The time of transport through the saturated zone is comparably rapid and can essentially be neglected. Only very mobile radionuclides are expected to reach a hypothetical receptor within 1,000 years after closure. The radionuclide ^{99}Tc is expected to arrive first followed by ^{129}I. The saturated zone has a very important role in providing dilution of contaminant fluxes before the contaminants reach a potential receptor's well. Though NRC had comments about saturated zone hydraulic conductivity values (and specific discharge), the saturated zone will provide a significant and durable reduction in risk. NRC was able to verify the unsaturated zone travel time estimates as well as the amount of dilution provided by the saturated zone. DOE has demonstrated that the all-pathways performance objective will be met with reasonable assurance for the compliance period.

Part of the evaluation of sensitivity and uncertainty analyses is to determine if uncertainties can, individually or in combination, result in exceedance of the performance objectives. NRC reviewed the sensitivity and uncertainty analyses provided by DOE. NRC reviewed the supporting technical documents to determine if DOE identified the significant uncertainties and estimated the impact of those uncertainties. DOE clearly described the sensitivity and uncertainty analyses that were performed. For sensitivity analyses, the parameters evaluated and the values assigned were provided. DOE discussed the results and provided accompanying charts and tables. For uncertainty analyses, DOE described the parameters that were uncertain, the probability distributions assigned for the parameters, and the results of the analyses.

No sensitivities were identified that would result in exceeding the period performance objective during the compliance period. The only topical area that had the potential by itself to exceed the compliance period performance objective during the sensitivity and uncertainty analyses period was the fraction of volatile isotopes that would be retained in the glass compared to the secondary wastes. If recycle is not used during vitrification (inventory cases 10A and 10B), the retention of volatile species in glass is sufficiently low such that the 0.25 mSy/yr (25 mrem/yr) performance objective could be exceeded. DOE used this result, in part, to select recycle as part of the reference design. For inventory cases 7b and 7c, the inventory of ^{129}I and ^{99}Tc increased by 64% and 12% respectively in the secondary wastes over the base case. The doses for case 7c increased by a proportional amount. The doses for case 7b increased by a much larger amount and approached the compliance period performance objective during the sensitivity and uncertainty analyses period. DOE explained that the dose result in case 7b was not proportional to the 64% increase in inventory of ^{129}I or the 12% increase in ^{99}Tc; the increase in total dose was much higher and can be attributed to the which secondary wasteforms the inventory was assigned to. This result reflects uncertainty in the secondary waste split, or which type of secondary waste the volatile species are retained in. Although the percentage of the total ^{99}Tc inventory disposed in the IDF in a grouted wasteform was not high (~1.7% in this inventory case), the different allocation increased the ^{99}Tc inventory in SSW by more than a factor of 20 over the base case. DOE's explanation of the results was complete and technically sound.

DOE showed that the compliance period doses were sensitive to changes in infiltration rates. This is because the contaminants do not reach the receptor in the compliance period because of the delay afforded by the unsaturated zone. If background infiltration rates are higher contaminants could reach the receptor in the compliance period. Though distribution coefficients for ^{99}Tc in an oxidizing environment are expected to be very small or near zero, small changes in the distribution coefficient can also impact the transport time. DOE also provided sensitivity analyses for the air pathway. None of the sensitivity cases resulted in doses exceeding the air pathways performance objective. The parameters examined by DOE were appropriate, and as discussed previously, were all based on the same conservative assumptions with respect to failure distributions for containers. Based on the information provided, DOE used proper methods to identify sensitive parameters.

Sensitivity analyses are useful to understand or evaluate the impact of a single or few uncertainties but can be more difficult to use and interpret when there are many uncertainties. The PA for Hanford has many uncertainties, most of which DOE assessed in the PA analyses or in response to NRC's RAIs (RPP-RPT-59958, 2018; DOE, 2021c). When uncertainties are combined in a sensitivity analysis the probability portion of the risk triplet could be increasingly unlikely. While the magnitude of the result may be substantial, the decrease in likelihood of occurrence could offset the increase in magnitude. DOE's uncertainty analyses were used to examine the global impact of uncertainties. Though DOE indicated that 98 parameters were included in the uncertainty analyses, many of the parameters that were uncertain were K_d values. The number of technical areas that were uncertain were more limited. As previously discussed, the methods DOE used for the uncertainty analyses were appropriate. As a result of uncertainty, some individual realizations showed groundwater doses could occur within the compliance period but the magnitude was very small (< 1 x 10^{-3} mSv/yr (0.1 mrem/yr)). The peak of the mean dose that occurred in the sensitivity and uncertainty analyses period was in general agreement with that observed with the deterministic base case – well below the compliance period performance objective. No individual realizations exceeded the 0.25 mSv/yr (25 mrem/yr) compliance period performance objective even well after 1,000 years after closure.

NRC requested additional information on DOE's evaluation of uncertainty (NRC, 2020a). NRC identified uncertainties that were not part of DOE's sensitivity or uncertainty analyses provided in the PA. In response to the RAI, DOE provided the results of new sensitivity analyses (DOE, 2021c). From the results of those analyses DOE did not identify new (single) uncertainties that could result in exceedance of the compliance period performance objective. For an exceedance to occur, an uncertainty or combination of uncertainties would need to shift the arrival of contaminants earlier and increase in magnitude. Those combinations, especially the shift in arrival time, are unlikely to very unlikely. DOE has demonstrated compliance with the 10 CFR Part 61.41 performance objective for the DOE compliance period.

There is not a requirement to demonstrate compliance with the performance objective after the compliance period. The uncertainties discussed previously in this report such as glass dissolution rate parameters, the deposition of volatile species inside glass containers, the SMRN, the potential for Stage III behavior of glass, and the inventory partitioning of volatile radionuclides in different waste streams could in combination increase (or decrease) the estimated doses during the sensitivity and uncertainty period. The uncertainties are unlikely to also shift the doses much earlier in time such that the compliance period performance objective would be exceeded.

Radon fluxes were estimated to be well below the standard by DOE. There is limited inventory of isotopes that decay to radon gas in the residual tank waste. A few tanks contain high amounts of uranium but collectively the amount of uranium and radium is limited. When the waste is disposed in the IDF, the disposal cell itself is quite deep and the design calls for a thick engineered cover to be placed over the waste. This cover will have layers to divert flow of infiltration into the waste, but some of these layers will also limit flow of gases to the land surface. Radon fluxes are quickly reduced by thick covers and if liquid saturations are high. Using simple modeling DOE demonstrated that radon fluxes are likely to remain well below the standard for an offsite member of the public.

4.15.2.2 Protection of the Inadvertent Intruder

NRC staff previously reviewed protection of the public during operations (10 CFR Part 61.43) with respect to closure of WMA C. The data and analyses methods were nearly identical for analysis of disposal of VLAW at the IDF, though there were some differences in timing of the scenarios resulting from assumptions about the robustness of engineered features.

Inadvertent human intruders are members of the public who may use the site at some time in the future after active institutional controls are no effective but be unaware of the presence of the radioactivity. DOE assumed that an inadvertent intrusion event could occur at 100 years following closure of the disposal facility (year 2151). This assumption is consistent with NRC guidance and requirements within Part 61. DOE also provided information that supported consideration of a longer institutional control period at Hanford. Because of significant environmental contamination issues, DOE expects to be completing remediation activities at the Hanford site well into the foreseeable future. The NRC reviewed the information provided and conclude that the assumption of a 100-year effective institutional control period at Hanford is likely to be conservative.

DOE evaluated both acute and chronic exposure scenarios for an inadvertent intruder. In developing the intruder scenarios, DOE assumed that humans would continue the land use activities that are consistent with past (e.g., recent decades) and present regional practices after the end of the active institutional control period. Because of the depth to waste, DOE did not quantify the impacts resulting from an excavation scenario. DOE provided appropriate basis that an excavation scenario was not credible based on the current design. For the analyses of intrusion at the IDF, DOE did not credit any engineered features (outside of depth for excavation) as being a deterrent to intrusion. The waste will be disposed in steel containers that may retain the characteristics of steel at the time of intrusion or for some period after that. DOE did not credit the characteristics of the waste containers in the analyses. NRC reviewed DOE's calculations and verified that the dose results were consistent with the inputs and assumptions. The projected acute and chronic dose results from intrusion into average waste concentrations are expected to be below the NRC and DOE performance objectives.

DOE did not include uncertainties or variability in the inadvertent intruder assessment that would directly impact soil concentrations and therefore intruder doses because the cause-and-effect relationship was linear and proportional. NRC does not support this approach if the objective of uncertainty analyses is to develop confidence that uncertainty and variability will not significantly impact the demonstration of compliance with the objectives. DOE's approach eliminates from further consideration key sources of uncertainty and variability with respect to establishing waste acceptance criteria. For example, the concentrations of radionuclides in the waste that will go through the VLAW processing steps is uncertain (NRC, 2020a). If this uncertainty was large enough it could directly impact the confidence that the performance objectives have been demonstrated with reasonable expectation. DOE's estimates of the inventory of key radionuclides have, over time, decreased more than they have increased. Uncertainty remains and it is likely to remain when wasteforms are being produced and disposed in the IDF. NRC staff recommends that all significant uncertainties should be included in uncertainty analyses for the inadvertent intruder (Recommendation #26). Inclusion of significant sources of uncertainty may impact design decisions. Inventory uncertainty, by itself, is not likely to invalidate DOE's conclusions but when combined with other uncertainties it may influence interpretation of the significance of remaining uncertainties.

NRC asked for additional information on the impact of variability in waste concentrations on the results of the intruder assessment (NRC 2020a). The variability of the waste will result from processing of waste from different tanks in different tank farms. DOE had initially presented information associated with the average waste. DOE presented information associated with waste variability and how it may impact demonstration of compliance with the performance objectives (DOE, 2021c). DOE explained that the intruder dose model was used to compute the concentration of each radionuclide in the impacted waste that would yield a dose to the intruder that is equal to 5 mSv (500 mrem) for an acute 40-hour exposure or 1 mSv/yr (100 mrem/yr) for a chronic annual exposure, consistent with DOE's performance measures for protection of the human intruder in DOE M 435.1-1. Because the IDF waste will be disposed in four lifts, with each lift containing one vitrified waste container or two stacked other containers, the resulting concentration limit is the average concentration in the impacted containers and is not a limit for each container. The resulting waste concentrations were incorporated into the IDF WAC, IDF-00002, Table G-1 (IDF-00002, 2019). The disposal limits for short-lived radionuclides were sensitive to the time of the inadvertent intrusion. In the analysis in response to the RAI, the time of intrusion (2278) was set to the minimum time that DOE must retain excavation restrictions (through institutional controls) on the Hanford Site to provide reasonable expectation that the dose to an inadvertent intruder would not exceed DOE's performance measures. The recommended institutional control period for the entire Hanford Site is the latest institutional control date specified for a waste site on the Hanford Site in DOE/RL-2001-41, *Sitewide Institutional Controls Plan for Hanford CERCLA Response Actions and RCRA Corrective Actions* (DOE/RL-2001-41, 2019). Using this date for intruder analyses was the recommendation from DOE-0431, *Recommendations for Institutional Control Time Period for Conducting DOE Order 435.1 Performance Assessments at the Hanford Site* (DOE-0431, 2019).

DOE has effective controls and a strategy in place to account for waste variability. Without an extended institutional control period (to year 2278 or 237 years) or operational constraints regarding stacking and placement of waste packages, waste variability could cause DOE's performance objectives to be exceeded. Waste variability is much less likely to result in an exceedance of NRC's performance objective compared to DOE's performance objective because NRC uses 5 mSv/yr (500 mrem/yr) for a chronic intruder. In response to the RAI, DOE did not provide the expected values and range of concentrations that will be produced for each secondary waste stream type. Some low-volume, high-concentration wasteforms may not be suitable for disposal in the IDF without additional operational constraints. DOE should quantify the impacts associated with each waste type that will be generated from the production of VLAW (e.g., SSW, GAC, HEPA filters, AgM, waste generated by decommissioning the facility) (Recommendation #27).

4.15.2.3 Protection of the Public During Operations

NRC staff previously reviewed protection of the public during operations (10 CFR Part 61.43) with respect to closure of WMA C. The criteria, requirements, procedures, and processes that DOE used to demonstrate that public health and safety would be protected during operations at the IDF was essentially identical to those for WMA C. NRC determined that the staff has reasonable assurance that 10 CFR Part 61.43 would be met.

DOE provided a detailed comparison of the DOE requirements applicable to protection of the public during operations in Section 5.2.5 of the draft waste evaluation (DOE, 2020a). Those requirements are sufficiently similar to NRC's requirements. The large distance between the facility and the nearest member of the public in the predominant direction of the wind provides a significant reduction if any releases were to occur during operations. DOE's historical estimates of the dose per year from all operations at the Hanford Site from airborne emissions ranged from 7.9×10^{-5} to 1.2×10^{-3} mSv (0.0079 to 0.12 mrem) from 2004 through 2013 (RPP-RPT-59958, 2018). This is a small fraction of the allowable annual dose.

DOE demonstrated that the Hanford radiological protection programs include a wide range of controls that NRC would require of a licensee (DOE is not an NRC licensee). DOE demonstrated the effective implementation of those programs for activities analogous to what will occur in the future at IDF. For the period 2011 to 2015, the average dose for an exposed worker was 0.52 mSv/yr (52.2 mrem/yr) which was a fraction of the allowable limit.

4.15.2.4 Site Stability

NRC staff previously reviewed site stability with respect to closure of WMA C. The wasteforms and facility for disposal of VLAW and other wastes at IDF are different. The site characteristics are the same from the perspective of potentially disruptive processes and events.

The NRC staff found the proposed design and closure activities to provide stability for the disposal site, disposal facility, and wasteform to be adequate. The analyses associated with siting considerations and disruptive processes and events such as flooding, liquefaction, seismicity, and volcanism were complete. The glass wasteform will have a very low porosity and void space in the containers (head space) will be filled with an inert material such as sand prior to closure. Backfilled soil that will be placed between waste containers will be compacted to limit subsidence and consolidation. NRC staff has reasonable assurance that the 10 CFR Part 61.44 performance objective will be met, and that the glass and cementitious wasteforms, filler material, waste containers, compacted backfill, and engineered cover will minimize void space and prevent subsidence and differential settlement that could occur due to consolidation.

As discussed in the TER for WMA C, the Hanford Barrier (an analog for the eventual closure cover at IDF) experienced erosion from a large thunderstorm event (NRC, 2020b). Because the closure cover for the IDF will use a similar design, NRC's previous recommendation (See Appendix A) associated with erosion design is valid for IDF (NRC, 2020b).

4.15.3 NRC Conclusions on DOE's Demonstration of Compliance with Performance Objectives

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, PA documentation, additional references, the PA model, and had technical discussions with DOE staff and contractors on the demonstration of compliance with the performance objectives for the IDF PA. DOE's performance objectives were sufficiently equivalent to NRC's. The NRC staff has the following conclusions:

- The information provided by DOE was clear and complete.
- DOE demonstrated with reasonable expectation that compliance with the performance objective for protection of an offsite member of the public after closure will be achieved.
- DOE demonstrated with reasonable expectation that compliance with the performance objective for protection of the public from inadvertent intrusion will be achieved.
- DOE demonstrated with reasonable expectation that compliance with the performance objective for protection of the public during operations will be achieved.
- DOE demonstrated with reasonable expectation that compliance with the performance objective for site stability will be achieved conditional on implementation of an engineered cover system with proper erosion protection.

The NRC staff has the following recommendations:

- DOE should include all significant uncertainties in uncertainty analyses for the inadvertent intruder. (Recommendation #26)

- DOE should quantify the impacts associated with each waste type that will be generated from the production of VLAW (e.g., SSW, GAC, HEPA filters, AgM, waste generated by decommissioning the facility). (Recommendation #27)

4.16 NRC Conclusions for Criterion B

The NRC staff evaluated the Draft WIR Evaluation for VLAW, the IDF PA and supporting documentation, computer models, quality assurance procedures and implementation, and many additional other documents and resources. As discussed in Section 1.4, the NRC staff considered all wasteforms generated from the processing of VLAW at the Hanford Site, in addition to DFLAW. Where necessary, the NRC staff performed independent calculations and analyses. Collectively, this information provides DOE's demonstration that Criterion B of DOE Order 435.1 is met with reasonable expectation.

In the NRC staff's review of WMA C at the Hanford Site, the NRC staff determined that the standards DOE applied to demonstrate compliance with DOE Order 435.1 are comparable to the requirements found in 10 CFR Part 61 (NRC, 2020b). DOE's use of "reasonable expectation" when compared to NRC's use of "reasonable assurance" is not materially different from a technical perspective. The requirements in 10 CFR 61.40 include that land disposal facilities be sited, designed, operated, closed, and controlled after closure such that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in 10 CFR 61.41 through 10 CFR 61.44. Because the IDF is already constructed, site selection was not part of the NRC staff's evaluation. The staff reviewed site characteristics as part of the PA. Site closure and institutional control will occur in the future and thus the NRC staff's evaluation was limited to review of the plans and expected outcomes.

As often necessary for complex technical analyses, DOE makes numerous assumptions either for simplification purposes or due to incomplete information. The conclusions of the NRC staff's review may be impacted if the following assumptions are not upheld:

- The closure year will be no sooner than 2051. The institutional control period will be a minimum of 100 years.
- DOE will demonstrate that minimal deposition of volatile species will occur within glass canisters during cooling and that volatile species in glass are relatively uniformly distributed.
- The closure cover will be installed and perform as designed.
- Stage III behavior of glass will remain unlikely in the IDF disposal environment.
- DOE will demonstrate that production-scale glass with recycle performs as well as laboratory-scale glass with no recycle.

The results of NRC staff's review related to Criterion B are as follows, with conclusions regarding the level of NRC staff's confidence that, given the existing uncertainties, DOE's demonstration of compliance will remain valid:

- DOE has demonstrated compliance with 10 CFR 61.41 (high).
- DOE has demonstrated compliance with 10 CFR 61.42 (moderate).
- DOE has demonstrated compliance with 10 CFR 61.43 (high).

- DOE has demonstrated compliance with 10 CFR 61.44 (high).

It would take many technical aspects of the assessment to change significantly in a negative direction for DOE's dose limit for an offsite member of the public after closure (10 CFR 61.41) in the 1,000-year compliance period to be exceeded. The impacts to a member of the public during operations are likely to be very small. The wasteforms are not likely to be subject to dispersal or fire and the distance to a member of the public is large. The demonstration of compliance with 10 CFR 61.44 is deemed to be high, however, is dependent on completion of a final erosion protection design and associated model support. The demonstration of compliance with 10 CFR 61.42 can be controlled by DOE identifying which wastes can be disposed in the IDF and implementing proper controls, design, and procedures for the disposal. DOE described the approach it will use to establish WAC and the NRC staff find it acceptable. However, if the analyses to establish the WAC do not account for each waste type and the variability of concentrations within a waste type, then the derived WAC may not sufficiently account for variability. DOE can eliminate this observation by analyzing all waste types including the range of concentrations expected within each waste type.

The recommendations provided in Section 4 are collated in Table 6-1. Table 6-1 identifies three categories of recommendations: (1) "DFLAW" means applicable to the Direct-Feed LAW that is evaluated in the Draft WIR Evaluation for VLAW; (2) "VLAW" means applicable to the remainder of the vitrified LAW or secondary wastes; and (3) "General" means if completed can improve the technical basis for DFLAW, VLAW, and/or future waste evaluations and is considered a best practice for performing waste evaluations. The recommendations from the WMA C TER that are applicable to VLAW are included in Appendix A.

5 CRITERION C – Assessment of Radionuclide Concentrations and Classification

This section summarizes the NRC staff's review of the information DOE submitted with respect to Criterion C of DOE Manual 435.1-1, which is demonstrating compliance that the waste (i.e., DFLAW) will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR 61.55 or will meet alternative requirements for waste classification and characterization as DOE may authorize.

Waste classification is a process through which the proper technical requirements for disposal of different wastes can be matched with the wastes being disposed. For the commercial disposal of LLW in the U.S., waste classification (along with other requirements) provides for the protection of an inadvertent intruder who may unknowingly contact waste in the future. For the disposal of DOE waste in a DOE disposal facility, a site-specific intruder assessment is completed which somewhat alleviates the need for waste classification from a public protection standpoint. The review entailed evaluation of the physical form of the waste and radionuclide concentrations and classification of the waste.

5.1 Waste Physical Form

In the Draft WIR Evaluation for VLAW, DOE indicated that LAW retrieved from underground storage tanks would be converted into glass or non-glass wasteforms. The non-glass wasteforms would be either solidified or encapsulated in cementitious forms. Some lesser quantities of other wastes would be generated, such as failed melters. The sections that follow address the different wastes that DOE anticipates will be generated from producing VLAW.

5.1.1 DOE's Assessment that Waste Will Be Incorporated In a Solid Physical Form

DOE anticipated two primary types of wasteforms will be generated from the processing of LAW – vitrified glass and cementitious. Vitrification of glass will generate two types of secondary wastes – liquid and solid. The secondary wastes are likely to be cementitious, but research is still ongoing, and DOE may elect to change the host matrix for those forms. The cementitious wasteforms will be either a solidified matrix with the waste distributed throughout or an encapsulated wasteform. See Table 2-1 for the volume of different waste streams that will be generated and disposed at the IDF. The volumes for secondary solid wastes (SSW) are as received at the treatment facility. The "as-disposed" volumes may differ depending on the treatment technologies used.

The product from the LAW Vitrification Facility will be vitrified borosilicate glass and the resulting immobilized waste will be poured into containers. The borosilicate glass will be highly stable with significant compressive strength and a high degree of chemical durability. Each container will be at least 90 percent filled. If necessary, inert material (such as glass or sand) will be added on top of the glass to meet this requirement (IDF-00002, 2019). The vitrified wasteform is anticipated to have a very low porosity of less than a few percent. The release rates from the glass were estimated to be fractions of a percent for 1,000 years.

The secondary wastes, both solid and liquid-solidified, will be more varied in their properties and packaging for disposal than the glass. DOE stated that secondary wastes would likely be disposed in discrete sections of the disposal facility because high alkalinity could impact the performance of glass wasteforms. Because the details of the SSW cementitious grout mix specification(s) and final disposal configuration for SSW had not been defined, DOE relied on available information from existing studies of cementitious materials considered representative of mixes that may be used for SSW encapsulation and/or solidification. SSW will be generated as the result of operations, in the form of radioactive debris and non-debris waste, such as melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, HEPA filters, and other process-related waste. Debris streams will undergo some type of volume reduction via compaction/super-compaction, sorting, and repackaging at an offsite treatment facility prior to disposal at the IDF (RPP-ENV-58562, 2016).

The ETF-generated LSW arise from treating liquid secondary waste from WTP operations. The operations include generation of wastes from low-level and mixed low-level liquid effluent from the melter primary off-gas treatment system, the LAW vitrification secondary off-gas/vessel vent treatment system, process vessel washes, floor drains, sumps, and vessel vent header drains. The characteristics of the resulting LSW grouts are addressed in the report PNNL-25194 (PNNL-25194, 2016). At the time of development of the PA, two waste matrices were under consideration for ETF wastes; a fly ash-based LSW grout and hydrated lime-based LSW grout (SRNL-STI-2016-00175, 2016). Different mixes were considered with dry bulk density ranging from 1.07 to 1.98 (g/cm^3) (66 to 123 lb/ft^3). Porosity ranged from 0.18 to 0.59.

Cementitious wastes are planned for disposal to reduce void space and decrease dispersion of the waste into the environment. DOE uses waste acceptance criteria (WAC) to determine what waste may be suitable for disposal in the IDF (IDF-00002, 2019). In the WAC, DOE specified that the wasteforms must have a minimum compressive strength of 0.6 megapascal (MPa) (85 psi). Cementitious wasteforms typically have compressive strength that is significantly higher than the WAC value.

Other wastes will undergo a variety of treatment processes to ensure that the waste will be incorporated into a solid physical form. Compaction and encapsulation with grout are the expected treatment processes. DOE has requirements in place as part of the WAC for the IDF to provide for filling of void spaces in waste containers.

5.1.2 NRC Review that Waste Will Be Incorporated In a Solid Physical Form

The NRC staff reviewed the information DOE provided that the wasteforms generated by the VLAW process would be incorporated into a solid physical form. In the Draft WIR Evaluation for VLAW, DOE provided information for the vitrified wasteform (DOE, 2020a). Liquid waste (which may contain suspended solids) will be combined with glass-forming precursors in the melter. After cooling, the wasteform will be solid with low porosity and sufficient compressive strength. In addition, the glass will be inside a stainless-steel container that may inhibit release for the foreseeable future. The void space in a container may be up to 15 percent. DOE stated that an inert material will be added to ensure that the void space is at most 10 percent. In terms of radioactive waste disposal, a 10 percent void fraction is low. The waste incorporated into the vitrified waste will be in a solid physical form.

The secondary wastes will be in a variety of wasteforms with different properties. Solidification and encapsulation will be used in concert with treatment processes such as compaction to reduce void space in the wastes to the extent practical. Cementitious wastes must cure to produce a solid physical form. Some wastes can sometimes inhibit the curing process. DOE experienced this inhibition when attempting in-tank stabilization of certain wastes (Agnew, 1997). It is important for DOE to perform verification testing for each waste type prior to producing different secondary wasteforms to ensure that the waste will be stabilized in a solid physical form (Recommendation #28). A fully-cured grout will result in a durable wasteform that has incorporated waste into a solid physical form.

5.1.3 NRC Conclusions on Waste Being Incorporated in a Solid Physical Form

The NRC staff reviewed the information DOE provided and determined that the wastes generated by the production of VLAW will be incorporated into a solid physical form. The NRC staff has the following recommendation on this topic:

- DOE should perform verification testing for each waste type prior to producing different secondary wasteforms to ensure that the waste will be stabilized in a solid physical form (Recommendation #28).

5.2 Radionuclide Concentrations and Classification

DOE is not using alternative requirements for wastes generated by the VLAW process. DOE provided waste classification calculations using the volume and radioactivity of the wasteforms assessed in the PA calculations to demonstrate that the wasteforms would be Class C or below. This section discusses NRC staff's review of the information DOE submitted regarding radionuclide concentrations and waste classification.

5.2.1 DOE's Assessment of Radionuclide Concentrations and Classification

DOE provided an estimate of the radionuclide concentrations and resulting waste class for DFLAW (DOE, 2020a). DOE referenced Table 1 and Table 2 of 10 CFR 61.55 as providing the concentrations of radionuclides to determine if the waste was Class C or less. Because the waste contains a mixture of both short- and long-lived radionuclides, DOE followed the procedure outlined in 10 CFR 61.55.

DOE estimated the sum-of-fractions (SOF) for each DFLAW campaign decayed to January 1 of the year in which the glass would be made. The maximum value of the SOF was 0.208 for campaign 12 and was primarily the result of the long-lived radionuclides ^{99}Tc, ^{239}Pu and ^{241}Am. DOE provided the SOF for long-lived and short-lived radionuclides for the entire DFLAW phase in Table 7-7 of the report RPP-CALC-63643, Rev. 4 (RPP-CALC-63643, 2021). The estimated SOFs was 0.115. The total DFLAW radionuclide concentrations were used to calculate SOFs instead of individual campaign data.

DOE established maximum radionuclide concentrations for the pretreated feed for the LAW Vitrification Facility that will serve as the basis for glass radionuclide concentration limits. DOE stated that the limits will ensure that the radionuclide concentrations in the VLAW glass will be less than Class C limits.

5.2.2 NRC Review of Radionuclide Concentrations and Classification

The NRC staff reviewed the information DOE provided in the Draft WIR Evaluation for VLAW (April 2020) as well as the information provided in report RPP-CALC-63643, Rev. 4 completed in April 2021. The IDF PA was completed in 2018, well in advance of the draft WIR evaluation. Information associated with the amounts of radionuclides and which waste stream they are associated with has changed between 2018 and 2020.

DOE used the correct methodology to estimate the waste classification. DOE identified the long-lived radionuclides found in Table 1 of 10 CFR 61.55 and the short-lived radionuclides found in Table 2 of 10 CFR 61.55. DOE first estimated the SOF associated with the long-lived radionuclides for DFLAW, determining that the SOF would be less than 1.0. Then DOE determined that the SOF associated with the short-lived radionuclides would be less than 1.0. The result was that the maximum DFLAW campaign had a SOF of 0.208 and the SOF averaged over all DFLAW was 0.115. The NRC staff used the information provided by DOE to verify the SOF calculations, using the inventory and volume numbers provided in Table 3-26 and 3-27 of the IDF PA (RPP-RPT-55858, 2018). Using those values and the glass density provided in the report, the NRC staff estimated that the SOF would be less than 1.0 but greater than 0.1, and therefore, the DFLAW would be Class C and suitable for disposal from a waste classification standpoint. In the report RPP-CALC-63643, Rev. 4, the amounts of key radionuclides present in VLAW glass (column 7 of Table 7-6) are much lower than found in the PA document or the PA model (RPP-RPT-59958, 2018). The NRC staff believes that column 7 is reflecting the total amount found in the DFLAW portion of the LAW glass production process and not the total amount found in VLAW glass. The SOF associated with the post-DFLAW glass production will be higher than the DFLAW portion.

The NRC staff also estimated the SOF associated with other wastes that will be generated as part of the LAW process. The SOF for all non-glass waste streams was less than 1.0 on a total waste volume and radionuclide content basis. The NRC staff did not have the information to examine the SOF on a batch or time series basis. The most problematic waste stream appears to be the silver mordenite, which had a SOF of 0.9 from ^{129}I. Batch-to-batch variability would likely result in some of this waste stream that would have a SOF greater than 1.0. In response to the NRC staff's RAI, DOE stated that the retention of ^{129}I in glass would be higher than previously estimated (which would reduce the amount in silver mordenite) (DOE, 2021c).

The NRC staff had technical observations associated with DOE's higher retention estimates. DOE did not provide estimates of the SOF for all waste streams resulting from the production of DFLAW or more generally LAW. It is recommended that when the inventory and final volumes of DFLAW are determined, DOE should revise their SOF calculations (Recommendation #29). It is also recommended that DOE provide to stakeholders the SOF for all waste streams generated by the LAW process (Recommendation #30).

5.2.3 NRC Conclusions on Radionuclide Concentrations and Classification

The NRC staff has the following conclusions on the determination of radionuclide concentrations and classification of the waste:

- DOE properly applied the methodology in NRC regulations for determining waste classification.
- DOE demonstrated that the SOF for DFLAW would be greater than 0.1 and less than 1.0, and therefore, the waste would be Class C.
- The DFLAW glass is suitable for near-surface disposal from the perspective of waste classification (other performance objectives apply).

The NRC staff has the following recommendations for determining radionuclide concentrations and classification:

- It is recommended that when the inventory and final volumes of DFLAW are determined, DOE should revise their SOF calculations (Recommendation #29).
- It is also recommended that DOE provide to stakeholders the SOF for all waste streams generated by the LAW process (Recommendation #30).

5.3 NRC Conclusions for Criterion C

The NRC staff reviewed DOE's Draft WIR Evaluation for VLAW, the IDF PA, additional references, the PA model, and had technical discussions with DOE staff and contractors on the determination of radionuclide concentrations and classification of the waste. Where necessary, the NRC staff performed independent calculations and analyses. Collectively, this information provides DOE's demonstration that Criterion C of DOE Order 435.1 would be met with reasonable expectation.

The results of the NRC staff's review related to Criterion C are as follows:

- DOE has demonstrated that the DFLAW wastes will be incorporated into a solid physical form.
- DOE has demonstrated that the DFLAW glass will be Class C waste, and therefore, is suitable for near-surface disposal (assuming other performance objectives are met).
- DOE used the correct methodology for determining SOF.

The recommendations provided in Section 5 are collated in Table 6-1. Table 6-1 identifies three categories of recommendations: (1) "DFLAW" means applicable to the Direct-Feed LAW that is evaluated in the Draft WIR Evaluation for VLAW; (2) "VLAW" means applicable to the remainder of the vitrified LAW or secondary wastes; and (3) "General" means if completed can improve the technical basis for DFLAW, VLAW, and/or future waste evaluations and is considered a best practice for performing waste evaluations. The recommendations from the WMA C TER that are applicable to VLAW are included in Appendix A.

6 OVERALL NRC REVIEW RESULTS AND CONCLUSIONS

The NRC staff completed a risk-informed, performance-based review of the Draft WIR Evaluation for VLAW, the IDF PA, and supporting documents and information. The Hanford Site and the residual waste in the underground tanks are very complex. Because the site has been operated for more than 80 years, there is a tremendous amount of information documenting past operations, technical assessments, field and experimental studies, and operational events. The results and conclusions in this TER are based on the information the NRC staff considered using a standard of reasonable assurance and comparison to the criteria in DOE Manual 435.1-1.

Overall, DOE demonstrated numerous good practices and adequate quality assurance. The NRC staff found that a substantial fraction of the PA documentation of the analyses and supporting contractor reports were clear, detailed, and traceable, with a few exceptions. In the Draft WIR Evaluation for VLAW, DOE evaluated the portion of the VLAW called DFLAW. However, the NRC staff believes the intent of the incidental waste process includes that all waste streams resulting from glass processing that can produce significant contribution to doses should be within the scope of the WIR evaluation. Therefore, the NRC staff review in this TER includes all the wastes associated with VLAW that will be disposed in the IDF at Hanford, to assess the cumulative impacts from disposal. These wastes include DFLAW, the remainder of the vitrified LAW, as well as other secondary wastes intended for disposal at the IDF. DOE evaluated the cumulative impacts of all these VLAW wastes in the IDF PA. The NRC staff provides conclusions for the DFLAW portion of the waste evaluated against DOE Manual 435.1-1 criteria, as well as observations about the remainder of the VLAW and secondary wastes that will be produced.

The NRC staff has the following general conclusions, followed by specific conclusions and recommendations for DFLAW and non-DFLAW wastes:

- DOE demonstrated numerous good practices and acceptable quality assurance in its clearly documented analyses.

- Limiting the scope of the evaluation to a particular wasteform (glass) resulting from waste processing adds uncertainty as to whether the cumulative impacts of the wastes that will be disposed at the IDF are acceptable. All waste streams resulting from processing that could produce a significant contribution to projected doses that were included within the IDF PA should be within the scope of the WIR evaluation. Alternatively, DOE should develop a separate evaluation that documents DOE's decision that all wastes disposed in the IDF meet the WIR criteria from DOE Order 435.1, with an indication of the point in the waste process flow that this determination is made.

- DOE did not adequately support the high retention values of ^{99}Tc and ^{129}I in glass provided in its response to the NRC staff's request for additional information (NRC, 2020a).

- The lack of verification plans for production-scale wasteforms is a large uncertainty given the novel wastes and recycling of off-gas used during vitrification. The NRC staff recommends that DOE collect operational data to verify that the actual wasteform performance is consistent with the wasteform performance in the Draft WIR Evaluation for VLAW.

DOE requested that the NRC staff provide conclusions for the DFLAW portion of the vitrified waste with respect to meeting DOE Manual 435.1-1 Criteria. For DFLAW, the NRC staff concludes:

- DOE demonstrated that the waste has been processed or will be processed to remove key radionuclides to the maximum extent that is technically and economically practical (Criterion A).
- DOE demonstrated that the waste will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C (Criterion B).
- DOE demonstrated that the waste will 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 the DOE Radioactive Waste Management Manual. The waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR 61.55 (Criterion C).

The following assumptions apply to the NRC staff's conclusions:

- DOE will produce wasteforms during operations that are of similar performance and characteristics to those currently estimated.
- Recycle of off-gases will not result in the buildup of deleterious species that significantly impacts glass performance.
- Cracking and the availability of cracked glass surface area for release will be comparable between production and surrogate data.
- DOE will achieve greater than 95 percent retention of ^{99}Tc and ^{129}I in glass using recycle.
- DOE will install a thick engineered cover with adequate erosion protection.

DOE included all low-activity wastes in the IDF PA analyses (RPP-RPT-59958, 2018) and included the combined doses from all low-activity wastes in the Draft WIR Evaluation for VLAW. DOE did not specifically identify the dose impacts resulting from the DFLAW portion of the waste, however, DOE provided sufficient information in the IDF PA to allow the NRC staff to understand the contributions to doses resulting from disposal of each of the low-activity wastes in the IDF, including DFLAW. DOE did not provide information associated with Criterion C for the non-DFLAW portions of the waste.

The NRC staff has the following observations associated with the non-DFLAW portion of the wastes:

- Select secondary wastes could be produced that exceed the Class C concentration limits (e.g., granular activated carbon (GAC)). These wastes would not meet Criterion C unless alternate criteria were applied.
- The remainder of the vitrified waste is likely to meet Criteria A, B, and C but the demonstration is not complete without DOE's demonstration that acceptable wasteforms can be made that account for the differences in the tank waste (e.g., soluble Sr, organics, minor species that buildup during recycle).
- The secondary wasteforms are under development, therefore, their performance is more uncertain. The risk-significance of the secondary wasteforms will be determined by the actual retention rates of volatile species in glass experienced after production begins.
- Waste variability may require the use of a longer institutional control period or the use of other mitigation actions to ensure the protection of inadvertent intruders.
- Impacts to water and air pathways during DOE's 1,000-year compliance period are likely to be well below established limits for all wastes (vitrified and non-vitrified).

The NRC staff has provided recommendations throughout this report for Criteria A, B, and C; These recommendations do not supersede the previous recommendations made for WMA C. Table 6-1 provides a listing of the recommendations in this TER and identifies three categories: (1) "DFLAW" means applicable to the Direct-Feed LAW that is evaluated in the Draft WIR Evaluation for VLAW; (2) "VLAW" means applicable to the remainder of the vitrified LAW or secondary wastes; and (3) "General" means if completed can improve the technical basis for DFLAW, VLAW, and/or future waste evaluations and is considered a best practice for performing waste evaluations. The technical recommendations designated as "VLAW" could be added to PA maintenance activities. The NRC staff believes they are collectively unlikely to impact DOE's demonstration that the criteria have been met. However, there is always the potential for unforeseen changes to information or understanding and implementing the recommendations provided may help to mitigate those uncertainties.

Appendix A provides the recommendations from the review of WMA C that were previously provided to DOE (NRC, 2020b) that are applicable to VLAW. DOE was unable to address those recommendations in the IDF PA because the analyses were completed prior to NRC issuing the WMA C TER. DOE should consider recommendations made in the NRC review of WMA C for applicability to VLAW (Recommendation #31).

The results of NRC staff's review of Criteria A, B, and C in this TER are being provided to DOE for consideration. NRC has no regulatory authority related to DOE's waste determination activities. DOE has stated it will consider the information in the NRC staff's TER and the comments from stakeholders before finalizing the WIR evaluation for VLAW (or DFLAW), which will contain the final waste determination of whether DFLAW can be managed as LLW. DOE does not have an obligation to implement recommendations provided by the NRC.

Table 6-1 Recommendations for DOE Based on NRC's Review of the Draft WIR Evaluation for VLAW

Number	Recommendation	Section	Applicability
1	Formally assess the separation of volatile species and disposition as HLW from a cost-benefit and risk reduction perspective. Because vitrification very effectively removes the two drivers of groundwater pathways doses from the waste, and off-gas systems exist to capture those radionuclides, disposition as HLW may decrease the future risks from the IDF.	3.2.2 – Alternative Treatment Technologies	DFLAW and VLAW
2	Continue to invest in research such as long-term field studies of isotopic migration to quantify long-term infiltration rates that may apply to areas of disturbance and coarse-grained soils (e.g., dunes).	4.4 – Infiltration	General
3	Complete barrier flow analyses with stochastically (geostatistically) generated material properties at finer scales including spatial variability for a given material type.	4.5.2 – Near-Field Hydrology	General
4	Evaluate if the numerical grid used for modeling near-field hydrology was sufficient and eliminate artifacts associated with coarse discretization of numerical models.	4.5.2 – Near-Field Hydrology	General
5	Complete laboratory measurements of flow through cracked glass surrounded by porous material to provide model support.	4.5.2 – Near-Field Hydrology	VLAW
6	Determine the amount of cracking for VLAW glass from analyses of samples of appropriate scale and composition using the cooling profile specification that will be used in the design.	4.6.2.1 – Glass Cracking	VLAW
7	Improve the basis for the retention of volatile species in VLAW glass and use more conservative values for retention of volatile species in glass in the base case evaluation until additional information is developed to support higher retention values.	4.6.2.3 – Retention and Recycle of Volatile Species	DFLAW and VLAW

Number	Recommendation	Section	Applicability
8	Compare the range of waste compositions considered for glass durability testing against the wastes present in the tank farms and the wastes expected to exit the pre-treatment tank (AP-106), evaluate whether the previous tests adequately considered the range of compositions expected in the waste stream, and perform additional glass durability tests as needed. It is also recommended that DOE assess the buildup of minor species resulting from recycling and include the resultant phases in glass durability testing.	4.6.2.4 – Glass and Waste Compositions	DFLAW and VLAW
9	Cold glass production-scale samples should be non-destructively or destructively examined to determine the amount of cracking.	4.6.2.4 – Glass and Waste Compositions	DFLAW and VLAW
10	Perform additional analyses to investigate the potential bias in results of glass release modeling as a result of using a coarse numerical grid.	4.6.2.5 – Modeling of Glass Performance	General
11	Provide an appendix to the PA that explicitly identifies the parameters used in the base case and other key cases to improve the transparency and usability of the PA and to facilitate future independent review.	4.7.2 – Analyses of Non-Glass Wasteform Performance	General
12	Determine the performance of non-glass wasteforms over the full range of waste variability (including minor species and organics) using testing over extended timeframes or with proper acceleration.	4.7.2.2 – Models Used to Calculate Release from Cementitious Wasteforms	General
13	Ensure that the WAC for the SSW and the cementitious wasteforms is consistent with the performance observed over the full range of testing.	4.7.2.3 – Parameters Assumed in Cementitious Wasteform Release Calculations	General
14	Perform verification testing of the cementitious wasteforms at the beginning of operations and after any significant process changes occur.	4.7.2.3 – Parameters Assumed in Cementitious Wasteform Release Calculations	General
15	Account for waste composition variability and process scale-up when developing specifications for non-glass wasteforms.	4.7.2.3 – Parameters Assumed in Cementitious Wasteform Release Calculations	General

Number	Recommendation	Section	Applicability
16	Identify waste emplacement distributions that will minimize impacts and utilize those for disposal operations	4.8.2 – Flow and Transport in the Unsaturated Zone	General
17	Further examine the geologic uncertainty associated with the Ringold E Unit. If characterization data is not sufficient to make a confident interpretation, then a pessimistic representation should be taken.	4.9.2 – Flow and Transport in the Saturated Zone	General
18	Provide a more complete analysis of the potential impact of the buildup of ^{99}Tc in the biosphere.	4.10.2 – Biosphere and Dose Assessment	General
19	DOE should clarify in the final WIR evaluation what institutional control period is being used. If a 100-year institutional control period is used, the waste variability analyses provided in DOE's RAI response demonstrates that the waste acceptance criteria (concentrations) would likely be exceeded unless other mitigating factors or approaches are considered. DOE should consider waste variability when establishing waste acceptance criteria.	4.11.2 – Inadvertent Human Intrusion	General
20	DOE should evaluate the potential impacts to inadvertent intruders of different concentrations of radioactivity in different types of SSW rather than only considering an average over all SSW.	4.11.2 – Inadvertent Human Intrusion	VLAW
21	Because of the importance of model support to the decision-making process, a dedicated plan, strategy, and document summarizing model support for the IDF PA could enhance confidence that the numerical models adequately project or bound future impacts.	4.12.2 – Model Support	General
22	DOE should compile all uncertainties that were "dropped" from further consideration based on local sensitivity analyses and include them in a global uncertainty analysis.	4.13.2 - Uncertainty	VLAW
23	DOE should place more emphasis on uncertainty analyses using the system model rather than local sensitivity analyses.	4.13.2 - Uncertainty	General
24	Based on present information and independent interpretation of the data and uncertainties, NRC estimates that retention of ^{99}Tc and ^{129}I in glass could be as low as 90% or greater than 99%. It is recommended that this uncertainty should be part of system model uncertainty analyses.	4.13.2 - Uncertainty	VLAW

Number	Recommendation	Section	Applicability
25	The hydraulic properties and other uncertainties associated with the flow of water through the glass wasteform for release modeling should be included in uncertainty analyses with the system model.	4.13.2 - Uncertainty	VLAW
26	DOE should include all significant uncertainties in uncertainty analyses for the inadvertent intruder.	4.15.2 – Demonstration of Compliance with the Performance Objectives	VLAW
27	DOE should quantify the impacts associated with each waste type that will be generated from the production of VLAW (e.g., SSW, GAC, HEPA filters, AgM, waste generated by decommissioning the facility).	4.15.2 – Demonstration of Compliance with the Performance Objectives	DFLAW and VLAW
28	DOE should perform verification testing for each waste type prior to producing different secondary wasteforms to ensure that the waste will be stabilized in a solid physical form.	5.1.2 – Waste Being Incorporated in a Solid Physical Form	DFLAW and VLAW
29	DOE should revise their SOF calculations when the inventory and final volumes of DFLAW are determined.	5.2.2 – Radionuclide Concentrations and Classification	DFLAW
30	DOE should provide to stakeholders the SOF for all waste streams generated by the LAW process.	5.2.2– Radionuclide Concentrations and Classification	VLAW
31	DOE should consider recommendations made in the NRC review of WMA C for applicability to VLAW. These recommendations are presented in Appendix A of this report.	All	DFLAW and VLAW

[#] *DFLAW* means applicable to the Direct-Feed LAW that is evaluated in the Draft WIR Evaluation for VLAW (shaded cells); *VLAW* means applicable to the remainder of the vitrified LAW or secondary wastes; *General* means if completed can improve the technical basis for DFLAW, VLAW, and/or future waste evaluations and is considered a best practice for performing waste evaluations.

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APPENDIX A APPLICABILITY OF NRC'S RECOMMENDATIONS FOR WMA C TO VLAW

Table A-1 provides the recommendations from the review of WMA C that were previously provided to DOE (NRC, 2020b) that the NRC staff have determined are likely to also be applicable to VLAW. DOE was unable to address those recommendations in the performance assessment for the IDF because the performance assessment was completed prior to NRC's comments. The NRC staff have shaded recommendations in Table A-1 that the staff believes are most risk-significant to VLAW.

Table A-1 Recommendations from NRC's Review of the Draft WIR Evaluation for WMA C that are Applicable to VLAW

Number	Recommendation	WMA C Section
1[a]	The isotopes ^{242}Cm and ^{94}Nb should be added as key radionuclides.	2.1 – Identification of Key Radionuclides
5	Alternative technologies should be assessed on a regular basis and DOE should examine technologies both within and external to the DOE.	2.2 – Removal to the Maximum Extent Practical
8	DOE should follow guidance for DOE Order 435.1 when evaluating potential peak dose impacts.	3.1 – Assessment Context
9	The approach to scenario and conceptual model development should identify significant interdependencies and interrelationships between FEPs that could result in plausible alternative future scenarios or alternative conceptual models. From the "future evaluations" recommendations, this is the most risk-significant.	3.2 – Future Scenarios and Conceptual Models
11	The uncertainty associated with future climates and the uncertainty in processes that climate affects (e.g., recharge rates) should be part of the scenario or conceptual model development.	3.3 – Current Climate and Recharge
12	The full range of uncertainty associated with long-term transient ecosystems at the Hanford Site should be discussed, including trends in invasive species encroachment and development.	3.3 – Current Climate and Recharge
13	The range for recharge rates applied to long-term, non-disturbed conditions should be expanded.	3.3 – Current Climate and Recharge

Number	Recommendation	WMA C Section
14	Information should be developed associated with rate at which a disturbed area revegetates and the impact on recharge rates, especially for extremely disturbed areas. It should be determined if revegetated areas have the same recharge rate as undisturbed areas with natural soil properties.	3.3 – Current Climate and Recharge
15	The effects of a transient ecosystem at the Hanford Site where big sagebrush is not the dominant fauna on estimated recharge rates should be evaluated.	3.3 – Current Climate and Recharge
16	The design criteria for the main component of the cover, the side slopes, and the toe of the side slopes should consider the methodologies and approaches found in NRC's NUREG-1623 (2002), or DOE should develop guidance on long-term erosion protection design.	3.4 – Engineered Barrier System
17[b]	An analysis should be completed to determine the PMP of the relative area and align the intended surface cover design for the C-Tank Farm with the results of the analysis. If DOE elects to take less credit for the engineered cover, then a less robust design may be appropriate.	3.4 – Engineered Barrier System
18[b]	From an infiltration standpoint, final design of the engineered surface cover should be risk-informed and consistent with the necessary performance to limit infiltration. The design should consider degradation of asphalt if asphalt is included as part of the surface barrier design, and technical bases for infiltration rates through side slopes should be provided.	3.4 – Engineered Barrier System
23	The HDW model should not be used to develop inventory estimates unless much broader uncertainty ranges are applied and if verification and validation activities are completed.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport
24	Greater transparency should be provided as to the source of the inventory information such that the assigned uncertainty ranges can be better understood and evaluated.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport
25	Uncertainty in the radiological inventory should be expanded. The uncertainty in analytical methods should be included.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport

Number	Recommendation	WMA C Section
28	Comparisons between data, process models, and the performance assessment model in the area of source-term release implementation is a good practice that should be implemented more regularly in future performance assessments.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport
31	Henry's Law constants should be set to expected values in base case calculations for the water pathway.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport
34	Broader ranges of values should be used for unsaturated hydraulic parameters. DOE should not truncate the probability distributions at the minimum and maximum values of the observed data.	3.6 – Flow and Transport in the Unsaturated Zone
35	The uncertainty ranges for K_d values for ^{79}Se, ^{129}I, ^{226}Ra, and the plutonium isotopes should be expanded.	3.6 – Flow and Transport in the Unsaturated Zone
36	Documentation of the CPGWM development, including model objectives, conceptualization, implementation, and application, should be integrated within the PA documentation. DOE should discuss limitations of the model results that can have a direct bearing on the use of the model to obtain concentration and dose results.	3.7 – Flow and Transport in the Saturated Zone
37[c]	A stronger technical basis for the saturated hydraulic conductivity value range should be provided. In addition, stronger technical bases should be provided for the single values used for both the hydraulic gradient and for the longitudinal field-scale dispersivity.	3.7 – Flow and Transport in the Saturated Zone
38[c]	The range of values used for flow velocity or Darcy flux and the longitudinal field- scale dispersivity in the sensitivity and barrier importance analyses should be expanded to encompass the full range of uncertainty associated with those parameters.	3.7 – Flow and Transport in the Saturated Zone
39	The location and width of the stream tube segments should be analyzed for their influence on the results.	3.7 – Flow and Transport in the Saturated Zone
40	Saturated zone thickness should be part of a sensitivity analysis.	3.7 – Flow and Transport in the Saturated Zone

Number	Recommendation	WMA C Section
41	Models used to simulate the release and transport of radionuclides should be consistent with the assumptions about the biosphere.	3.8 – Biosphere Characteristics and Dose Assessment
42	The dose results for Native American receptors at Hanford should be provided to increase transparency with potentially impacted stakeholder groups.	3.8 – Biosphere Characteristics and Dose Assessment
44	The mixing assumptions associated with drill cuttings should be reconsidered to ensure the assumed mixing depths are consistent with projected land use for the chronic intruder scenarios.	3.10 – Inadvertent Human Intrusion
45	Consistent approaches to fruit and vegetable ingestion should be used for the onsite and offsite receptors.	3.10 – Inadvertent Human Intrusion
46	Site-specific values for biosphere parameters should be used when available.	3.10 – Inadvertent Human Intrusion
47	Measurements of mass loading values that can be assigned to an acute intruder (well driller) should be completed.	3.10 – Inadvertent Human Intrusion
48	Radon should be included with the dose impacts to the inadvertent intruder.	3.10 – Inadvertent Human Intrusion
52[d]	Model support should be improved. Modeling should be performed to demonstrate that the PA model can reproduce the real-world observation associated with in-leakage to tanks.	3.13 – Model Support
53	The approach to uncertainty assessment should be iterative or include most parameters as uncertain in the assessment. More parameters should be uncertain in an initial uncertainty assessment and then can be eliminated in a final uncertainty assessment if they are found to be insignificant.	3.14 - Uncertainty
54	In future uncertainty assessments it should be ensured that the tails of the distributions are not truncated.	3.14 - Uncertainty

Number	Recommendation	WMA C Section
55	Methods for sensitivity and uncertainty analyses that globally evaluate uncertainties in a risk-informed context should be used. In a system model with numerous uncertainties, the impact of those uncertainties cannot be determined with one-at-a-time evaluations; it is conceptually flawed to use one-at-a-time evaluations and should not be used in future waste evaluations to evaluate the impact of uncertainties.	3.14 - Uncertainty
56	Plausible uncertainties should be included in the probabilistic system model (or through some other method if the global impact of all types of uncertainties are communicated in the results).	3.14 - Uncertainty

[a] DOE provided information in the response to NRC's RAI to address ^{94}Nb.

[b] Based on the credit taken for the surface cover, not likely to be risk-significant for VLAW.

[c] Addressed in RAI responses.

[d] Model support should be improved but in-leakage to tanks is not applicable.